GE Nuclear Energy

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Licensing Topical Report

BWR Owners' Group Long-Term Stability Solutions Licensing Methodology

GE Nuclear Energy

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GE NUCLEAR ENERGY

LICENSING TOPICAL REPORT

BWR OWNERS' GROUP

LONG TERM STABILITY SOLUTIONS LICENSING METHODOLOGY

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ABSTRACT

Compliance with stability licensing criteria set forth in 10CFR50 Appendix A, General Design Criterion (GDC-12), is achieved by either preventing stability-related neutron flux oscillations or detecting and suppressing the oscillations prior to exceeding specified acceptable fuel design limits. The BWR Owners' Group (BWROG) has developed long-term solutions which incorporate either prevention or detection and suppression features, or use a combination of both features to ensure compliance with GDC-12. Methodologies have been developed to support the licensing of these long-term solutions.

For prevention features, a methodology has been developed which determines the region of the operating domain which may be susceptible to instabilities. The method uses conservative inputs, along with criteria which account for uncertainties in the model to ensure a low probability of an oscillation occurring outside the defined region. This methodology can be used to define an exclusion region in which operation will be precluded by either administrative controls or by automatic actions.

Detection and suppression solutions use the Local Power Range Monitors (LPRMs) as the basic detection devices, and differ primarily in the way in which LPRM signals are combined and evaluated to detect the presence of an oscillation. A methodology has been developed that relates the fuel thermal response in the limiting fuel assembly to the LPRM signals. This methodology can be used to confirm that a specific detection and suppression solution provides protection of the Minimum Critical Power Ratio (MCPR) Safety Limit. Examples of solution options that are being evaluated by the BWR Owners' Group are provided to demonstrate the application of the described methodologies. These examples provide sufficient detail to allow generic approval of the methodologies so that final hardware and software designs can be performed with an established analytical basis.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 12, 1993

L. A. England, Chairman
BWR Owners' Group
Gulf States Utilities
River Bend Station
North Access Rd.
St. Francesville, LA 70775

Dear Mr. England:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORTS NEDO-31960 AND NEDO-31960 SUPPLEMENT 1, "BWR OWNERS GROUP LONG-TERM STABILITY SOLUTIONS LICENSING METHODOLOGY" (TAC NO. M75928)

The staff has completed its review of the Topical Reports NEDO-31960 and NEDO-31960 Supplement 1," submitted by the BWR Owners Group (BWROG) by letters dated May 31, 1991 and March 16, 1992. These reports describe and justify the use of several BWROG developed long-term solutions to BWR stability issues and the methodologies developed for evaluating appropriate setpoints and performance criteria for these solutions.

We find the solutions of NEDO-31960 and Supplement 1 to be acceptable for referencing in license applications to the extent specified, and under the limitations delineated in NEDO-31960 and Supplement 1 and the associated NRC technical evaluation. The enclosed safety evaluation defines the basis for acceptance of these topical reports.

We do not intend to repeat our review of the matters found acceptable as described in NEDO-31960 and Supplement 1 when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the application of NEDO-31960 and Supplement 1.

In accordance with procedures established in NUREG-0390, it is requested that the BWROG publish accepted versions of this topical report, within three months of receipt of this letter. The accepted versions shall include an "A" (designating accepted) following the report identification symbol.

Should our criteria or regulations change so as to invalidate our conclusions concerning the acceptability of the report, the BWROG or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Ashok C. Thadani, Director
Division of Systems Safety and Analysis

Enclosure:
NEDO-31960 Evaluation

ENCLOSURE

SAFETY EVALUATION REPORT ON "BWR OWNERS' GROUP LONG-TERM STABILITY SOLUTIONS LICENSING METHODOLOGY" NEDO-31960 AND SUPPLEMENT 1

1 INTRODUCTION

The Boiling Water Reactor Owners' Group (BWROG) has submitted to the U.S. Nuclear Regulatory Commission (NRC) the Topical Report NEDO-31960, "Long-Term Stability Solutions Licensing Methodology," (Ref. 1) and Supplement 1 (Ref. 2) for staff review. The long-term solutions described in these reports consist of conceptual designs for automatic protection systems developed by the BWROG with its contractor, General Electric Company (GE), to either prevent stability-related neutron flux oscillations or to detect and suppress any oscillations that may occur. The reports also describe the methodologies that have been developed to establish setpoints and demonstrate the adequacy of the protection systems to prevent violation of the critical power ratio (CPR) safety limits in compliance with General Design Criteria (GDC) 10 and 12 in Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). The BWROG has requested that the NRC staff review these reports and approve the following:

* The overall regional exclusion and detection and suppression methodology, including the overall treatment of uncertainties.

* The solution concepts and associated licensing approach for each option.

* The application of the methodology to the solution concepts.

* The BWROG/GE position concerning safety classification of new and existing hardware; that is, the safety classification of all existing and interfacing equipment should not change when new or modified stability long-term solution hardware is installed.

The BWROG plans to proceed with the selection of options and specific hardware design based on NRC's approval of the proposed concepts and associated methodologies.

2 EVALUATION

The solution options proposed by the BWROG in NEDO-31960 are:

I <u>Exclusion Region</u> A region in the high-power/low-flow area of the power/flow map outside of which instabilities are very unlikely is calculated for each representative BWR type using well-defined procedures. If the reactor is operated within this exclusion region, an automatic protective action is initiated to exit the region. This action is based exclusively on power and flow measurements; the presence of oscillations is not required for its initiation. Four solutions for the Type I option have been proposed by the BWROG, although not all have been completely developed:

I-A Immediate protective action is taken when entering the exclusion region. This action can be either a scram or a select rod insert (SRI).

I-B Same as I-A, but the protective action when entering the exclusion region can be bypassed if a stability monitor is operational and detecting sufficiently stable conditions (e.g., a decay ratio less than 0.6).

I-C Protective action is taken if the following two conditions both exist: (1) the reactor is operating inside the exclusion region and (2) an average power range monitor (APRM) oscillation (of small magnitude) is detected.

I-D A few small-core plants with tight inlet orifices have a reduced likelihood of out-of-phase instabilities. For these plants, the existing unfiltered, flow-biased APRM scram provides sufficient protection. In addition, administrative controls are proposed to maintain the reactor outside the exclusion region.

II <u>Quadrant-Based APRM Scram</u> In a BWR/2, the quadrant-based APRM is capable of detecting both in-phase and out-of-phase oscillations with sufficient sensitivity to initiate automatic protective action to suppress the oscillations before safety margins are compromised.

III <u>LPRM-Based Detect and Suppress</u>. Local power range monitor (LPRM) signals or combinations of a small number of LPRMs are analyzed on line by using three diverse algorithms. If any of the algorithms detect an instability, automatic protective action is taken to suppress the oscillations before safety margins are compromised. Two options have been proposed by the BWROG: Option III and Option III-A. The main difference between the two is in the hardware implementation. Option III requires a new Class 1E computerized system. For Option III-A, newly designed digital replacements of the existing APRM amplifier cards will be used and a smaller number of LPRM detectors in a revised configuration will be required. Conceptually, the algorithms are similar in both solutions.

The NRC contractor, Oak Ridge National Laboratory (ORNL), assisted the staff in reviewing the topical reports. ORNL has provided a technical evaluation report (TER) that is included as Attachment 1. The TER describes the results of the staff's review of the functional performance criteria for the proposed protection systems and of the assumptions, principles, and models inherent in the methodologies used to define protection system stability boundaries and setpoints. The staff's evaluation of hardware safety classification follows.

At a meeting on March 26, 1992, with the NRC staff, the BWROG proposed that for the long-term stability solution options relying on "APRM flow biased scram" recirculation drive flow signals, the use of existing hardware be allowed in the new protection system. The recirculation flow drive system, although highly reliable, is not designed to Class 1E standards. The staff, therefore, requested additional information on plant-specific arrangements of the existing recirculation drive flow instrument channels, channel integrity and independence, the failure rate data for each component in the flow channels, and the failure indication alarms. The response, which the BWROG transmitted with a July 17, 1992, letter (Ref. 3) provides the results of a survey among 9 licensees for 12 operating plants. In general, redundant flow channels exist in these arrangements. The failure history of the channel components (from eight BWR units covering 84 reactor-years) shows the failures to be random and the failure rate to be insignificant. For failure indication, the output signal from a flow channel is compared to the output signal from another flow unit. The comparator activates an alarm when two flow signals differ more than the specified tolerance. Alarms also are activated when the comparator fails high or low. Isolators are provided between flow units and between the comparator and the APRM circuitry and the alarm circuitry. However, the survey results indicate that many operating plants do not meet the configuration in BWROG

Viewgraph 3/26-7, "Drive Flow Signal Path," which was shown at the meeting on March 26, 1992.

The staff will review the hardware design details on a plant-specific basis. In general, it finds the proposed concept to be acceptable, but may require modifications for some plants.

3 CONCLUSIONS

The staff has reviewed the licensing basis for long-term solutions to BWR stability proposed by the BWROG and adopts the recommendations described in the attached TER. The regulatory positions with respect to the specific approvals requested by the BWROG are summarized below:

(1) Methodology

The exclusion region calculation methodology described in NEDO-31960 and its Supplement 1 is acceptable for defining the Option I-A exclusion region and the Options III and III-A exclusion boundaries outside of which the detect and suppress action may be deactivated. The overall treatment of uncertainties is acceptable for the selection of initial conditions and for the selection of oscillation contours and the treatment of failed LPRM sensors for Options III and III-A. The methodology is acceptable for evaluating the protection provided by the Option II quadrant-based APRM scram. Specific procedures for application of the methodology consistent with documentation and calculations submitted for this review should be developed and documented by BWROG.

(2) Solution Concepts

(a) Options I-B and I-C have not been developed in

detail by the BWROG and, therefore, will not be considered acceptable as long-term solutions until fully developed by the BWROG and reviewed and approved by the staff. Option I-D is still under review and its acceptability as a long-term solution depends, to a large degree, on the details of calculations that are not yet available. Attachment 1 identifies some concerns about the Option I-D reliance on predictive calculations to conclude that the out-of-phase mode of oscillation will be avoided. To address these concerns and to provide reasonable assurance that out-of-phase oscillations will be avoided by I-D plants, it may be necessary to incorporate strict operational controls on axial and radial power distribution and to enhance the capability to recognize operating conditions that are approaching instability by other means such as on-line stability monitoring. Core stability sensitivities are illustrated by experience with the instability event on August 15, 1992, at Washington Nuclear Power Unit 2, in which oscillations developed outside of the stability exclusion regions because of a combination of fuel, core design and control rod patterns which resulted in conditions unfavorable to core stability, and conditions that an NRC inspection team concluded to be vulnerable to out-of-phase instability (Ref. 5). The staff will evaluate the acceptability of Option I-D when the calculations for the lead plant are submitted. If the lead plant analyses are acceptable, the staff will evaluate detailed calculations for all plants that may propose Option I-D (e.g. Duane Arnold, Vermont Yankee, Monticello, and FitzPatrick). If individual plant analyses are inconclusive because of large uncertainties involving assumed operating

conditions, the quality of administrative controls and available core monitoring to reduce instability vulnerabilities will be considered in evaluating the Option 1-D acceptability for a specific plant.

(b) The implementation of Option I-A is an acceptable long-term solution for any type of BWR, subject to the following conditions:

(i) Specific reload confirmation procedures should be developed so that for every reload, the licensee can either confirm the applicability of old exclusion region settings or set a new exclusion region boundary.

(ii) The exclusion boundary setpoints for this option should be sufficiently bounding to avoid changes on a cycle-by-cycle basis. Major setpoint changes should be expected only if the fuel design changes significantly.

(iii) When establishing reactor trip setpoints for the power/flow exclusion region scram, operational restrictions on other parameters important to stability (e.g., radial and axial power distribution during low flow power maneuvering) that are consistent with the assumptions of the exclusion boundary analyses should be addressed, including the need for technical specifications, and factored into the setpoint evaluation.

(iv) Select rod insert (SRI) may be used in conjunction with Option I-A, but a full scram should occur if the reactor does not exit the

region within a reasonable period of time (about a few seconds).

(c) Option II is an acceptable long-term solution for implementation in BWR/2s, which have quadrant-based APRM scrams. For implementing Option II, plant-specific analyses should show that the quadrant-based APRM scram provides sufficient protection against out-of-phase instability modes to avoid the violation of CPR safety limits.

(d) Options III and III-A are acceptable long-term solutions for implementation in any type of BWR, subject to the following conditions:

(i) All three algorithms described in NEDO-31960 and Supplement 1 should be used in Option III or III-A. These three algorithms are high LPRM oscillation amplitude, high-low detection algorithm, and period-based algorithm.

(ii) The validity of the scram setpoints selected should be demonstrated by analyses. These analyses may be performed for a generic representative plant when applicable, but should include an uncertainty treatment that accounts for the number of failed sensors permitted by the technical specifications of the plant's applicant.

(iii) Implementation of Option III or III-A will require that the selected bypass region outside of which the detect and suppress action is deactivated be defined in the technical specifications.

(iv) If the algorithms detect oscillations, an automatic protective action should be initiated. This action may be a full scram or an SRI. If an SRI is implemented with Option III or III-A, a backup full scram must take effect if the oscillations do not disappear in a reasonable period of time or if they reappear before control rod positions and operating conditions have been adjusted in accordance with appropriate procedural requirements to permit reset of the SRI protective action.

(v) The LPRM groupings defined in NEDO-31960 to provide input to the Option III or III-A algorithms are acceptable for the intended oscillation-detection function. These LPRM groupings are the oscillation power range monitor for Option III or the octant-based arrangements for Option III-A. The requirements for a minimum operable number of LPRM detectors set forth in NEDO-31960 are acceptable.

(e) Options I and II do not protect the fuel against single-channel instability, and the protection provided by Options III and III-A for single-channel instability is not highly reliable. When implementing the long term solution, a procedure to review the thermal hydraulic stability of lead use assemblies (LUA) in a core reload should be established. The review should ensure that inclusion of the LUA as proposed in the core reload is very unlikely to result in single-channel instability.

(3) Safety Classification

As a minimum, the recirculation drive flow channel should comply with the requirements of the Electrical and Electronics Engineers, Standard 279 (Ref. 4), which include the single-failure criterion, component quality, channel independence, and the capability for test and calibration. Isolation devices are required to be qualified for their application. No credible failure at the output of an isolation device should prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases. The plant-specific submittal should include the specification documentation for the isolation device. In addition, because Solution I-A involves an automatic reactor scram function, any modification to the reactor protection system trip function requires a submittal to the NRC proposing a change in the technical specifications. The plant-specific technical specification change should include limiting conditions for operation, action statements, allowable out-of-service times, surveillance tests, and test frequency commensurate with the importance to safety of the system. The detailed technical specification requirements should be addressed generically during review of the detailed hardware design.

4 REFERENCES

1. NEDO-31960, "BWR Owners' Group,Long-Term Stability Solutions Licensing Methodology," May 1991.

2. NEDO-31960, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," March 1992.

3. Letter from C. L. Tully (BWROG) to A. C. Thadani (NRC), "Response to RAI on Stability Report NEDO-31960," dated June 5, 1992," July 17, 1992.

4. Institute of Electrical and Electronics Engineers, Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

5. Letter from J. B. Martin, NRC, to A. L. Oxsen, Washington Public Power Supply System, "NRC Augmented Inspection of Washington Nuclear Project, Unit 2", September 29, 1992.

Attachment 1

ORNL/NRC/LTR-92/15

Contract Program:	Technical Support for the Reactor Systems Branch (L1697/P2)
Subject of Document:	Licensing Basis for Long-Term Solutions to BWR Stability Proposed by the BWR Owners' Group
Type of Document:	Technical Evaluation Report
Author:	José March-Leuba
Date of Document:	August 1992
NRC Monitor :	T. L. Huang, Office of Nuclear Reactor Regulation

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U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400

SUMMARY

This report documents the main conclusions and recommendations derived from our review of the Boiling Water Reactor Owners' Group (BWROG) long-term solutions of the stability issue as described in NEDO-31960 and its Supplement-1 (references 1 and 2). Overall, this review is very positive. Our main conclusion is that all three of the proposed solution types (I, II, and III) are technically sound and, in our opinion, any of them will solve the stability issue if properly applied.

Although not specifically related to the stability issue, the fact that implementation of a new reactor protection function will most probably result in an increased number of challenges to the reactor protection system may lead to a new safety problem unless the number of unnecessary challenges is minimized by design. It is recognized that the normal function of these solutions is to provide an automatic protection action (i.e., a scram or a runback) if either oscillations are detected or the exclusion region is entered; however, the implementation of this function must be performed carefully in order to minimize the number of unnecessary actuations while maintaining a very high probability to perform the intended safety function.

Detailed recommendations, including some qualifiers and reservations, are specified in the main text of this report. A condensed summary of these recommendations follows:

1. Approve the overall licensing methodology described in NEDO-31960 and its Supplement 1 for Solutions I-A, II, and III. This methodology includes the treatment of uncertainties and the selection of initial conditions and calculation parameters. The approval should be conditioned to assure plant-specific consistency with the axial (2.0) and the radial (end-of-cycle Haling) peaking factors assumed for the core power distribution calculation parameters.

2. Do not approve Solution concepts I-B and I-C because of its lack of detailed development and/or interest by the BWROG.

3. Do not approve Solution concept I-D at this time until the final evaluations that NRC has requested have been performed. This recommendation does not imply a rejection of Solution I-D; the approval of Solution I-D depends on the details of calculations that have not yet been performed by the BWROG.

4. A select rod insert (SRI) is an acceptable automatic protection action for any of the approved solutions (I-A, II, or III) as long as a full scram takes effect if either the oscillations do not disappear or the reactor does not exit the exclusion region within a reasonable period of time (a few seconds). The exclusion region must be examined prior to each plant operating cycle to assure consistency with the axial and radial power peaking distribution assumed in the exclusion region boundary calculations.

5. The BWROG must establish a criteria to limit radial and axial peaking factors during startup operations to those values considered for the analyses of the exclusion region.

The main technical issue of significant relevance that still remains to be solved is the reload-dependent confirmatory analyses required to assert the applicability of the previous-cycle safety settings and, in particular, the applicability of "old" exclusion regions to new types of fuel and loading patterns. The BWROG is aware of this problem and is currently developing a methodology

for these cycle-dependent confirmations that is based on a "response surface" approach. The goal is that these confirmatory calculations should be expected to be positive most of the time: major setpoint changes should only be expected following significant fuel design changes. The documentation of this reload-confirmation methodology is expected in Supplement 2 to NEDO-31960 that should be published in the spring of 1993. Supplement 2 will also contain a correlation to estimate the loss of critical-power-ratio margin as a function of the power oscillation amplitude. This correlation is necessary to confirm the setpoints required for Solution III as well as the nonprotected region for Solution I-D.

BACKGROUND

Following the March 1988 instability event in the LaSalle BWR, the BWROG initiated a task to investigate actions that industry should take to resolve the BWR stability issue as an operational concern. Through analysis,[3] the BWROG found that the current plant protection system, that is based on a scram on high average power range monitor (APRM) signal, may not provide enough protection against out-of-phase modes of instability; thus, the BWROG decided that a new automatic instability suppression function was required as a long-term solution and that this function should have a rapid and automatic response which does not rely on operator action.

The BWROG does not plan to solve the stability problem on a "generic" basis, but it has proposed three different options instead. It will be up to the individual licensees to choose which solution will be implemented in their reactor. The options currently being considered by the BWROG are:

I Exclusion Region. A region outside which instabilities are very unlikely is calculated for each representative plant type using well-defined procedures. If the reactor is operated inside this exclusion region, an automatic protective action is initiated to exit the region. This action is based exclusively on power and flow measurements, and the presence of oscillations is not required for its initiation. Four concepts of type I have been proposed by the BWROG:

I-A Immediate protection action (either scram or SRI) upon entrance to the exclusion region.

I-B Same as I-A, but the exclusion region can be bypassed if a stability monitor is operational and detecting sufficiently stable conditions (for instance, decay ratio less than 0.6)

I-C Protection action is taken if two conditions are satisfied: (1) the reactor is operating inside the exclusion region (defined similarly as in Solution I-A), AND (2) an APRM oscillation (of small amplitude) is detected.

I-D Some small-core plants with tight inlet orifices have a reduced likelihood of out-of-phase instabilities. For these plants, it is claimed that the existing flow-biased high APRM scram provides sufficient protection. In addition, administrative controls are proposed to maintain the reactor outside the exclusion region.

II Quadrant-Based APRM Scram. In a BWR/2. the quadrant-based average-power-range monitor is capable of detecting both in-phase and out-of-phase oscillations with sufficient sensitivity to initiate automatic protective action to suppress the oscillations before safety margins are compromised.

III LPRM-Based Detect and Suppress. Local power range monitor (LPRM) signals or combinations of a small number of LPRMs are analyzed on-line by using three diverse algorithms. If any of the algorithms detects an instability, automatic protective action is taken to suppress the oscillations before safety margins are compromised. Two different options have been considered by the BWROG: Solution Concept III, and Solution Concept III-A. The main differences between the two is in the hardware implementation: Solution III requires a new Class 1E computerized system, and Solution III-A may use newly designed digital replacements of the APRM amplifier cards that will require a smaller number of LPRM detectors. Conceptually, the algorithms are (or may be) similar in both solutions.

CONCLUSIONS AND RECOMMENDATIONS

Positive conclusions

1. Overall, the BWROG has done an excellent job of addressing the stability issue in operating reactors. The BWROG has recognized that a problem exists, and they have attempted to solve it in a technically competent manner instead of performing analyses that would defend inaction.

2. The three proposed solution types (I, II, and III) are technically sound and, in our opinion, any of them will solve permanently the issue if properly applied.

3. The solutions can be implemented in existing reactors in a relatively straightforward manner without compromising their intended function.

4. The analyses techniques proposed by the BWROG in their licensing methodology appear to be sufficient to verify the effectiveness of these solutions in the lead plants.

5. The proposed BWROG procedures to generate input data for exclusion region calculations appear to be sufficiently conservative enough. Even though these procedures do not call for absolutely bounding values for all parameters, the conservatism is derived from the fact that reasonably bounding values are used for all parameters at the same time. In the real world, forcing one parameter towards its bounding limit is incompatible with having other parameters at their limit. The conservative nature of these procedures is verified through the use of transient confirmatory analyses under expected operating conditions, which include startup, pump runbacks, and loss feedwater conditions.

6. The application of Solution II to Oyster Creek has shown that the quadrant-based APRM scram provides sufficient protection for either in-phase or out-of-phase oscillations in a BWR/2.

7. The three proposed algorithms for Solution III appear to be able to detect oscillations in a manner reliably enough that automatic suppression action can be taken by the protection system. The good detection sensitivity of the period-based algorithm allows for fairly tight scram setpoints; therefore, this solution does not need to rely heavily on difficult calculations to show sufficient safety margin under a wide range of conditions and fuel types.

8. The arguments presented in NEDO 31960 about the expected oscillation modes are convincing, although they are not absolutely bounding in the case of the single-channel oscillation. We agree that the most likely oscillation modes will be either in-phase (or corewide) or out-of-phase (or regional). Higher order regional modes are not likely, because of their increased eigenvalue separation. We however have some minor reservations with respect to the single-channel type of instability (see reservation 10 in the next section and recommendation 12).

Reservations

1. Even though there are only three general types of solutions (Solutions I, II, and III), at least seven possible implementations (Solutions I-A, I-B, I-C, I-D, II, III, and III-A) have been proposed at one time or another. A more general type of solution that would apply to all reactors would have been preferable.

2. Some solutions (especially Solution I-A, regional exclusion with scram upon entry to the region) will most probably be implemented with margins as tight as possible to avoid unnecessary scrams. This approach might result in cycle-dependent implementations that would require new safety-system setpoints based on cycle-specific data for each reload. This is not a desirable feature in a long-term solution.

3. Solution I-B (exclusion region with bypass based on stability monitor) has not been developed in detail, and it appears to have been abandoned by the BWROG. If solution I-B were still under consideration, we would have reservations with respect to the ability of stability monitors to measure the decay ratio of the out-of-phase instability mode with sufficient accuracy to allow a bypass of the exclusion region scram. For example, in the Ringahls-1 tests,[4] the measured decay ratio was about 0.7 at 70% power and an instability was observed at 72.5% power. This event clearly casts a shadow on the viability of Solution I-B as an option.

4. Solution I-C (delta APRM flux scram) has not been developed in detail by the BWROG. If some licensee would want to pursue this option, we would have to look in more detail at the scram setpoint. The methodology used to estimate this setpoint should be similar to the one used for Solution III, including uncertainties and failed LPRM signals.

5. Solution I-D (small cores with tight inlet orifices) relies too strongly on decay ratio calculations that predict that the oscillation mode is very likely to be corewide. In this solution, the flow-biased scram does not appear to give significant protection against out-of-phase instabilities should they occur. Although these calculations will be documented in Supplement 2 of NEDO 31960 (due spring 1993), it is expected that there will be an area

within the exclusion region where the flow-biased scram does not provide protection for out-of-phase oscillations.

6. We have some concerns about the methodology to estimate the stability of the out-of-phase (or regional) mode of oscillation. The BWROG proposes to use an acceptance region defined in a two-dimensional plane with the FABLE-calculated corewide and hot-channel decay ratios as coordinates (see Fig. 5-1 of reference 2, NEDO-31960/S1). The applicability of this acceptance region to determine whether a reactor condition is likely to oscillate in-phase or out-of-phase may impact the approval of Solution I-D. The two main concerns that we have about the methodology that defines core-channel decay ratio acceptance criteria are:

   6.1 Core-channel decay ratio acceptance criteria were developed by using test data and other calculations. In all these benchmark cases, the actual radial and axial power shapes were used with FABLE to estimate the core and hot-channel decay ratio. The BWROG, however, proposes to distinguish between in-phase and out-of-phase oscillation modes based on this acceptance criteria but using the conservatively defined "procedure" power shapes as inputs instead of the best estimate shapes. Although we agree that the procedure power shapes result in a conservative exclusion region, they may bias the results towards the in-phase mode of oscillation by using nonconsistent axial power shapes (flat for corewide and extremely bottom peaked for the hot channel). In summary, the data base used to develop the acceptance criteria do not envelop the conditions for the intended use (i.e., cannot distinguish accurately between in-phase and out-of-phase modes).

   6.2 The out-of-phase mode of instability is a function of how strong the flow feedback is, and that is represented qualitatively by the channel decay ratio in the acceptance criteria. However, the out-of-phase mode is also a function of the eigenvalue separation between the fundamental and first azimuthal neutronic modes. The eigenvalue separation is not included in the acceptance criteria, which represent only "typical" loading patterns and core sizes. It is conceivable that other loading patterns might result in different acceptance criteria.

7. Reducing the number of false positives (i.e. scrams when it was not required) for Solution III (LPRM-based detect and suppress) is crucial for the solution to work; however, the BWROG may take this false-scram avoidance to such an extreme that the solution will not work. Minor problems with electronic noise, controllers out of tune, or many other unknown parameters may result in failure to scram when required, if this solution is not carefully designed. This is the reason we recommend (as proposed by the BWROG) that several diverse algorithms be implemented simultaneously for Solution III.

8. Solutions III, I-C, and I-D depend partly on a correlation that relates the change in critical power ratio (CPR) caused by a neutron power oscillation. It is not clear that such a correlation exists or how many independent parameters it must contain. The BWROG has been working towards developing this correlation and is trying to define it in a conservative manner. BWROG expects to complete this correlation development in February 1993. The correlation documentation will be included in the Supplement 2 to NEDO-31690 that is expected in the spring of 1993.

9. The applicability of the delta-CPR correlation (see paragraph 8 above) to new fuels or fuels from different vendors is not clear. This point is being addressed by the BWROG, and a formal position is expected in Supplement 2 to NEDO 31960.

10. Reactor operators have a large degree of freedom to chose control rod patterns and power distributions during startup at low powers. Some of these "achievable" power distributions may result in instabilities outside the exclusion region, even if the reload confirmation procedures were successful. Criteria must be set by the BWROG to assure the operator that the reactor is within the limits where the Solution I exclusion region is applicable.

11. Under normal conditions, single-channel instabilities are not probable, because these conditions are likely to induce an out-of-phase instability before the single-channel instability develops. This argument, however, is based on the fact that many channels of the same type are loaded, and therefore, if one channel is close to instability, many channels will also be unstable and are likely to produce a global out-of-phase oscillation. This is not the case, however, with lead use assemblies (LUAs), where perhaps only one channel of that type is loaded. If this LUA had stability characteristics quite different from those of the rest of the core, a single-channel instability in the LUA could be possible. For this reason, we are recommending that the thermohydraulic stability of all LUAs be determined (see recommendation 13).

Recommendations

1. Approve the overall exclusion region calculation methodology as described in NEDO-31960 and its Supplement 1. The results of these exclusion region calculations may be used as part of the implementation of Solutions I and III.

2. Approve the overall treatment of uncertainties described in NEDO-31960 as it applies to the selection of initial conditions for exclusion region calculations and its confirmatory runs.

3. Pending review of the specific reload confirmation procedures that should be outlined in a second supplement to NEDO-31960, approve Solution Concept I-A for implementation in any BWR line with the following design objectives:

   3.1 Specific reload confirmation procedures must be developed so that for every reload, the licensee can either (1) confirm the applicability of old exclusion region settings, or (2) set a new exclusion region boundary.

   3.2 Favor implementations of Solution I-A that are not expected to change the exclusion boundary setpoints on a cycle-by-cycle basis. Confirmatory calculations should be expected to be positive most of the time; major setpoint changes should only be expected following significant fuel design changes.

   3.3 Select rod insert (SRI) may be used in conjunction with Solution I-A, but a full scram must take effect if the reactor does not exit the region within a reasonable period of time (of the order of a few seconds).

4. Do not approve Solution Concept I-B at this time. Solution I-B has not been developed in detail by the BWROG. If a licensee chose to implement Solution I-B, they would have to resolve the question of whether a noise-based stability monitor can provide adequate protection against instabilities in the out-of-phase mode.

5. Do not approve Solution Concept I-C at this time. Solution I-C has not been developed in detail by the BWROG.

6. Do not approve Solution Concept I-D at this time until lead plant confirmation analyses are performed, documented, and reviewed. It is expected that Supplement 2 to NEDO 31960 (due spring 1993) will contain confirmation analyses for the Duane Arnold plant that will allow a detailed review and a final decision on the acceptability of Solution Concept I-D. This recommendation does not imply a rejection of Solution I-D; the approval of Solution I-D depends on the details of calculations that have not yet been performed by the BWROG.

7. Approve Solution Concept II for implementation in BWRs with quadrant APRM scram (i.e. in any BWR/2). Oyster Creek has already submitted technical specification changes that implement this solution (see Ref. 5).

8. Approve Solution Concept III for implementation in any BWR line with the following design objectives:

   8.1 To avoid unexpected problems, several diverse algorithms must be used to detect oscillations. Automatic protection action must initiate if either of the algorithms detects oscillations (i.e., the algorithm outputs are connected by a logical OR, not a logical AND).

   8.2 The three algorithms described in NEDO-31960 and its supplement may be used in Solution III. These three algorithms are (1) high LPRM oscillation amplitude, (2) high-low detection algorithm, and (3) period-based algorithm. Preferably, all three algorithms should be used.

   8.3 The licensees that implement these algorithms must demonstrate by analyses the validity of the scram setpoints selected. These analyses may be performed on a representative-plant basis when applicable but must include an uncertainty treatment that takes into account the number of failed sensors permitted by technical specifications.

   8.4 The scram setpoints should be selected such that at least one of the algorithms has a large probability of detecting the oscillation and initiating protective action to prevent violation of any fuel safety criterion.

   8.5 If the algorithms detect oscillations, an automatic protection action must be initiated. This action may be a full scram or an SRI. If an SRI were to be implemented with Solution Concept III, a full scram must take effect if either (1) the oscillations do not disappear in a reasonable period of time, or (2) the reactor remains inside the exclusion region as defined by the general regional exclusion methodology of Solution I-A.

9. The LPRM groupings defined in NEDO 31960 to provide input to the Solution III algorithms appear appropriate for the intended oscillation-detection function. These LPRM groupings are: the oscillation power range monitor (OPRM) for Solution III or the octant-based arrangements for Solution III-A. The minimum requirements for operable number of LPRM detectors set in NEDO 31960 appear appropriate.

10. Approve the overall treatment of uncertainties described in NEDO-31960 as it applies to the selection of oscillation contours and failed LPRMs for Detect and Suppress Concepts (Solution III).

11. Implementation of Solution III will require the documentation of the selection of the bypass region outside which the detect and suppress action is deactivated.

12. The BWROG must establish a criteria to limit the actual radial and axial peaking factors during startup operations to those values considered for the analyses of the exclusion region. This criteria must be based on parameters or information readily available to the operator in the control room. Defining this criteria must be part of the reload confirmation analyses.

13. Establish a procedure to review the thermohydraulic stability of lead use assemblies (LUA). Solutions I and II do not protect the reactor in the case of a single-channel instability, and the protection for Solution III is limited. These instabilities are not likely if many bundles of one type are loaded in the core, but they could be possible if the wrong type of LUA were to be loaded. Thermohydraulic stability analyses must be required during LUA review if Solutions I or II are used.

REFERENCES

1. General Electric Company, *BWR Owners' Group Long-Term Stability Solutions Licensing Methodology*, NEDO-31960, May 1991.

2. General Electric Company, *BWR Owners' Group Long-Term Stability Solutions Licensing Methodology*, NEDO-31960 Supplement 1, March 1992.

3. General Electric Company, *Fuel Thermal Margin During Core Thermal Hydraulic Oscillations in a Boiling Water Reactor*, NEDO-31708, June 1989

4. B-G Bergdahl and R. Oguma, "BWR Stability Investigation in Ringhals-1 Measurement Data from October 26, 1989." *Proceedings of The International Workshop on Boiling Water Reactor Stability, Holtsville, N.Y., 17—19 October 1990*, pp 142—159, OECD/NEA/CSNI Report No 178, October 1990.

5. Oyster Creek Nuclear Generating Station, *Technical Specification Change Request No. 191*, Docket No. 50-219, October 9, 1991.

APPENDIX A

LAPUR AUDIT CALCULATIONS
OF Solution I EXCLUSION REGION CALCULATIONS

AUDIT CALCULATIONS

A series of audit calculations were performed with the LAPUR code to verify the results presented by the BWROG that were based on FABLE/BYPSS calculations. All relevant input data used in the FABLE/BYPSS was made available for this review, and we set up LAPUR input decks that were representative of the conditions modeled by FABLE. The main result of these calculations is presented in Table A.1 and Figs A.1 and A.2. We observe that the maximum difference between LAPUR- and FABLE-calculated decay ratios is 0.09. This can be considered as excellent agreement and representative of the differences in modeling of both codes.

This type of code-to-code benchmark is not as good as a code-to-data benchmark, but it assures that gross modeling errors or systematic errors in the preparation of the input decks have not occurred. Furthermore, it ensures that "data fudging" is not taking place to obtain desired results, because all the data has to be made available and is evaluated for expected value ranges.

Both codes, LAPUR and FABLE, have been benchmarked against data from actual stability tests with satisfactory results. In general, it is recognized that this type of frequency-domain codes has an accuracy better than 20%. Thus, if a decay ratio of 0.8 or smaller is calculated, it is highly probable that stable reactor operation will result. Decay ratios larger than 0.8 result in smaller probabilities of stable operation. Note, however, that large errors are possible if the proper data are not used as input to the code. The 20% error quoted above is for detailed test benchmarks where extreme care is taken to reproduce the exact axial and radial power shapes, core pressure drops, and reactivity coefficients; calculations using approximate descriptions of the core operating condition are likely to result in larger errors.

The axial power shapes assumed in the BWROG analysis are: (1) fairly uniform (end-of-cycle Haling) shape to calculate the corewide decay ratio, and (2) strongly bottom peaked (2.0 peaking at node 3/24) to calculate the hot-channel thermohydraulic decay ratio. It is well known that the high-power channels (maybe 25% of the total number of channels) have the most influence in the stability of the reactor. This is due to the fact that the adjoint flux and density reactivity coefficients are higher in the high-power channels. Furthermore, hot channels tend to have bottom-peaked power shapes, that may be more unstable. To test the validity of the uniform power shape assumption, we ran two cases to determine corewide stability boundary: (1) with all channels having the same Haling power shape and (2) with a graded axial power shape, so that the hot channels have a bottom-peaked shape (2.0 at node 3/24), but the core average is the same as in case (1). The chosen power shapes are drawn in Figs. A.3 and A.4 for a BWR/3 and BWR/5 respectively. The out-of-phase and hot-channel decay ratio calculations were based on the graded power shapes of Figs. A.3 and A.4. The out-of-phase decay ratio was calculated by LAPUR assuming an eigenvalue separation of \$1.00 between fundamental and first harmonic neutronic modes. This \$1.00 value is a representative, but not bounding value of the eigenvalue separation. These results show that the uniform (Haling) power shape is more conservative at lower flows, but the use of bottom peaked graded shapes results in higher decay ratios at higher flows.

Reload Confirmation Procedures

The main technical issue of significant relevance that still remains to be solved is the reload-dependent confirmatory analyses required to assert the applicability of the previous-cycle safety settings and, in particular, the applicability of "old" exclusion regions to new types of fuel and loading patterns. The BWROG is aware of this problem and is currently developing a methodology for these cycle-dependent confirmations that is based on a "response surface" approach. The goal is that these confirmatory calculations should be expected to be positive most of the time; major setpoint changes should only be expected following significant fuel design changes. The documentation of this reload-confirmation methodology is expected in Supplement 2 to NEDO-31960 that should be published in the spring of 1993.

Of particular concern is how the reload procedures will be used to evaluate startup power distributions. For example, the root cause of a recent instability event in a BWR/5 has been determined to be the extreme radial (1.92) and axial (up to 1.87) peaking factors during the startup. This extreme power distribution was apparently not covered by the standard exclusion region calculations, which assumed a more mild radial power peaking factor. Nevertheless, the operator was allowed to have that extreme distribution without violating any thermal limits.

Figure A.5 shows a comparison of the equilibrium-cycle exclusion region for the Perry reactor (a BWR/6) and the exclusion region that results if the actual axial and power shapes from the recent BWR/5 event are used. As it can be observed, the standard BWR/6 exclusion region from NEDO-31960 is not as conservative as the actual region. Therefore, we have recommended that the BWROG must establish a criteria to limit the actual radial and axial peaking factors during startup operations to those values considered for the analyses of the exclusion region. This criteria must be based on parameters or information readily available to the operator in the control room.

Table A.1. LAPUR-FABLE/BYPSS benchmark/audit calculations

Reactor type	Power (%)	Flow (%)	Corewide decay ratio		Hot-channel decay ratio	
			FABLE	LAPUR	FABLE	LAPUR
BWR/3	42	30	0.77	0.68	0.34	0.28
	52	45	0.46	0.47	0.19	0.17
	71	45	0.65	0.64	0.35	0.38
	84	60	0.45	0.41	0.21	0.28
BWR/5	42	30	0.65	0.73	0.50	0.50
	52	45	0.39	0.37	0.34	0.32
	71	45	0.56	0.50	0.55	0.62
	84	60	0.30	0.32	0.31	0.39

Table A.2. Typical LAPUR5X input deck for a BWR/3

```
LAPUR5X :: BWROG I BWR3 71/45
  1
988.8, 502.3, 1782.9, 44.01E6, 0.101, 0.035, 0.80, 1.0
  2
7, 24, 0, 1, 0, 0, 0, 1, 0
  3
5, 25, 25, 25, 25, 25
  4
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875,  30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875,  30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875,  30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875,  30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875,  30.48
  5
        0.27, 0.86, 1.02, 1.06, 1.08, 1.09, 1.10,
        1.11, 1.14, 1.15, 1.16, 1.16, 1.17, 1.17,
        1.17, 1.16, 1.15, 1.14, 1.11, 1.06, 0.98,
        0.85, 0.58, 0.27, 0.00
        0.92, 1.64, 2.00, 1.88, 1.70, 1.53, 1.37,
        1.25, 1.15, 1.07, 1.00, 0.95, 0.90, 0.86,
        0.81, 0.78, 0.74, 0.70, 0.66, 0.60, 0.53,
        0.44, 0.33, 0.19, 0.00
        0.20, 1.20, 1.40, 1.60, 1.51, 1.40, 1.30,
        1.22, 1.15, 1.09, 1.04, 1.00, 0.97, 0.94,
        0.90, 0.87, 0.84, 0.81, 0.78, 0.73, 0.67,
        0.59, 0.48, 0.33, 0.00
        0.20, 1.00, 1.10, 1.30, 1.29, 1.24, 1.19,
        1.14, 1.11, 1.08, 1.05, 1.03, 1.00, 0.99,
        0.96, 0.95, 0.93, 0.91, 0.89, 0.85, 0.81,
        0.75, 0.67, 0.54, 0.00
        0.03, 0.13, 0.12, 0.13, 0.32, 0.55, 0.77
        0.96, 1.15, 1.29, 1.41, 1.49, 1.57, 1.63
        1.70, 1.72, 1.74, 1.75, 1.72, 1.67, 1.54
        1.29, 0.69, 0.02, 0.00
  7
7,  1,   1,   1,   1,   1,   1,   1
  9
7, 127.2, 127.0, 98.6, 129.3, 97.4, 127.3, 25.0
 10
7, 36.8, 36.8, 36.8, 36.8, 36.8, 36.8, 229.0
 11
7, -0.280, -0.280, -0.280, -0.280, -0.280, -0.280, -0.280
 13
7,  0.,  0.,  0.,  0.,  0.,  0.,  0.
 14
7,  81,  87,  73,  106,  91,  202,  84
 15
7,  60,  60,  60,  62,  60,  62,  62
 16
7,  1,   1,   1,   2,   1,   2,   2
 17
    2, 411.48, 411.48
 18
    2, 231.24, 238.96
 19
    2, 97.97, 101.15
 20
    2, 97.97, 101.15
 21
    2, 1.33, 1.34
 22
```

```
     2,   0.1,   0.1
  23
     2,   0.1,   0.1
  24
     2,   1.3,   1.3
  25
     2,  0.125,  0.125
  26
 7,   1,    1,    1,    2,    1,    2,    2
  27
     2,  10.44,  10.33
  28
     2, 1.0439, 1.0414
  29
     2, 0.5581, 0.5581
  30
     2, 0.0373, 0.0373
  31
     2, 0.0813, 0.0813
  32
     2, 0.2675, 0.2347
  33
     2, 0.0114, 0.0114
  34
 7,   1,    1,    1,    1,    1,    1,    1
  35
     1,    1
  36
        411.48
  37
          1.40
  53
  1.E-3,  1.E-3,  1.E-3,  2.E-5,  1.E-3,  1.E-9,  1.E-2,  5.E-8
  54
     1, 12
  0
```

Table A.3. Typical LAPUR5W input deck for a BWR/3

```
 7, 1, 1
2
      81,     87,     73,    106,     91,    202,     84
3
      411 -48
4
      0.2,  0.3
5
 1    6
6
             1
7
  6, 0.185E 3, 1.226E 3, 1.096E 3, 2.210E-3, 0.647E 3, 0.236E 3
8
  6,  0.0124,   0.0305,   0.1110,   0.3010,   1.1300,   3.0000
9
             1
10
    1       0 0
11
             1
12
    1     4 E 5
17
      -2.5E 03
18
    1       7
19
            1,     1,     1,     1,     1,     1,     1
20
          1.2,    1.0,    0.8,    0.6,    0.4,    0.2,    0.0
21
        -5.64,   5.66,   14.06, 18.84,  20.51,  20.98,  21.44
22
  12,  0.20, 0.30, 0.40, 0.43, 0.46, 0.50, 0.53
     0.56, 0.60, 0.63, 0.66, 0.70
23
 0   1   1   1   1   1   0   0   0   0   0   0   0   0
 0   0   0   0   0   0   0   0
24
 1   1   1   1   1   1   1   1   1
```

```
28
  1.
29
  5     0.0     0.5     -1.0     -1.5     -2.0
30
  0
0
```

Table A.4. Typical LAPUR5X input deck for a BWR/5

```
LAPUR5X :: BWROG I BWR5 71/45
  1
988.8, 505.7, 2359., 48.7E6, 0.095, 0.035, 0.80, 1.0
  2
7, 24, 0, 1, 0, 0, 0, 1, 0
  3
5, 25, 25, 25, 25, 25
  4
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 30.48
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 15.875, 15.875, 15.875, 15.875
        15.875, 15.875, 15.875, 30.48
  5
        0.319 , 0.9116, 1.1074, 1.1675, 1.1771, 1.1688, 1.1563
        1.1448, 1.1338, 1.1236, 1.1168, 1.1144, 1.115 , 1.1168
        1.1176, 1.1162, 1.1104, 1.0959, 1.0671, 1.0159, 0.9304
        0.7936, 0.6085, 0.2713, 0.0
        0.9200, 1.6400, 2.0000, 1.8800, 1.7000, 1.5300, 1.3700
        1.2500, 1.1500, 1.0700, 1.0000, 0.9500, 0.9000, 0.8600
        0.8100, 0.7800, 0.7400, 0.7000, 0.6600, 0.6000, 0.5300
        0.4400, 0.3300, 0.1900, 0.00
        0.6243, 1.4709, 1.6902, 1.6185, 1.5084, 1.4012, 1.2969
        1.2163, 1.1473, 1.0909, 1.0404, 1.0037, 0.9664, 0.9361
        0.8977, 0.8743, 0.8427, 0.8105, 0.7778, 0.7276, 0.6671
        0.5856, 0.4788, 0.3253, 0.0
        0.6279, 1.2755, 1.3809, 1.3471, 1.2940, 1.2406, 1.1870
        1.1442, 1.1067, 1.0752, 1.0465, 1.0253, 1.0033, 0.9852
        0.9619, 0.9475, 0.9278, 0.9074, 0.8863, 0.8531, 0.8118
        0.7536, 0.6716, 0.5385, 0.0
        0.1017, 0.6170, 0.7994, 0.9319, 1.0072, 1.0526, 1.0894
        1.1146, 1.1341, 1.145 , 1.1640, 1.1795, 1.1998, 1.2183
        1.2395, 1.2494, 1.2567, 1.2508, 1.2230, 1.1694, 1.0680
        0.8977, 0.6653, 0.2213, 0.0
  7
7,   2,   3,   4,   5,   5,   5,   5
  9
7, 121.11, 121.60, 121.72, 122.39, 121.26, 121.96, 32.94
 10
7,  27.7,  27.7,  27.7,  27.7,  27.7,  27.7, 193.0
 11
7, -0.280, -0.280, -0.280, -0.280, -0.280, -0.280, -0.280
 13
7,   0.,   0.,   0.,   0.,   0.,   0.,   0.
 14
7,   83,   87,  100,  110,  122,  170,   92
 15
7,   62,   62,   62,   62,   62,   62,   62
 16
7,   1,    1,    1,    1,    1,    1,    1
 17
    1  411.48
 18
    1  238.96
 19
    1  101.15
 20
    1  101.15
 21
    1    1.34
 22
```

```
    1    0.1
23
    1    0.1
24
    1    1.3
25
    1    0.125
26
7,  1,   1,   1,   1,   1,   1,   1
27
    1    10.33
28
    1    1.0414
29
    1    0.5581
30
    1    0.0373
31
    1    0.0813
32
    1    0.2256
33
    1    0.0114
34
7,  1,   1,   1,   1,   1,   1,   1
35
    1    1
36
       411.48
37
        1.40
53
  1.E-3   1.E-3   1.E-3   2.E-5   1.E-3   1.E-9   1.E-2   5.E-8
54
    1     25
56
 11  12  13  0
 0
```

Table A.5. Typical LAPUR5W input deck for a BWR/5

```
LAPUR5W .. BWROG I BWR5 71/45
 1
  7, 1, 1
 2
       83.    87.    100,    110,    122,    170,    92
 3
       411.48
 4
       0.2,  0.3
 5
  1   6
 6
              1
 7
   6, 0.185E-3, 1.226E-3, 1.096E-3, 2.210E-3, 0.647E-3, 0.236E-3
 8
   6,   0.0124,   0.0305,   0.1110,   0.3010,   1.1300,   3.0000
 9
              1
10
     1      0.0
11
              1
12
     1    4.E-5
17
       -2.5E-03
18
     1        7
19
             1.     1.     1.     1.     1.     1.     1
20
           1.2,    1.0,    0.8,    0.6,    0.4,    0.2,    0.0
21
           6.05,   3.63,   11.81, 17.19,  19.84,  20.78,  21.70
22
   12,  0.20, 0.30, 0.40, 0.43, 0.46, 0.50, 0.53
       0.56, 0.60, 0.63, 0.66, 0.70
23
  0   1   1   1   1   1   0   0   0   0   0   0   0   0
  0   0   0   0   0   0   0   0
24
 1   1   1   1   1   1   1   1   1
28
   1.
29
   5     0.0     0.5     1.0    -1.5    -2.0
30
   0
 0
```

Table A.6. LAPUR calculations for a typical BWR/3
(DR = decay ratio, NF = natural frequency of oscillation)

Flow (Mlb/h)	Power (MW)	Corewide*		Out-of-phase		Hot-channel		Corewide**	
		DR	NF (Hz)	DR	NF (Hz)	DR	NF (Hz)	DR	NF (Hz)
10	800			0.93	0.31	0.80	0.27	1.13	0.31
20	800	0.82	0.26	0.36	0.34	0.32	0.29	0.64	0.34
20	1000			0.65	0.38	0.48	0.34	0.85	0.38
20	1200			0.95	0.41	0.67	0.37	1.07	0.41
29.4	1060	0.68	0.32	0.28	0.41	0.28	0.36	0.50	0.42
29.4	1200	0.79	0.34						
29.4	1500			0.79	0.50	0.55	0.45	0.86	0.50
44	1306	0.47	0.37	0.14	0.48	0.17	0.42	0.29	0.48
44	1783	0.64	0.44	0.45	0.59	0.38	0.54	0.51	0.58
44	2000	0.69	0.43	0.66	0.62	0.49	0.57	0.64	0.61
44	2500	0.85	0.51	1.05	0.67	0.84	0.62	0.98	0.66
50	2500			0.82	0.70	0.59	0.65	0.73	0.70
50	2600			0.85	0.70	0.66	0.66	0.80	0.70
50	3000	0.79	0.57						
58.7	2109	0.41	0.50	0.22	0.68	0.28	0.66	0.27	0.66
58.7	3000			0.64	0.84	0.50	0.77	0.51	0.81
58.7	3200			0.82	0.85	0.59	0.79	0.62	0.84

* Using average axial power shape for all channels.
** Using graded power shapes of Fig. A.3.

Table A.7. LAPUR calculations for a typical BWR/5
(DR = decay ratio, NF = natural frequency of oscillation)

Flow (Mlb/h)	Power (MW)	Corewide*		Out-of-phase		Hot-channel		Corewide**	
		DR	NF (Hz)	DR	NF (Hz)	DR	NF (Hz)	DR	NF (Hz)
10	500	0.76	0.17	0.52	0.22	0.74	0.20	0.75	0.22
10	600	1.06	0.18	0.74	0.24			0.96	0.24
20	800	0.76	0.24	0.49	0.31	0.40	0.27	0.65	0.31
20	1000	0.96	0.27	0.84	0.35	0.63	0.31	0.93	0.34
32.5	1396	0.73	0.35	0.68	0.46	0.50	0.42	0.72	0.45
32.5	1500	0.78	0.36	0.83	0.47	0.57	0.43	0.82	0.46
32.5	1600	0.83	0.38						
48.7	2200			0.72	0.63	0.53	0.59	0.57	0.61
48.7	2360	0.50	0.49	0.87	0.65	0.62	0.60	0.69	0.63
48.7	2500	0.55	0.50	0.99	0.66	0.70	0.62	0.80	0.64
48.7	3000	0.71	0.55						
48.7	3500	0.94	0.60						
55	2500			0.68	0.69	0.51	0.65	0.49	0.67
55	2700			0.84	0.71	0.61	0.67	0.62	0.69
55	3000	0.50	0.57	1.07	0.75	0.78	0.70	0.84	0.72
55	4000	0.82	0.66						
65	2791	0.32	0.58	0.32	0.78	0.39	0.75	0.23	0.74

* Using average axial power shape for all channels.
** Using graded power shapes of Fig. A.3.

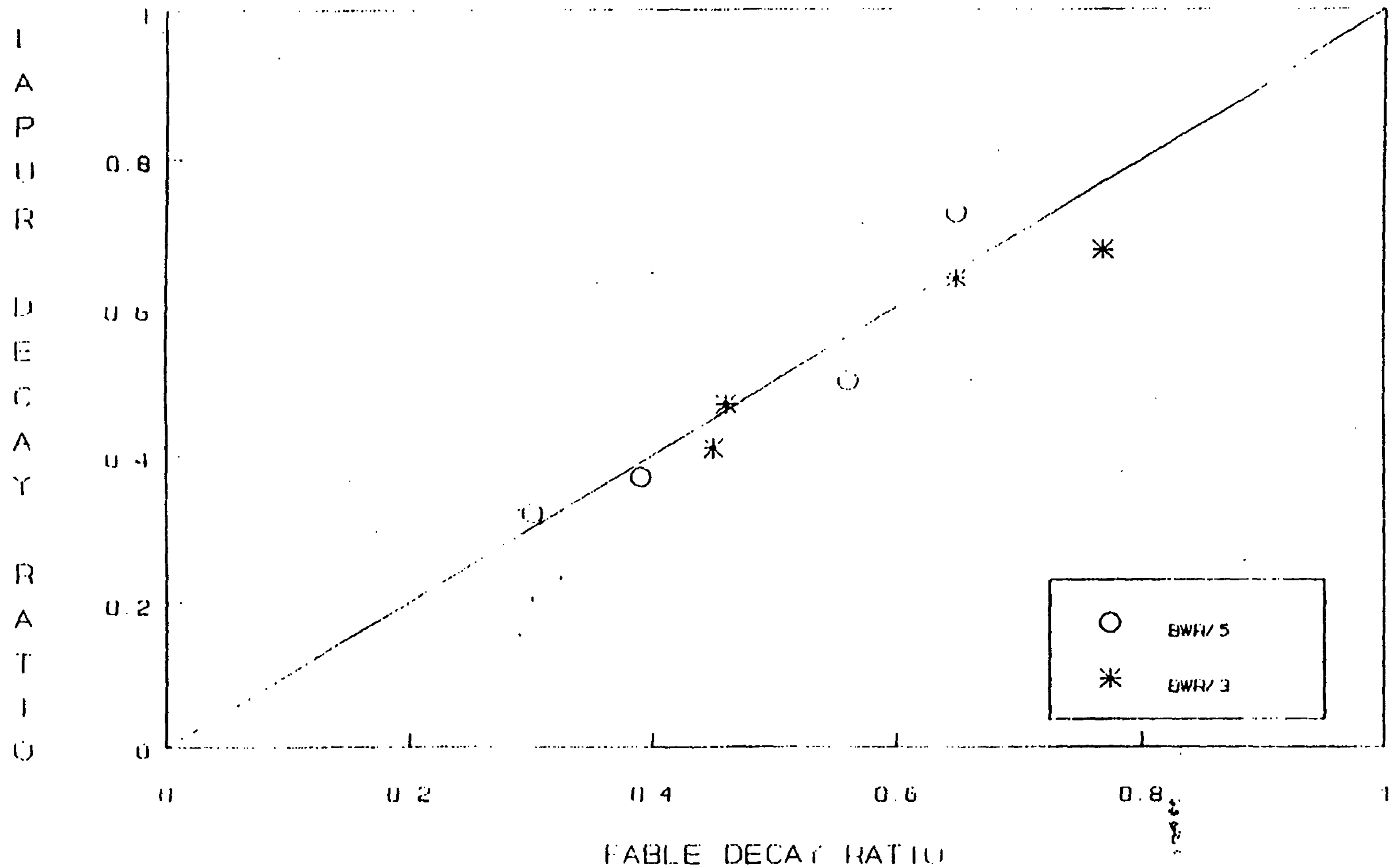

Figure A.1. Comparison between corewide decay ratios calculated by LAPUR and FABLE.

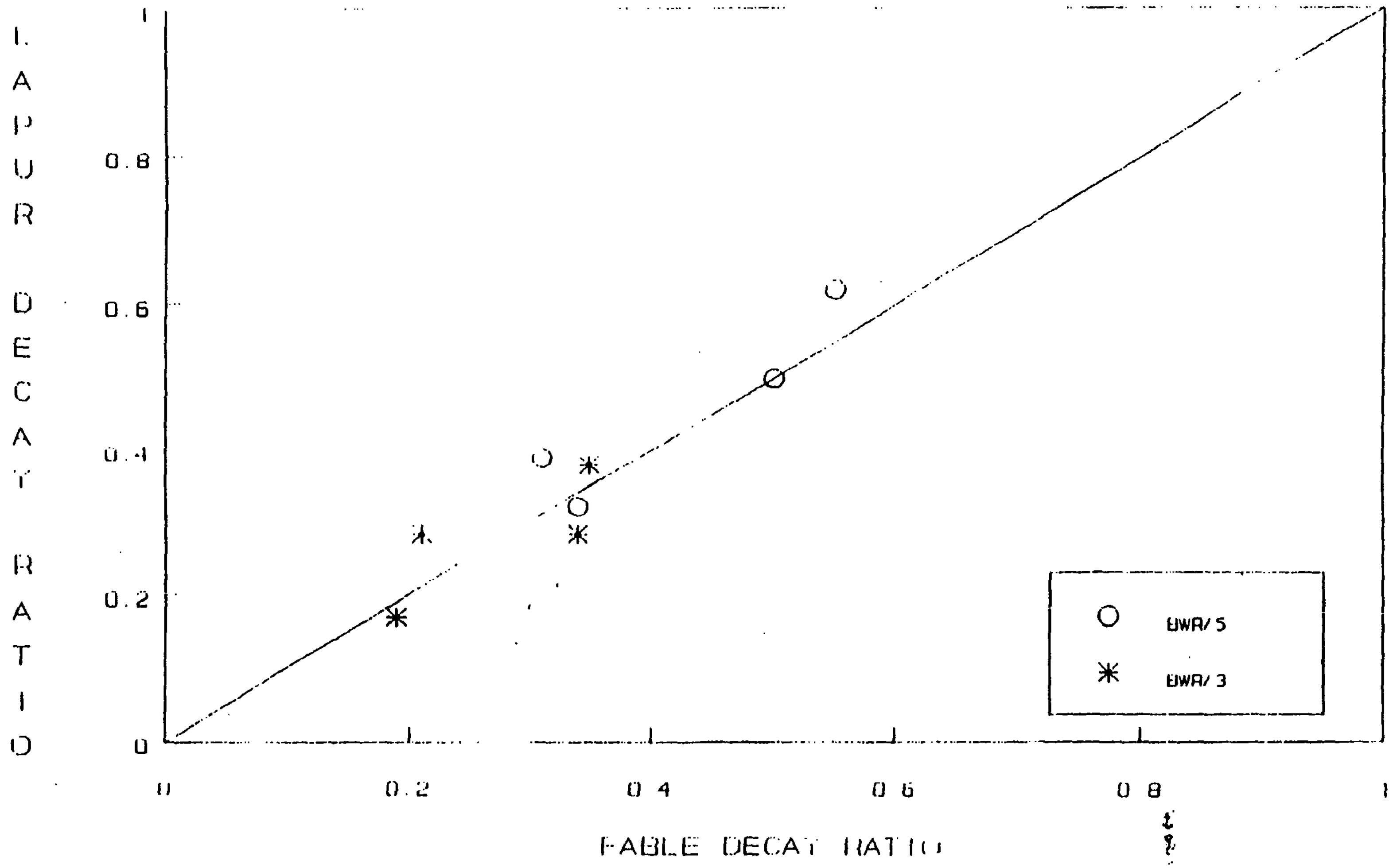

Figure A.2. Comparison between hot-channel decay ratios calculated by LAPUR and FABLE.

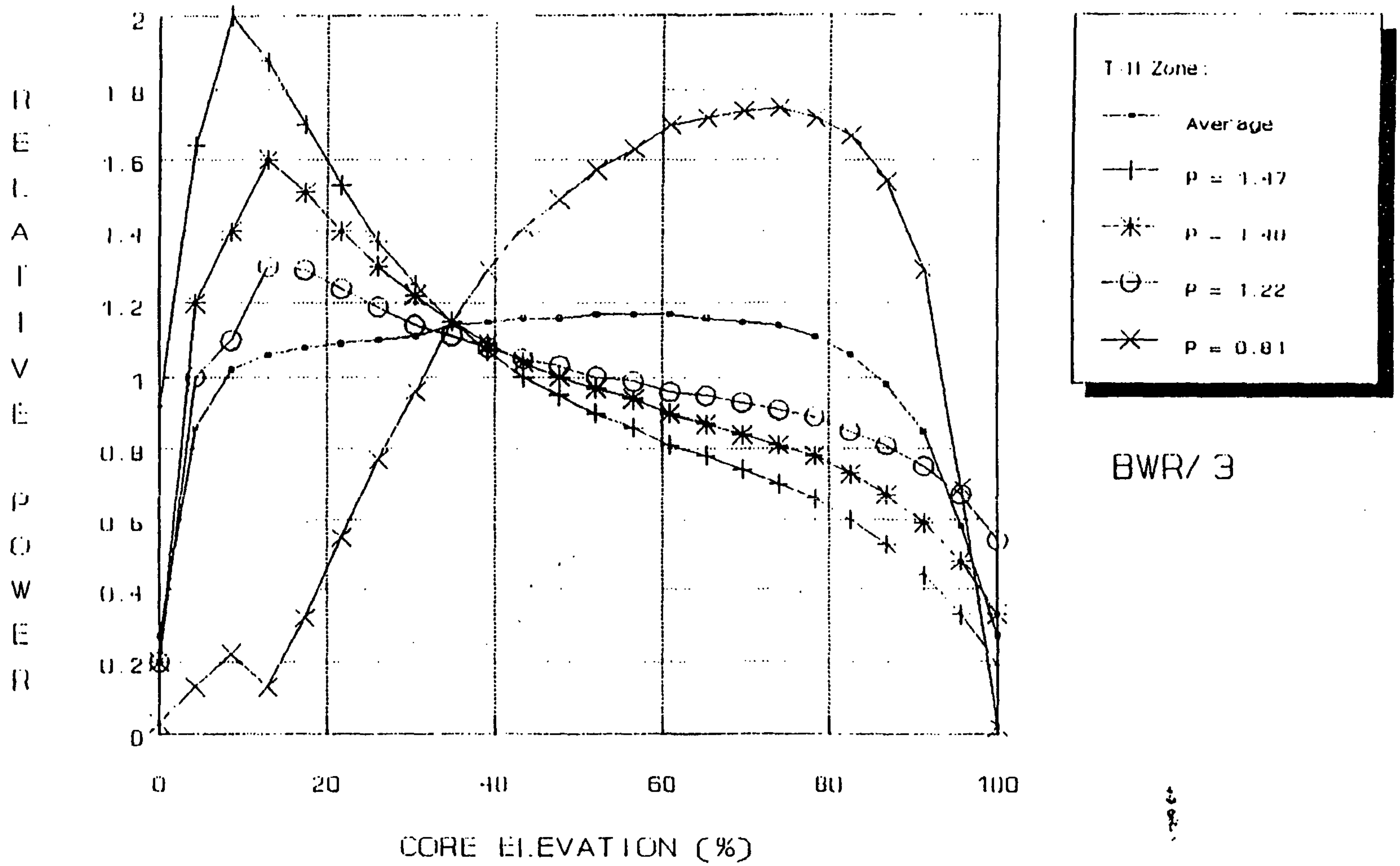

Figure A.3. Power shapes used for hot-channel and out-of-phase mode calculations in a typical BWR/3.

NEDO-31960-A

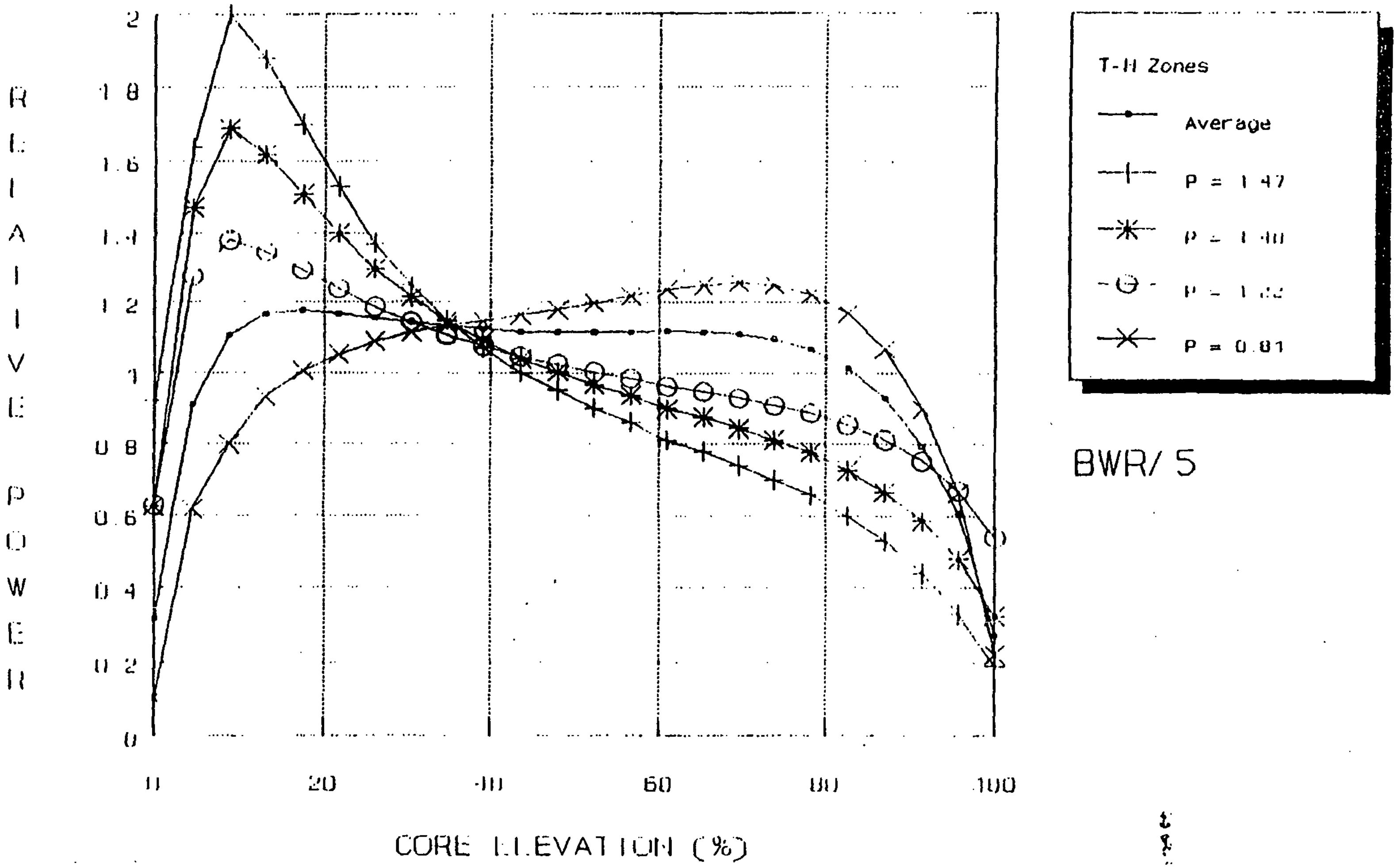

Figure A.4. Power shapes used for hot-channel and out-of-phase mode calculations in a typical BWR/5.

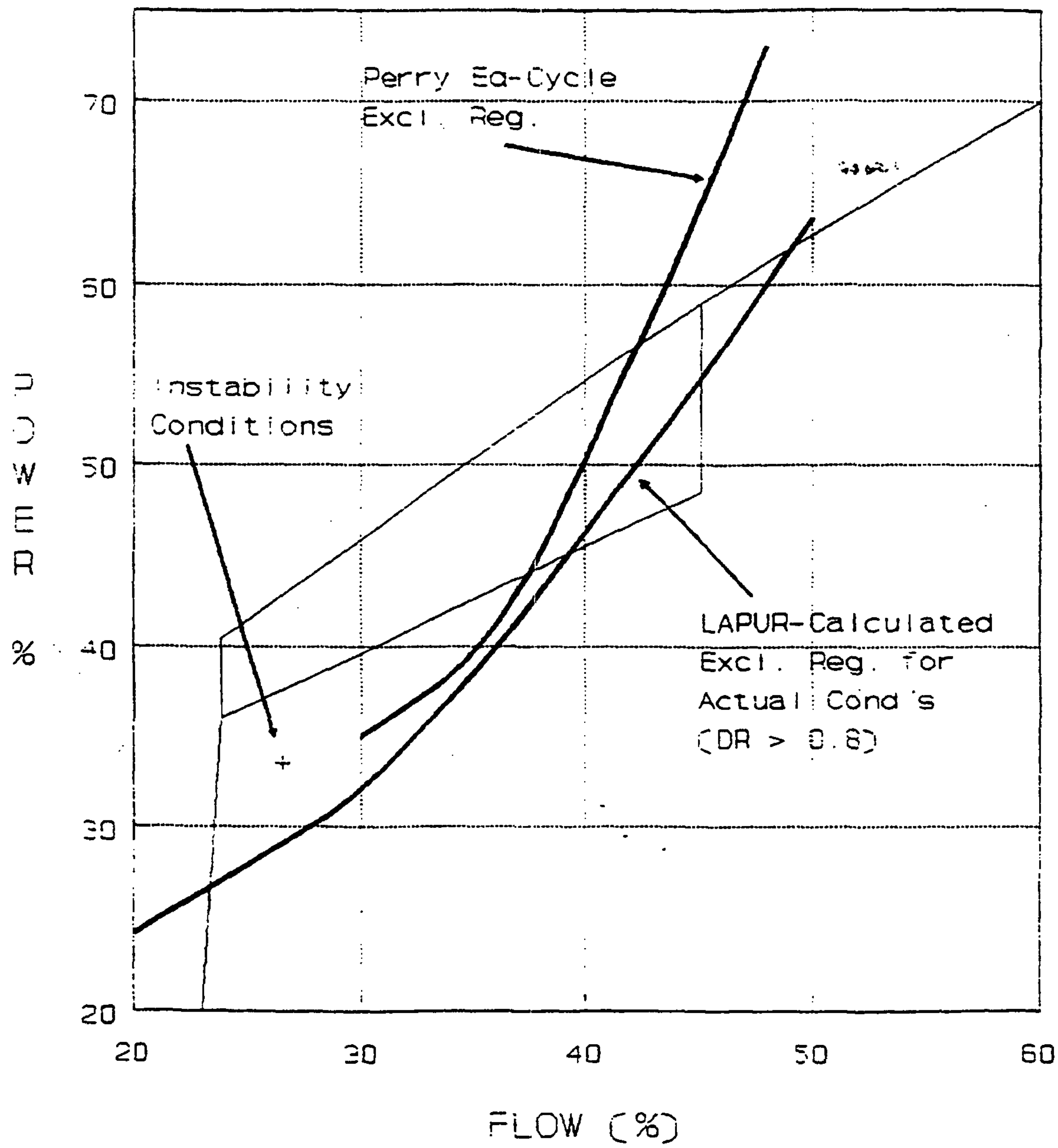

Figure A.5. Comparison between equilibrium-cycle exclusion region and the exclusion region for the specific operating conditions of a BWR/5 instability event.

APPENDIX B
Solution I-D REQUIREMENTS

This section defines a minimum set of items that will have to be provided to complete our review of Solution I-D. These items should be provided for the initial Solution I-D lead plant (Duane Arnold).

1. Describe how the exclusion region for administrative control actions will be calculated and defined.

2. Describe in detail the proposed administrative control actions if the reactor enters the exclusion region.

3. Describe any hardware or administrative control rod block functions that will be associated with the exclusion region. Specifically, describe how these functions are calculated and defined and what type of automated or operator action is required.

4. Describe in detail the information that the operator relies on to provide these administrative controls. In particular, describe how the information is presented to the operator and its "safety classification" (i.e. Class 1-E or not). Explain why this safety classification is adequate.

5. Describe what indications the operator would have in the control room if a power oscillation (either in-phase or out-of-phase) were to develop. Describe the operator actions required under these circumstances.

6. Provide analyses showing the area inside the exclusion region where the flow-biased scram does not provide protection for out-of-phase instabilities. These calculations determine the nonprotection line, which is defined as the line in the power-flow map below which the flow-biased scram does not provide automatic protection. Two lines must be defined:

   6.1 The nonprotection line at the 95% probability level with the initial CPR at technical specification limits.

   6.2 The nonprotection line at the 50% probability level with the expected initial CPR.

7. Provide reasonably bounding analyses showing that oscillations in the out-of-phase mode are highly unlikely in Solution I-D plants operating below the 50%-level nonprotection line. These calculations must be performed along the 50%-level nonprotection line and include at least the following cases:

   7.1 Calculations of core and hot-channel decay ratios using the standard BWROG procedures for exclusion region calculations (NEDO 31960). These calculations must show that the core decay ratio is significantly larger than the hot-channel decay ratio so that the predicted mode of oscillation for these conditions is in-phase. Provide documentation of the radial power distribution (in particular the hot-channel peaking

factor) used in these calculations, and justify why the chosen peaking factors are conservative.

7.2 Calculations of core and hot-channel decay ratios using conservatively defined bottom-peaked power shapes that are more representative of startup conditions than the standard BWROG procedure. These calculations must include axial and radial power shapes representative of (1) normal startup and (2) operation with failed feedwater heaters. Document the actual power shapes used and justify their conservatism.

LAPUR CALCULATIONS RELATED TO Solution I-D

A series of calculations have been performed with the LAPUR code to confirm the validity of the BWROG claim that small cores with tight inlet orifices are not likely to have out-of-phase instabilities. The results of our analyses show that indeed (as claimed by the BWROG) small cores and tight inlet orifices are beneficial for the out-of-phase mode. However, this benefit does not appear to be sufficient to completely discard the possibility of out-of-phase instabilities in these types of reactors; therefore, we have requested that the BWROG perform the calculations described in the preceding section. Table B.1 shows some of the results of these analyses.

Effect of tight inlet orifice

For the calculations presented in Table B.1, we prepared a representative LAPUR input deck (shown in Table B.2) with a single thermohydraulic region and calculated the corewide and out-of-phase decay ratios as a function of the inlet restriction coefficient to simulate the differences between Solution I-D plants and others. In plants where solution I-D is applicable, the inlet restriction coefficient is of the order of 35 to 40 velocity heads, while other plants have values of the order of 25 to 30 velocity heads; for example, Duane Arnold (the proposed lead plant for Solution I-D) has an inlet orifice diameter of 2.09 inches, compared to 2.43 inches for LaSalle. We have to note that the conditions (especially the axial power shape) chosen for these analyses are not representative of normal operation, but they are achievable and not necessarily bounding; these conditions were chosen because they tend to excite the out-of-phase mode more than the corewide. Two main conclusions can be drawn from the results in Table B.1:

(1) The smaller inlet orifice by itself does not preclude the possibility of out-of-phase instabilities. For example, at 35 velocity heads, the out-of-phase mode is predicted to be unstable (decay ratio greater than 0.8) even at large eigenvalue separations of $1.5.

(2) The smaller orifice by itself does not guarantee that the corewide mode will dominate and become unstable before the out-of-phase mode does. For example, at 35 velocity heads, the out-of-phase decay ratio is 0.90 at $1.0 subcritical, while the corewide decay ratio is only 0.84.

In summary, even though smaller (tight) inlet orifices are beneficial and tend to stabilize the out-of-phase mode, increasing the orifice coefficient by about 10 velocity heads reduces the out-of-

phase decay ratio by only 10% to 20% depending on the actual circumstances. Therefore, tight inlet orifice plants are less likely to have out-of-phase instabilities, but given that it only results in a 10% to 20% reduction, this effect by itself is not sufficient to preclude out-of-phase instabilities.

Effect of smaller cores

Smaller cores affect the stability of the out-of-phase mode by increasing the neutron leakage on the core periphery. Larger leakage rates tend to increase the eigenvalue separation between the fundamental and first azimuthal harmonic; the larger the separation, the more stable the out-of-phase mode (see Table B.1 for an example). Our evaluation analyses using the LAPUR code indicate that the net effect of reducing the core size in half is to reduce the out-of-phase decay ratio by 10% to 15%. This evaluation assumes constant loading patterns and fuel types; positive or negative changes of larger magnitude can be achieved by altering the loading patterns or fuel type. Therefore, we conclude that the net effect of the small core size by itself (although beneficial) is not sufficient to preclude out-of-phase instabilities in Solution I-D plants.

In first approximation (assuming a homogenous, cylindrical core), the eigenvalue separation of the first azimuthal mode, ρ_s, is given by

$$\rho_s = \frac{D \Delta B^2}{\nu \Sigma_f} \qquad \textbf{(B-1)}$$

where D is the diffusion coefficient, $\nu\Sigma_f$ is the fission cross section, and ΔB^2 is the difference in geometric buckling between the fundamental and the first azimuthal modes.

The geometric buckling in a cylinder is approximately proportional to the inverse of the radius square and, therefore, is somehow inversely proportional to the number of bundles in the core. Consequently, if core A has half the number of bundles as core B, core A should have approximately twice the eigenvalue separation of core B. From Table B.1, we observe that doubling the eigenvalue separation results in a reduction of decay ratio of the order of 10% to 15%.

The eigenvalue separation, however, depends on many more parameters than just the core size. For instance, super low leakage loading patterns (SL^3P) have very low leakage and result in significantly lower eigenvalue separation than in a core the same size with a conventional loading pattern. Another parameter that affects the eigenvalue separation is the fission cross section [see Eq. (B-1)]; therefore, fuels with high enrichment (to allow for longer refueling cycles) should result in smaller eigenvalue separation that can negate the advantages of the small core.

In summary, the core size is an important parameter that affects the eigenvalue separation, but it is not the only one. It is, thus, hard to justify what the eigenvalue separation of a Solution I-D really is.

Operating experience

An additional argument against Solution I-D is the fact that Swedish reactors [for example, Ringahls-1 (see Reference 5)] have experienced out-of-phase instabilities. Swedish BWRs have very tight inlet orifices and have relatively small cores (for instance, Ringahls-1 has only 648 fuel bundles).

Table B.1. LAPUR-calculated decay ratios as a function of inlet orifice size

		Inlet restriction coefficient (velocity heads)					
		25 vh	30 vh	35 vh	40 vh	45 vh	50 vh
Average-channel decay ratio		0.62	0.57	0.52	0.48	0.44	0.41
Corewide mode decay ratio		0.86	0.85	0.84	0.83	0.82	0.81
Out-of-phase mode decay ratio, if eigenvalue separation is	$\rho = -\$0.5$	1.08	1.02	0.96	0.92	0.87	0.83
	$\rho = -\$1.0$	1.06	0.97	0.90	0.83	0.77	0.71
	$\rho = -\$1.5$	1.01	0.90	0.81	0.73	0.66	0.61

Table B.2. LAPUR5X input for Solution I-D analyses

```
LAPUR5X Test case for BWORO Sol I D
  1
977.0, 490.0, 1000.0, 20.E6, 0.0, 0.0, 0.63, 1.0
  2
1, 24, 0, 1, 0, 0, 0, 1, 0
  3
1, 25
  4
        15.24,  15.24,  15.24,  15.24,  15.24,  15.24,  15.24
        15.24,  15.24,  15.24,  15.24,  15.24,  15.24,  15.24
        15.24,  15.24,  15.24,  15.24,  15.24,  15.24,  15.24
        15.24,  15.24,  15.24,  45.53
  5
        0.95,  1.60,  1.80,  1.70,  1.55,  1.45,  1.30
        1.20,  1.15,  1.10,  1.00,  0.95,  0.92,  0.90
        0.86,  0.83,  0.80,  0.78,  0.72,  0.67,  0.62
        0.50,  0.40,  0.20,  0.00
  7
1,    1
  9
1,   764
 10
1,  30.0
 11
1, -0.280
 13
1,    0.
 14
1,   764
 15
1,    62
 16
1,    1
 17
        1    411.29
 18
        1    238.96
 19
        1    102.09
 20
        1    102.09
 21
        1    1.36
 22
        1    0.1
 23
        1    0.1
 24
        1    1.3
 25
        1    0.125
 26
1,      1
 27
        1    10.42
 28
        1    1.0400
 29
        1    0.5586
 30
        1    0.0373
 31
        1    0.0813
 32
        1    0.1356
 33
        1    0.0114
 34
1,      1
 35
        1    1
 36
             411.29
 37
             1.40
 53
```

```
 1.E-3   1.E-3   1.E-3   2.E-5   1.E-3   1.E-9   1.E-2   5.E-8
54
    1      25
0
```

Table B.3. LAPUR5W input for Solution I-D analyses

```
LAPUR5W Test Case for BWROG Sol I D
1
 1, 1, 1
2
      764
3
      411.29
4
      0.40, -1.5
5
1   6
6
             1
7
    6   1.95E-4   1.10E-3   9.67E-4   2.09E-3   6.58E-4   1.34E-4
8
    6   0.0127    0.0317    0.0115    0.0331    1.40      3.87
9
             1
10
    1     0.0
11
             1
12
    1   3.29E-5
17
    -2.64 E-03
18
    1        7
19
             1      1      1      1      1      1      1
20
          1.2    1.0    0.8    0.6    0.4    0.2    0.0
21
         -2.662   8.006   18.751   23.450   26.545   27.381   27.805
22
7, 0.1, 0.15, 0.2, 0.25, 0.3, 0.35, 0.4
23
 0  1  1  1  1  1  0  0  0  0  0  0  0  0
 0  0  0  0  0  0  0  0
24
 1  1  1  1  1  1  1  1  1
28
      1.
29
3, -0.5, -1.0, -1.5
30
      0
0
```

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1.0 INTRODUCTION

1.1 BACKGROUND

The stability licensing basis for all U.S. nuclear power plants is set forth in GDC-12. This requires assurance that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. Historically, compliance with GDC-12 was achieved by demonstrating that thermal-hydraulic stability induced neutron flux oscillations were not expected. More recently, operating experience indicates that the thermal-hydraulic stability characteristics of a BWR are strongly influenced by a number of design and operating parameters which will vary depending upon the specific plant conditions at the time of the event. This was recognized and recommendations for special operator actions were provided in GE Service Information Letter (SIL) 380 Revision 0 (August 11, 1982) and Revision 1 (February 10, 1984). These recommendations were accepted by the NRC as providing adequate compliance with the detection and suppression provision of GDC-12.

Following the March 9, 1988 LaSalle-2 event, evaluations indicated that under certain conditions, margins to safety limits may be less than previously expected. These findings were promptly reported to the NRC and interim corrective actions (ICAs) recommended by GE and the BWR Owners' Group were implemented at all U.S. plants. NRC Bulletin 88-07 Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)" (Reference 1) endorsed the ICAs and the BWR Owners' Group program to develop generic long-term solutions to the stability issue.

The BWROG program has successfully developed the necessary design and evaluation methodology to analyze thermal-hydraulic stability and has identified several viable approaches to the long-term resolution of the stability issue. Details of this methodology are provided in the body of this report and examples of the current solution concepts are discussed in Section 4.0 and Appendix A.

1.2 HISTORY OF BWROG PROGRAM

Following the 1988 LaSalle-2 event, the BWR Owners' Group and GE undertook a program to evaluate the implications of thermal-hydraulic stability for plant operation. Specifically, an engineering scoping analysis (Reference 2) was performed to evaluate the margin to the MCPR Safety Limits during regional oscillations and determine the plant instrumentation response during the oscillations. That study indicated the potential for a safety limit violation for certain large amplitude regional oscillations. As a result, ICAs were identified to supplement the general guidelines given in GE SIL 380, Revision 1. These ICAs provided the basis for the actions promulgated by the NRC in Bulletin 88-07, Supplement 1, and were promptly implemented by all domestic BWR utilities.

Subsequently, the BWR Owners' Group initiated a program to develop generic long-term solutions to the stability issue. Throughout this program, the NRC has been kept fully informed regarding its progress and direction. Feedback from the NRC has been considered in the selection of specific options for development and in the preparation of a licensing approach. The generic approach to resolving the stability issue was documented in the BWROG Report "Licensing Basis for Long-Term Solutions to BWR Stability" (Reference 3) and a supplement to that report (Reference 4) containing information on two additional solution options.

To provide a statistically-based assessment of the protection provided by various detection and suppression options, the BWROG program supported development of methodology to evaluate oscillation characteristics and plant instrumentation response. This methodology was developed by GE and technically reviewed by the Stability Committee of the BWR Owners' Group and was discussed with the NRC on several occasions. Both GE and the BWR Owners' Group have concluded that this methodology provides a technically sound basis for the design and evaluation of the various long-term solution options.

The BWROG development activities have also produced the long-term solution options described in Appendix A to this report. These options have been discussed with the NRC on numerous occasions. It is the BWR Owners' Group

position that these solutions provide an appropriate level of protection for stability-related neutron flux oscillations and fully comply with the requirements of GDC-12.

1.3 APPROACH TO RESOLUTION OF THE STABILITY ISSUE

Based on guidance from the NRC, the following criteria for resolution of the stability issue were established:

(1) Solutions must satisfy GDC-12.

(2) Solutions should not depend upon development of a new transient boiling dryout correlation.

(3) Some form of automatic suppression function is appropriate (the specific function may vary for different plant type applications).

(4) Solutions should minimize the need for cycle specific analysis which requires NRC review.

(5) Solutions must be applicable to all current fuel designs and operating strategies.

Although observed thermal-hydraulic oscillations in BWRs have not resulted in fuel failures in nearly 800 reactor-years of operation, BWROG studies have shown that regional oscillations could potentially result in conditions that could violate the specified MCPR Safety Limit. While it is recognized that the MCPR Safety Limit is overly conservative for application to thermal-hydraulic oscillations, this limit will be used at this time as the basis for resolution of the stability issue. In fact, additional margin will be added to the safety limit as the design goal for those options which rely on detection and suppression of oscillations. This approach provides a high degree of assurance that the safety limit will not be violated for any expected oscillation event.

Using a conservative design goal is a prudent approach, since the large number of parameters potentially affecting the stability performance of a given plant makes analysis of all possible combinations impractical. The combination of a design goal providing substantial margin to the safety limit and an analysis based on the range of expected operating conditions provides a high degree of assurance that the safety limit will not be violated as a result of events that potentially could occur during the life of a plant.

In addition to detection and suppression based solutions, other solutions have also been studied and are included in this report. One alternative solution provides a trip upon entering a conservatively defined "exclusion region" which encompasses the power and flow conditions in which oscillations could occur. Another alternative, applicable only to a certain class of plants, demonstrates that the mode of potential oscillations can be well characterized by analysis. For this class of plants, existing protective features are shown to provide adequate protection.

The objective of all solutions is to provide automatic protection for oscillations that have the potential to occur in a plant lifetime.

1.4 PURPOSE OF THIS REPORT

This report describes the analytical methodologies developed for the BWROG to resolve the thermal-hydraulic stability issue. Solution concepts using this methodology are also being submitted for NRC review and acceptance (Appendix A). Since introduction of any new protection system has the potential for a significant impact on plant operation, it is the purpose of this licensing topical report to establish the basis for an understanding between the NRC and the BWR Owners' Group on acceptable analytical methodology and solution concepts. NRC acceptance of the methodologies described in Sections 5.0 and 6.0 is required to further optimize solution concepts as the BWR Owners' Group proceeds with the hardware/software design phase. NRC acceptance of the specific concepts described in Section 4.0 and Appendix A is required to allow utilities to select a preferred solution.

2.0 SUMMARY AND CONCLUSIONS

Licensing methodology and long-term solution concepts developed by GE in support of the BWROG Stability Program are described. The methodologies and solution concepts consider both the prevention and the detection and suppression approaches and provide the basis for protection system designs which are applicable to all BWRs in the US. It should be noted that several of the concepts take advantage of unique features of a particular class of plant and are not generically applicable. Other concepts apply to all plants and provide flexibility in the selection of a solution approach for all plants.

The stability methodologies described in Sections 5.0 and 6.0 and the long-term solution concepts described in Appendix A are being submitted for NRC review and approval. Other applications of these methodologies are possible and alternate concepts may be proposed in the future.

Solution descriptions and general requirements are summarized in Section 4.0 and are described in more detail in Appendix A. The methodology used to define the range of power/flow conditions under which stability related oscillations are expected is discussed in Section 5.0. This methodology provides the basis for defining the region in the power/flow map in which operation will not be allowed under prevention options. Section 6.0 discusses the methodology used to support the detection and suppression options which take advantage of the ability to detect oscillations and to initiate appropriate actions to suppress them.

3.0 DEFINITIONS

3.1 POWER/FLOW MAP

The power/flow map (Figure 3-1) depicts the possible power and flow combinations for a GE BWR.

Rod lines (e.g., line A on Figure 3-1) are lines of constant control rod configuration and xenon concentration that are traversed by changes in recirculation (core) flow. Increases in recirculation flow cause both reactor core flow and power to increase. Conversely, decreasing flow causes movement down line A, thereby decreasing power.

Line B on Figure 3-1 represents power changes without flow changes accomplished by (1) control rod manipulation, (2) feedwater temperature changes, and (3) xenon concentration changes. Withdrawal of control rods, a reduction in feedwater temperature, or a decrease in xenon concentration (core-wide relative to previous conditions) produces an increase in power. Insertion of control rods, an increase in feedwater temperature, or an increase in xenon concentration causes power to move down line B.

3.2 AUTOMATIC SUPPRESSION FUNCTION (ASF)

The ASF associated with any of the stability solutions initiates control rod insertion without operator actions, such that the region of potential instability (Figure 3-2) is exited quickly or oscillations are suppressed prior to violation of the MCPR Safety Limit.

In addition to full reactor scram, some BWR designs have the capability to automatically insert a limited number of control rods using Select Rod Insert (SRI). Control rods selected for the SRI function are scrammed upon receipt of the initiation signal with an associated rapid power reduction. The power reduction that can be achieved with SRI depends on the number and location of the control rods selected, the control rod pattern and the cycle exposure.

3.3 PEAK-TO-MINIMUM/AVERAGE ((P-M)/A)

During oscillations, a minimum (M) and peak (P) power will be observed. The difference is the peak-to-minimum value (P-M). To provide a relative measure of the oscillation, the peak-to-minimum value can be divided by the average (A). This provides a measure of the oscillation magnitude normalized to its average value during the oscillations. (P-M)/A is a term that is applied to many parameters such as bundle powers, LPRM signals, etc.

3.4 PLANT GROUP

A Plant Group is comprised of those plants which are similar with respect to parameters important to thermal-hydraulic stability. Evaluations performed for a representative plant are applicable to all plants within the group.

3.5 DECAY RATIO/GROWTH RATE

Decay ratio is a measure of the stability of an oscillating system and is defined as the value of one peak in the oscillation to the amplitude of the peak immediately preceding it (e.g., x_n/x_{n-1}). The amplitude is measured relative to the average amplitude of the signal (x = P-A). A stable system is characterized by a decay ratio of less than 1.0; an unstable system has a decay ratio greater than 1.0 (Figure 3-3). Decay ratios greater than 1.0 are referred to as growth rates.

3.6 OSCILLATION CONTOUR

An oscillation contour is the spatial distribution of oscillation magnitudes in the core. It will generally be expressed as a normalized value, (P-M)/A. For core-wide oscillations, the oscillation contour is uniform across the core, since, at each point in the core, in the x-y plane, the oscillation magnitude normalized to the initial steady-state value at that point is the same. For regional oscillations, the (P-M)/A value varies in the x-y plane.

3.7 ACRONYMS AND ABBREVIATIONS

ANF	-	Advanced Nuclear Fuels
APRM	-	Average Power Range Monitor
ARI	-	Alternate Rod Insertion
ASF	-	Automatic Suppression Function
ATWS	-	Anticipated Transient Without Scram
BOCn	-	Beginning-of-Cycle n
BOEC	-	Beginning-of-Equilibrium-Cycle
BWR	-	Boiling Water Reactor
BWROG	-	Boiling Water Reactor Owners' Group
CMFLPD	-	Core Maximum Fraction of Limiting Power Density
CPR	-	Critical Power Ratio
DR	-	Decay Ratio
EOCn	-	End-of-Cycle n
EOEC	-	End-of-Equilibrium-Cycle
EW	-	East-to-West
FCV	-	Flow Control Valve
FFWTR	-	Final Feedwater Temperature Reduction
FMCPR	-	Final Minimum Critical Power Ratio
FRTP	-	Fraction of Rated Thermal Power
FWHOS	-	Feedwater Heater Out-of-Service
GDC	-	General Design Criteria
GE	-	General Electric Company
ICA	-	Interim Corrective Actions
ICPR	-	Initial Critical Power Ratio
LBS	-	LPRM-Based System
LCO	-	Limiting Condition for Operation
LOFH	-	Loss of Feedwater Heating
LPRM	-	Local Power Range Monitor
MCPR	-	Minimum Critical Power Ratio
MOCn	-	Middle-of-Cycle n
MOEC	-	Middle-of-Equilibrum-Cycle
MSIV	-	Main Steam Line Isolation Valve
NESW	-	Northeast-to-Southwest
NMS	-	Neutron Monitoring System

NS	-	North-to-South
NUMAC	-	Nuclear Measurement and Control
NWSE	-	Northwest-to-Southeast
OPRM	-	Oscillation Power Range Monitor
PRM	-	Power Range Monitor
RBM	-	Rod Block Monitor
RPS	-	Reactor Protection System
RPT	-	Recirculation Pump Trip
SER	-	Safety Evaluation Report
SRI	-	Select Rod Insert
STPM	-	Simulated Thermal Power Monitor (STP, TPM)
WRNM	-	Wide Range Neutron Monitor
3D	-	Three-Dimensional

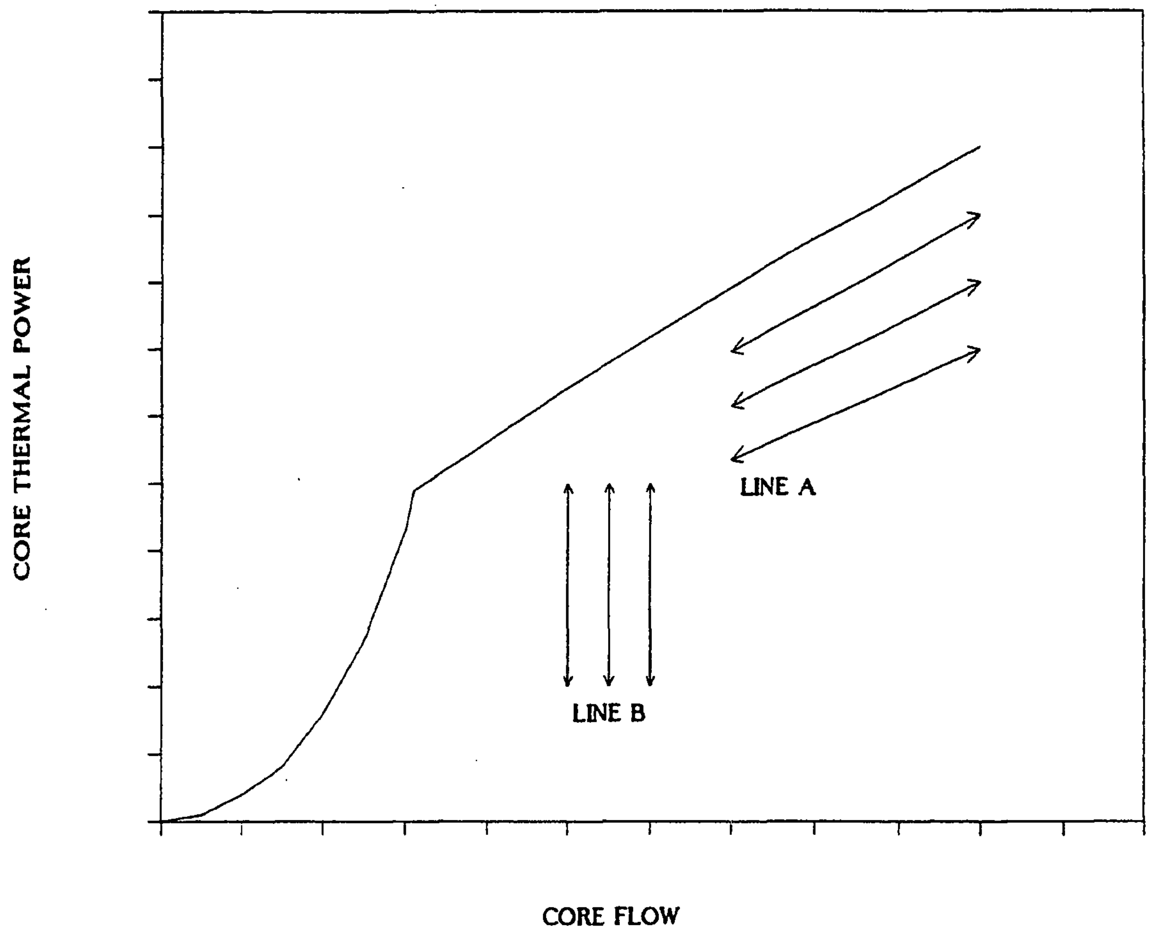


FIGURE 3-1. POWER/FLOW MAP

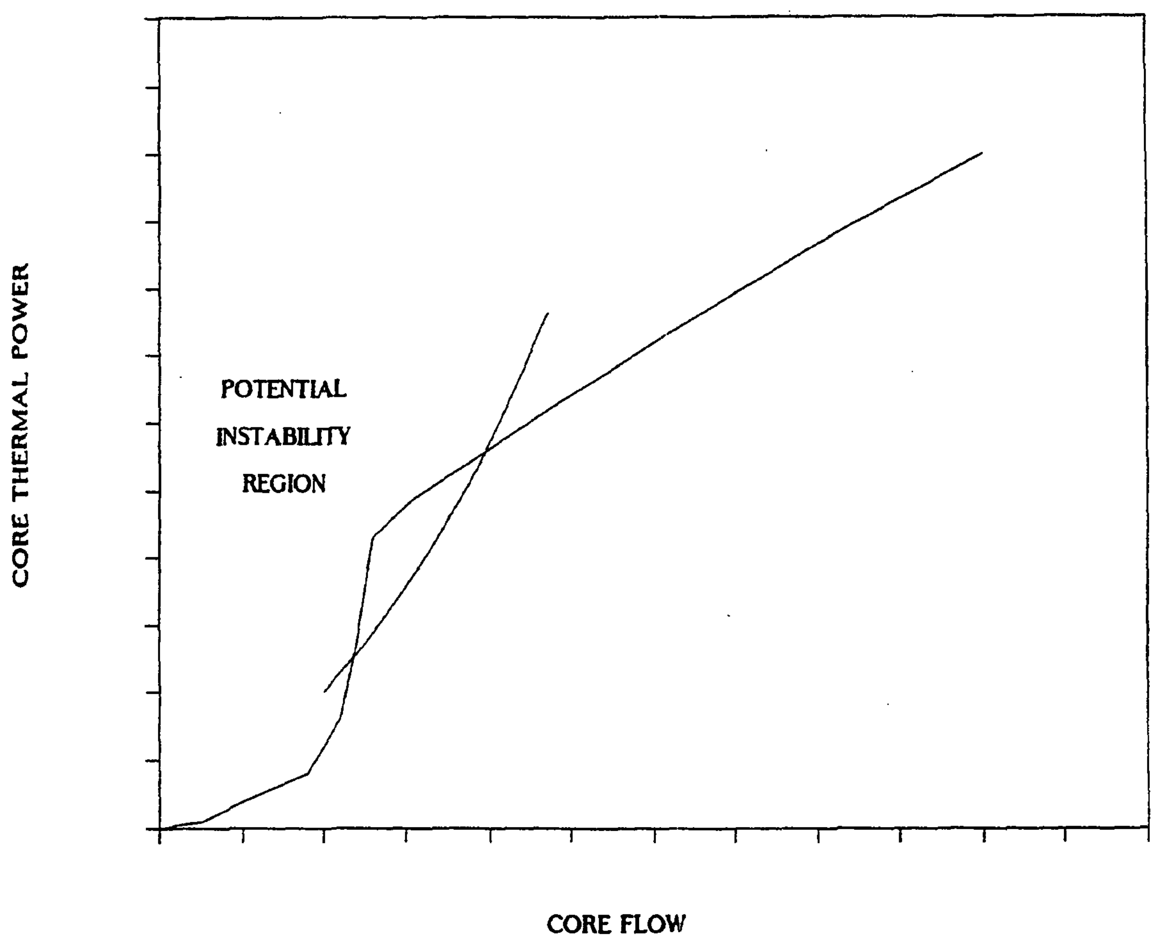


FIGURE 3-2. POTENTIAL INSTABILITY REGION

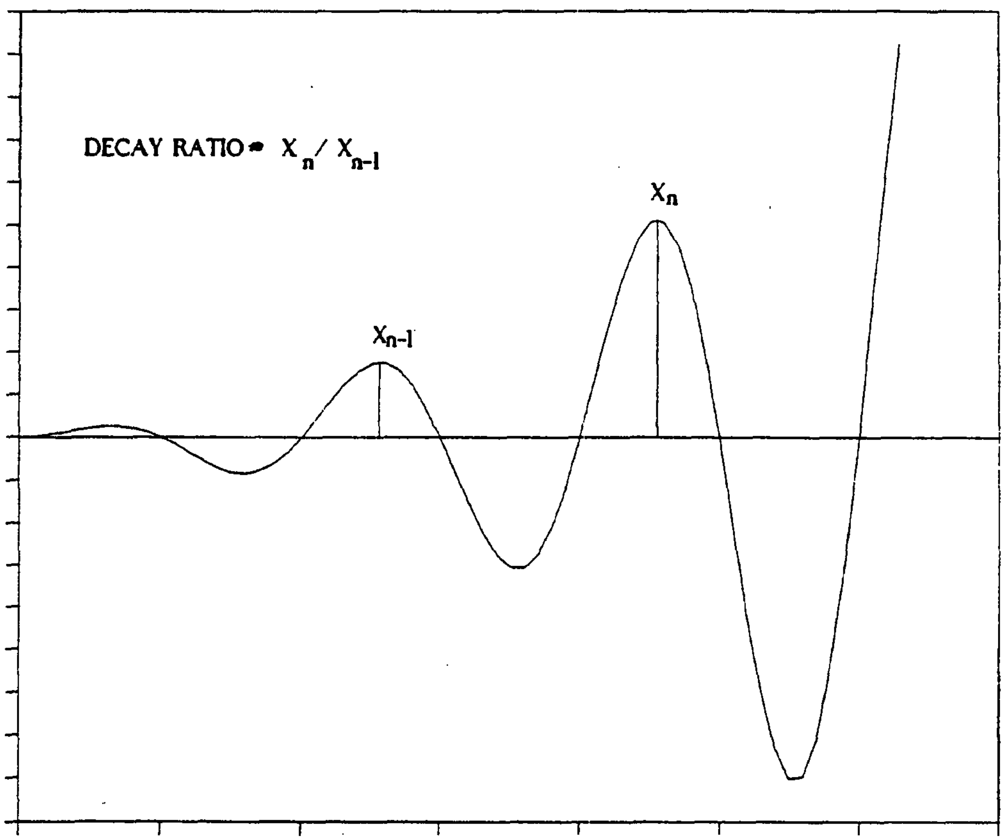


FIGURE 3-3. DECAY RATIO/GROWTH RATE

4.0 SOLUTION DESCRIPTIONS AND GENERAL REQUIREMENTS

4.1 SOLUTION CONCEPTS

GDC-12 states that the reactor and associated protection systems must be designed such that power oscillations are not possible, or can be readily detected and suppressed without exceeding specified fuel design limits. Compliance with GDC-12 can be demonstrated by calculating decay ratios for allowable operating conditions and restricting operation, if necessary, so that potentially unstable power/flow conditions are not encountered. This is called the prevention approach.

Alternatively, the detection and suppression approach may be used to satisfy GDC-12 by using existing, new, or modified plant instrumentation to detect and suppress oscillations prior to exceeding fuel design limits. The typical detection and suppression system monitors local or average neutron flux, and initiates an Automatic Suppression Function (i.e., scram or Select Rod Insert) when oscillating signals reach a predetermined level.

The prevention and detection and suppression concepts, either individually or in combination, can be applied to create a long-term solution to the stability issue. Since December 1988, the BWROG Stability Committee's primary objective has been to develop long-term solutions that meet regulatory requirements while minimizing unacceptable operating and modification impacts. Based on several BWROG and GE discussions with the NRC, automatic protection has been incorporated as an essential element of the long-term solutions proposed in this report.

Because of the variety of plant types, and the need to accommodate differing operating philosophies and owner-specific concerns, several solution alternatives are being pursued. For several specific BWR units, existing systems and plant features already provide sufficient detection and suppression of reactor instabilities. For these units, the methodology described in Sections 5.0 and 6.0 (or similar methodology described

separately by the BWR/2 owners) is applied to demonstrate the sufficiency of existing systems. For most BWR units, new or modified plant systems will be required. A summary description of all BWROG long-term solutions is provided in Section 4.3.

4.2 SUPPORTING METHODOLOGY

Implementation of either or both of the prevention and detection and suppression concepts requires specific analyses to determine or confirm the acceptability of the allowable operating region and to confirm the acceptability of detection and suppression systems and trip setpoints. These analyses create a new safety analysis basis.

Application of the prevention concept requires use of computer models, such as frequency domain computer codes, which can calculate accurate decay ratios. Both the fundamental mode of oscillation and any higher order modes which are expected must be considered in the methodology. Decay ratio analyses must account for uncertainties in the computer model employed, and variations and uncertainties in key input parameters such as power distribution and moderator density reactivity coefficient. Various fuel dimensions and thermal-mechanical designs must also be considered. The GE/BWROG approach uses the FABLE code (Reference 5) and an overall conservative methodology to calculate core and channel decay ratios as a function of core power and core flow. Confirmatory analyses are used to demonstrate the conservatism of this approach (Section 5.0).

Detection and suppression schemes use the Local Power Range Monitors (LPRMs) to generate a signal that is monitored for oscillations. The LPRMs can be monitored individually, or combined in a number of different ways. A detection and suppression methodology must consider how the LPRMs respond in time and space to expected modes of oscillations. It must also consider the impact of LPRMs out of service, the failure of a single protection subsystem channel, and the time it takes to detect an oscillation and automatically achieve suppression.

The MCPR response of limiting fuel bundles is compared to the MCPR Safety Limit to determine the acceptability of the fuel response to an oscillation. Use of the MCPR Safety Limit as the design criterion is conservative because fuel cladding damage is not likely to occur during brief periods of departure from nucleate boiling followed by the quenching that occurs during BWR density wave oscillations. Thermal-hydraulic testing has confirmed the conservatism of the MCPR Safety Limit under these conditions.

To demonstrate that the MCPR Safety Limit is not violated during oscillations, a methodology has been developed which relates LPRM response to the thermal-hydraulic response of the limiting fuel bundle in the core during oscillations. The uncertainties in this relationship, and the range of initial plant parameters that impact fuel thermal margin, must also be considered.

Both the prevention and detection and suppression methodologies must accommodate differences in fuel bundle designs, reload core designs, and operating strategies. The GE/BWROG approach accomplishes this by using conservative analyses and requiring reload verification of certain cycle-specific parameters when necessary.

4.3 SOLUTION DESCRIPTIONS

Summary descriptions of all long-term solutions considered by the BWROG follow. Five of these solutions, Options I-A, I-C, I-D, III and III-A, are described in more detail in Appendix A.

Solution descriptions for Options I-A, I-B, I-C, I-D, II, and III were provided previously in References 3 and 4. Options I-B and II have not changed substantially since the previous report and are therefore not included in this report. This report recognizes that alternate designs which use the methodology described in Sections 5.0 and 6.0 may be developed later to improve the sensitivity to instabilities or to further reduce the possibility of a plant scram from a false trip signal. Additionally, all common

analytical methodology sections are now consolidated in Sections 5.0 and 6.0, and are not repeated in the solution descriptions.

All Option I solutions have the common feature of restricting operation in a predefined region of the power/flow map. Option III-A has not been previously documented, but is a variation on the use of individual LPRMs for detecting instabilities.

4.3.1 Option I-A: Regional Exclusion

The regional exclusion option is designed to prevent operation in the high power/low flow region of the power/flow map by initiating an ASF upon entry into the region. Analytical methodology described in Section 5.0 is used to define the power and flow boundaries of the region. The trip function is provided by a modified APRM flow-biased system which would result in either a scram or automatic exit from the region. A rod block would occur before the region is entered to provide a warning to the operator and prevent inadvertent entry resulting from control rod withdrawal. This option satisfies GDC-12 by preventing the onset of oscillations, and is further described in Appendix A.

4.3.2 Option I-B: Regional Exclusion With Stability Monitor

This solution is the same as Option I-A with an option to bypass the ASF and enter the exclusion region if a stability monitor shows sufficient decay ratio margin. The decay ratio margin would have to be sufficient to accommodate a flow reduction or loss of feedwater heating event inside the region. This option is considered viable, but not as desirable as other options. It is being pursued by the BWROG on a low priority basis, and is not discussed in-depth in this report and approval is not requested.

4.3.3 Option I-C: Administratively-Controlled Regional Exclusion With Modified APRM Flux Trip

This option uses an administratively-controlled exclusion region and a new APRM-based neutron flux trip system. The region will be identical to the region defined in Option I-A, and is administratively-controlled so that planned operation within the region is avoided. If an unplanned operational event results in the exclusion region being entered, the new neutron flux trip system will be automatically armed, and an oscillation that might occur would be automatically detected and suppressed before the MCPR Safety Limit is violated.

The Option I-C APRM neutron flux trip occurs at a specified level above the average APRM value. A trip is expected only if oscillations occur, or if normal neutron flux noise or some other non-stability perturbation creates a sufficient flux increase. Option I-C uses both administratively-controlled prevention and automatic detection and suppression to meet GDC-12. It has the potential operating advantage over Option I-A of not tripping every time the exclusion region is entered.

Recent estimates of the trip setpoint for this option were made using the detection and suppression methodology described in Section 6.0. These estimates indicate that the setpoints would be sufficiently near the neutron flux noise levels as to make this option less desirable than anticipated for some plant types. This option is further described in Appendix A.

4.3.4 Option I-D: Administratively-Controlled Regional Exclusion with Flow-Biased APRM Neutron Flux Scram

This option uses both administratively-controlled prevention and automatic detection and suppression to assure compliance with GDC-12. It is applicable to four or five U.S. BWRs that have relatively tight fuel inlet orificing. The solution uses a flow-biased APRM neutron flux scram system. The inlet orificing results in core-wide oscillations being the dominant mode for these units.

During planned operations, GDC-12 compliance is accomplished by administratively avoiding the region of potential instability (as defined by the Section 5.0 methodology), thereby preventing oscillations from occurring. If that region is entered as the result of an unplanned operational event, the flow-biased APRM neutron flux scram system provides direct detection and suppression of core-wide oscillations prior to exceeding the MCPR Safety Limit. Confirmation of appropriate protection for the low probability regional oscillations will be demonstrated using the detection and suppression methodology described in Section 6.0. This option is described further in Appendix A.

4.3.5 Option II: Quadrant APRM (BWR/2)

This option demonstrates that the existing quadrant-based APRM system of the BWR/2 plant types will initiate a reactor scram early enough to avoid violating the MCPR Safety Limit if oscillations should occur. The BWR/2 APRM system is unique in that LPRM instrument assignments to the APRMs are arranged in separate quadrants of the reactor. BWR/2s, therefore, have a substantial APRM response to a postulated regional oscillation. The instrumentation design, coupled with the low BWR/2 power density, readily supports a detection and suppression method of meeting GDC-12.

The analyses confirming the adequacy of this approach are similar to the other detection and suppression options and are described in depth in Appendix C of Reference 3. Plant specific submittals will be made to justify the Quadrant APRM system for BWR/2s.

4.3.6 Option III: LPRM Based Oscillation Power Range Monitor

This option, designated an Oscillation Power Range Monitor (OPRM), uses a microprocessor to monitor groups of LPRM signals. The grouping of LPRM signals and related system description information is described in more detail in Appendix A. Upon identification of neutron flux oscillations characteristic of a thermal-hydraulic instability, the system initiates an ASF that suppresses the oscillation. The system uses an algorithm which contains sufficient logic to prevent actuation by most or all non-instability plant

events. The ASF is automatically bypassed at high flow or low power to avoid spurious actuations.

Detection and suppression compliance with GDC-12 will be demonstrated for all expected modes of oscillations, using the analytical methodology described in Section 6.0. By taking advantage of the strong response associated with a local power-based detection system, the system will be designed to trip only if a true thermal-hydraulic oscillation occurs; spurious or noise related signals should not actuate an OPRM trip. No restriction on power and flow operations will be required.

4.3.7 Option III-A: Alternative LPRM-Based System

This option, designated a LPRM-Based System (LBS), is very similar to Option III. It also uses a microprocessor to monitor groups of LPRM signals and initiates an ASF upon identification of neutron flux oscillations characteristic of a thermal-hydraulic instability. The main difference between Options III and III-A is in the number and grouping of LPRM signals in the various reactor protection system channels. Details of the Option III-A design concept are contained in Appendix A. Similar to Option III, detection and suppression compliance with GDC-12 is demonstrated using the methodology described in Section 6.0.

5.0 REGIONAL EXCLUSION LICENSING METHODOLOGY

The objective of the Regional Exclusion Licensing Methodology is to define a power/flow region where instability can occur. The boundary of this exclusion region is established through the use of an analysis procedure which is demonstrated to be conservative relative to expected operating conditions. This procedure does not attempt to define input values which are individually bounding with respect to stability, but uses a combination of inputs which together yield calculated decay ratios that are conservative relative to expected operating conditions.

In addition to steady-state operating conditions, operational events may result in unstable conditions. Therefore, a set of event-based calculations has been performed to confirm that the procedure adequately addresses anticipated operational occurrences. The limiting events analyzed are core flow reduction and loss of feedwater heating (LOFH).

This approach may be used as a solution in its entirety, where the exclusion region boundary is enforced by an ASF that will result in exiting the exclusion region upon entry. This approach may also be used to define a region outside of which automatic protective functions, designed to detect and suppress oscillations, are not required to be operational. Finally, this approach may be used to define a region where extra precautions may be warranted, or a region that is to be administratively avoided to provide an additional level of protection for a solution.

5.1 STABILITY CRITERIA

Calculations of core and channel decay ratios will be performed using the FABLE/BYPSS code and compared to stability criteria to define the exclusion region boundary. The FABLE/BYPSS code is a frequency domain code which is used to calculate the channel and core decay ratios. This model is described in Reference 5. The FABLE/BYPSS code has been qualified against test data using best estimate inputs to represent the plant operating conditions. The qualification showed good agreement at high decay ratios with an overall

conservative bias. The bias was examined as a function of power, flow, and power density. A bias correction was then applied to the calculated values to determine a best estimate decay ratio. Results of the qualification are shown in Figure 5-1. Decay ratios calculated by this procedure are accurate to within a standard deviation of 0.08. A model uncertainty of 0.2 was applied to the calculation of core and channel decay ratios. This is an uncertainty of more than two standard deviations and is consistent with NRC approved uncertainties for licensing calculations using the FABLE code (References 6 and 7). The wide range of possible operating conditions are addressed by the selection of input conditions for the analysis (Section 5.2).

The FABLE/BYPSS code is used to directly calculate both core-wide (in-phase) and channel decay ratios for various modes of instability. Post-event analyses of the Garigliano, Vermont Yankee, Caorso, and Leibstadt stability tests and the LaSalle Unit 2 operating event have been used to establish a relationship between core and channel decay ratios (Figure 5-2). Each data point in this figure represents a limit cycle oscillation in the mode specified. Regional oscillations have occurred for FABLE/BYPSS calculated core and channel decay ratios less than 0.8.

To better understand regional oscillations, a separate frequency domain analytical model was developed which included a power feedback transfer function for individual channel hydrodynamic calculations. This model was used to estimate the regional decay ratio by simulating the power feedback from a group of channels. This feedback, when combined with a single channel hydrodynamic calculation, generates a decay ratio indicative of regional instability. This model was benchmarked against the above test data with good results. Using this model, the combination of core and channel hydrodynamic decay ratios which could potentially result in regional oscillations was mapped (Figure 5-3). From the results, for channel decay ratios less than 0.5 the effective channel decay ratio when the power feedback from surrounding channels was included was never greater than 1.0, indicating a threshold below which regional oscillations are not expected to occur.

Based on FABLE/BYPSS qualification to test data (Figure 5-1), Caorso and Leibstadt test data, and the estimated regional decay ratio calculations (Figures 5-2 and 5-3), Figure 5-4 was developed as the criteria for determining the potential for the various modes of instability based on FABLE/BYPSS calculated decay ratios. The cross-hatched region of the figure represents a region of potential instability. The lower boundary of this region is used as the stability criterion. Core and channel decay ratios at or below this lower boundary are considered to meet the criteria. Those conditions that lie above the lower boundary do not meet the criteria.

5.2 REGION BOUNDARY DEFINITION PROCEDURE

To determine a region boundary, FABLE/BYPSS calculations are performed over a range of power/flow conditions to determine a line of constant stability margin as defined by the stability criteria. Inputs and calculational procedures are chosen to collectively provide conservative results relative to stability. As such, some inputs are best estimate and others are conservative. The following sections describe the inputs for the FABLE/BYPSS procedure.

5.2.1 <u>Void Coefficient</u>

The most negative point model nuclear void coefficient (in terms of nuclear void coefficient/delayed neutron fraction) in the cycle is used. This void coefficient is based on a representative core and fuel design for the specific plant group being evaluated. The void coefficient is transformed into a moderator density reactivity coefficient for input to FABLE. Since this void coefficient will not necessarily correspond to a point in the cycle which produces the least stable condition based on other input parameters (e.g., power distribution), this is a conservative input. Since void coefficient may vary as a function of core and fuel designs, sensitivity studies are performed to determine the sensitivity of the region boundary to these changes.

5.2.2 Thermal-Hydraulics Data

Standard design values for thermal-hydraulics data are used in the analysis. These values are consistent with GE methods for other transient and accident analyses and are necessary to ensure consistency between the various analytical calculations performed for a stability analysis (e.g., nuclear and thermal-hydraulic).

5.2.3 Axial Power Shapes

(1) Hot Channel Decay Ratio

The axial power shapes for calculation of the maximum channel decay ratio are shown in Figure 5-5. Sensitivity studies show that the channel decay ratio is greater with a bottom-peaked axial power shape. The hot channel power shapes shown in Figure 5-5 conservatively bound expected operating conditions. The axial power shapes assumed for the hot channel are dependent on plant conditions. For forced circulation operation that is representative of startup conditions, a highly bottom peaked axial power shape is assumed. For natural circulation operation resulting from the trip of both recirculation pumps, a somewhat less severe bottom-peaked power shape is assumed, since the initial high power conditions have less peaked axial power shapes. For operation at relatively high power/flow conditions, a less peaked power shape is also used to reflect the expected range of conditions. Figures 5-6 through 5-9 compare the hot channel axial power shape assumed in the procedure with actual power shapes encountered during instabilities at Caorso, Leibstadt and LaSalle Unit 2.

(2) Core Decay Ratio

FABLE/BYPSS sensitivity studies for core decay ratio show that one of the limiting axial power shapes for core stability is a flat shape. The end-of-cycle (EOC) Haling power distribution at rated power/flow conditions (all control rods withdrawn) has a relatively flat shape compared to other exposures. Therefore, the EOC Haling full power core average axial power distribution is chosen for the core decay ratio calculations at all

power/flow conditions with forced circulation. As core flow is reduced, the axial power distribution will tend to shift towards the bottom of the core and the use of a full power/flow Haling power shape will tend to be overly conservative. Therefore, for conditions at natural circulation flow, an EOC Haling calculated at natural circulation is used. Figures 5-10 through 5-12 show examples of the core average axial power distributions, with some comparisons to actual plant conditions during instabilities (Caorso, Leibstadt and LaSalle Unit 2).

The axial power shapes (hot channel and core average) are independently selected to provide conservative estimates of the channel and core decay ratios, even though the combination of power shapes is very unlikely (i.e., bottom-peaked channel with a flat core average power shape). Because the stability criteria are based on a combination of core and channel decay ratios, simultaneously maximizing both decay ratios is a conservative assumption.

5.2.4 Radial Power Distributions

A minimum of eight channel groups are used to model the radial power distribution. In general, one group is used for the peripheral bundles, six groups for the average channels, and one group for each hot channel for each fuel type. The radial power distribution is generated by the 3D BWR Core Simulator Code (Reference 8) based on an EOC Haling (all control rods withdrawn) condition. The number of bundles in each average channel group is determined to ensure that the total power is evenly distributed among the average channel groups. The average channel groups are comprised of channels of the same fuel type. The radial peaking factor for the hot channels is chosen to bound those calculated from the EOC Haling condition. Since radial peaking factor may vary as a function of core and fuel design and operating strategy, sensitivity studies are performed to determine the variation of the region boundary as a function of radial peaking factor.

5.2.5 Pellet-Clad Gap Conductance

Core average pellet-clad gap conductances are determined for each fuel type at the appropriate core thermal power condition using currently approved licensing models. A multiplier (1.6 for gap conductances calculated with the current licensing model) is applied to the gap conductance to provide a value consistent with models used during the comparison of FABLE/BYPSS to test data.

5.2.6 Other Inputs

Additional required inputs such as plant heat balance data, recirculation loop resistance and fuel physical parameters and material properties are based on standard design values.

5.2.7 Conservatism of Procedure

The Region Boundary Definition Procedure along with the criteria defined in Section 5.1, provides a conservative method for predicting an exclusion region. Results generated using the procedure have been compared to previous stability tests and shown to provide a conservative estimate of the core and channel decay ratios calculated using best estimate inputs of the actual measured plant conditions. For the Vermont Yankee Cycle 8 stability tests, the Peach Bottom Cycle 3 stability tests, the Caorso instability event in June 1982 and the Leibstadt Cycle 1 stability tests, the procedure always resulted in conservative core and channel decay ratios (average of 0.20 higher decay ratio). In addition to these demonstrations of the procedure conservatism, confirmation calculations have been performed for a BWR/6 plant (Section 5.4) to demonstrate that, under more realistic combinations of input conditions, the region boundary defined by the procedure provides margin to potential instabilities.

5.3 APPLICATION OF REGION BOUNDARY METHODOLOGY

Stability is dependent on many core and fuel parameters, as well as operational strategies that may affect power distribution. Those parameters that have the most significant effect on stability have been identified by

various sensitivity studies performed over the years. Some of these parameters, specifically axial power distribution, have been conservatively specified in the procedure to eliminate the need to perform additional sensitivity studies. Other parameters such as void reactivity coefficient, radial peaking factor and fuel assembly design may still vary from plant to plant and cycle to cycle. To develop a generic basis for the definition of exclusion regions, plant groups are defined which share a variety of common features. In general, these plant groups closely follow the BWR product lines (e.g., BWR/3) because of their similar features. A representative plant is chosen from the plant group to form the basis for the generic region boundary definition for the particular plant group. Sensitivity studies are performed to allow other plants within the group to apply the same generic boundary, with modifications to the boundary where appropriate to account for differences in plant, fuel, and core designs.

For each plant group, a set of calculations is performed for the representative plant to define the location of the boundary. This is anticipated to require calculations at approximately three to four power/flow points along the boundary per plant group. The location of the boundary is defined as the collection of power/flow points at which the procedure defined in Section 5.2 produces a combination of core and channel decay ratio which meets the stability criteria (Figure 5-4). Typically, calculations will be performed along a constant rod line, at successively lower core flows until a point is found that meets the criteria of Figure 5-4. The region boundary point along this rod line may be determined by interpolating between two analyzed points, one below the stability criteria and one above the criteria of Figure 5-4. Additionally, the two points should be separated by no more than 5% of rated core flow to ensure reasonable linearity for interpolation. A similar procedure may be used by analyzing points with successively higher power, at constant core flow. Again, interpolation may be used provided the two analyzed points straddle the criteria of Figure 5-4 and are separated by no more than 5% power. An example application of the methodology has been performed for the BWR/6 plant group (Table 5-1), with the Perry plant chosen as the representative plant for analysis.

5.3.1 BWR/6 Region Boundary Definition

The important stability characteristics of the Perry plant compared to the other BWR/6 plants in the U.S. are shown in Table 5-1. Since fuel design, void coefficient, and radial power distribution may vary from cycle to cycle, these parameters must be considered separately when determining the applicability of a generically-determined region boundary. In general, the important stability features are common to the BWR/6 plants and Perry is a representative plant. The region boundary calculations were performed for two cycles of operation at Perry to cover the range of potential void coefficients that may be expected as a plant operates from early cycles to equilibrium cycle conditions. Cycle 2 and a hypothetical equilibrium cycle were evaluated for Perry.

(1) Perry Cycle 2 Evaluations

The Cycle 2 evaluations were based on the actual Perry as-loaded core which contains GE 8x8 fuel. EOC2 Haling power shapes were generated at rated conditions using the GE 3D BWR Simulator. These power shapes are used in the radial power distribution and core average axial power distributions per the procedure defined in Section 5.2. Figure 5-13 shows the full power Haling core average axial power shape which is used for the average channel power shapes when evaluating conditions at forced circulation that are generally expected during startup. Figure 5-13 also shows the calculated core average axial power shape that results from a flow runback to natural circulation from the full power Haling condition. This axial power shape is used for the average channels when evaluating natural circulation conditions. The most negative void coefficient during the cycle is chosen for all analyses and occurs at EOC-1000 MWd/ST.

For Cycle 2, there are two fuel types and, therefore, nine channel groups are used in the analysis. The nine channel groups comprise two hot channels (one for each fuel type), six average channels (three per fuel type) and one peripheral channel. The number of fuel bundles for each average channel group is chosen to ensure that each channel group has the same total integrated power.

The first set of calculations was performed along a high rod line, at points 1 and 2 shown in Figure 5-14. For these forced circulation conditions, the hot channel axial power shape is shown in Figure 5-5. The average and peripheral channels use the EOC Haling core average axial power shape at rated power, as calculated by the GE 3D BWR Simulator (Figure 5-13). The core and channel decay ratios for these two conditions are shown in Table 5-2. Similar calculations were performed along a lower rod line at points 3, 4, 4A, and 4B shown in Figure 5-14. These results are also summarized in Table 5-2.

The last set of calculations was performed at natural circulation conditions at points 5 and 6 shown in Figure 5-14. For these conditions, the EOC Haling core average axial power shape at natural circulation conditions was used for the average and peripheral channels. The hot channel axial power shape is also different from the assumption for the forced circulation calculations (Figure 5-5). The results of these calculations are also summarized in Table 5-2.

The results of the Cycle 2 region boundary calculations relative to the stability criteria defined in Section 5.1 (Figure 5-4) are summarized in Figure 5-15. The region boundary is determined from the intersection of the results with the criteria limits. For the Cycle 2 results, the region boundary is determined by interpolation between points 1 and 2, 4 and 4A, and points 5 and 6, respectively. The region boundary can therefore be defined for Cycle 2 as the line drawn through these points. Figure 5-16 shows the Cycle 2 boundary for Perry.

(2) Perry Equilibrium Cycle Evaluations

A projected equilibrium cycle for Perry was evaluated containing all GE 8x8 fuel. The same basic procedure was used to perform the equilibrium cycle calculations as was used in the Cycle 2 analysis. Figure 5-17 and Table 5-3 summarize the points analyzed for the equilibrium cycle. The decay ratios are compared to the stability criteria in Figure 5-18 and the region boundary is determined by interpolation between points 7 and 8, 10

and 11, and 12 and 13. The region boundary defined by the equilibrium cycle analysis is shown in Figure 5-19.

5.4 REGION BOUNDARY CONFIRMATION

The procedure defined in Section 5.2 uses a combination of best estimate and conservative inputs to generate conservative core and channel decay ratios relative to the stability criteria defined in Section 5.1. The objective of the procedure is to define a region boundary outside of which the probability of oscillations is acceptably low. To confirm that the procedure is conservative, a series of calculations has been performed using actual combinations of inputs from predicted reactor operating conditions. These conditions cover a wide range of potential operating states that would be expected to occur for a plant. The conditions include startup operations, flow runbacks from full power operation and loss of feedwater heating events that are initiated near the region boundary.

During startup, control rods are withdrawn to establish a pattern that is consistent with the pattern expected at rated power. In general, these control rod patterns are established at low core flows to allow the maximum flexibility in obtaining the rated control rod pattern. During such operations, it is expected that quasi-steady-state conditions near the region boundary may be established and it must be demonstrated that these conditions are appropriately bounded by the procedure assumptions. Calculations have been performed using actual and predicted startup control rod patterns at various cycle exposures to simulate the power distributions and reactor conditions expected during startup operations.

The majority of an operating cycle consists of full power operation with control rod patterns that ensure all operating limits are satisfied. Should a reactor shutdown be required, or an inadvertent flow reduction occur that does not result in entering the exclusion region, operation near the region boundary may occur. Under these conditions, the power distribution is determined by the rated power control rod pattern and the power distribution change due to the flow runback. A set of calculations has been performed starting from full power conditions (equilibrium xenon) and simulating a flow

reduction that ends near the region boundary. These calculations define the inputs for the stability analysis.

The final confirmation calculations that have been performed involve the potential for a loss of feedwater heating event (LOFH) that could occur while operating near the region boundary. Although in general only a small fraction of the cycle will result in operation near the region boundary, stability is known to be sensitive to the increase in inlet subcooling caused by loss of feedwater heating. For expected LOFH events near the region boundary (expected temperature loss defined to be 60°F), an evaluation was performed to determine the resultant decay ratios at the final operating conditions of the LOFH event. It was assumed that the event resulted in conditions that were near the region boundary but did not enter the exclusion region. Entering the region would result in initiation of an ASF or would enable the trip function of a detection and suppression system thereby providing automatic protection.

For steady-state and event-based calculations used to confirm the restricted region boundary, the same basic procedure as outlined in Section 5.2 was used. However, the axial and radial power distribution and void coefficient were based on calculations using the GE 3D BWR core simulator code (Reference 8) at the conditions chosen for the analysis (i.e., control rod pattern, cycle exposure, etc.).

5.4.1 Startup Condition Evaluations

For BWR/6 plants, the startup path begins with the recirculation pumps on low speed, with the flow control valves (FCV) at their maximum open position. Control rods are withdrawn until sufficient power and feedwater flow is established to clear interlocks designed to prevent pump cavitation should core flow increase beyond a minimum value. These interlocks are typically set near 30% of rated core thermal power. Once these interlocks are cleared, the FCVs are closed to their minimum position and the associated recirculation pump is transferred to high speed. The location of the region where these upshifts are performed, compared to the exclusion region boundary is shown in Figure 5-20. Five cases were performed at these conditions, covering Cycle 2 and equilibrium cycle conditions. Each case was evaluated by first

determining a critical rod pattern at the defined conditions, assuming no xenon was present. This is representative of startup conditions. Control rod patterns were chosen to ensure that all thermal limits were met. The radial and axial power distributions were then calculated using the GE 3D BWR Simulator. The decay ratio results are summarized in Table 5-4 for the confirmation calculations performed at these conditions. The results are also shown in Figure 5-21. These results demonstrate that the stability criteria of Section 5.1 are met.

After the recirculation pumps have been shifted to high speed, the FCVs are opened until approximately 45-50% of rated core flow is attained. Control rods are then withdrawn until the desired rod pattern is achieved, the maximum rod line is reached, or thermal limits are reached. This condition is also shown in Figure 5-20. The same procedure was used to calculate the power distributions of these conditions as was used in the previous startup cases. The results are summarized in Table 5-4 and Figure 5-21. Again, all cases meet the stability criteria of Section 5.1.

5.4.2 Flow Runback Evaluations

The majority of an operating cycle is spent at rated power conditions and, therefore, this is a likely condition from which a flow reduction event may be initiated and ultimately result in operation near the exclusion region boundary. In general, flow reduction events are caused by recirculation pump trips (single or dual pump trips) or runbacks of the recirculation pump speed (BWR/3-5) or FCVs (BWR/5-6). It is possible that a single recirculation pump trip, flow runback in one loop, or partial runback of the flow in both recirculation loops could result in operation just outside the exclusion region boundary. These conditions near the exclusion region boundary are primarily determined by the initial full power operating conditions and the change in conditions caused by the flow reduction. Since full power xenon is present, the control rod patterns can be significantly different than those experienced during startup.

A set of conditions was simulated from full power, with various cycle exposures and initial core flows for the Perry Cycle 2 and equilibrium cycle

cores described and analyzed in the previous sections. The Cycle 2 cases were evaluated using actual control rod patterns from plant experience. Control rod patterns were also developed for the equilibrium cycle based on standard core management practices, and were selected to ensure that all core operating limits were satisfied at the full power condition. The GE 3D BWR Simulator was used to establish the full power conditions, including equilibrium xenon. The xenon is then assumed to remain constant and the core flow reduced to a point just outside the exclusion region boundary. The power distribution at these final conditions was used as input to the stability analysis. The void coefficient was determined at the same cycle exposure as the power distribution.

The conditions analyzed are shown in Figure 5-20. The calculated decay ratios are summarized in Table 5-4 and in Figure 5-21, where they are compared to the stability criteria of Section 5.1. All of the analyzed conditions meet the stability criteria of Section 5.1.

5.4.3 Loss of Feedwater Heating Evaluations

During startup or shutdown from high power, the reactor will be operated with margin to the exclusion region boundary to reduce the probability of an inadvertent ASF initiation. It is possible during this time that a loss of feedwater heating event could result in an increase in core thermal power to a point just below the region boundary. This will change the power distribution and could potentially result in a less stable condition. To confirm that the stability procedure provides sufficient margin to account for the changes due to feedwater temperature transients, confirmation calculations were performed. The calculations assumed that operation begins near the region boundary and a decrease in feedwater temperature occurs (assumed to be 60°F, which corresponds to a feedwater temperature drop at rated power of approximately 100°F). The reduced feedwater temperature increases the core inlet subcooling resulting in a positive reactivity insertion and an increase in core thermal power.

The GE 3D BWR Simulator was used to calculate the initial conditions just prior to the LOFH event. Control rod patterns were chosen to ensure that all

core operating limits were satisfied prior to the LOFH event. The Simulator was used to calculate the final core thermal power and power distribution based on the lower feedwater temperature. These final power distributions and the final power level were used in the stability analysis to determine the decay ratios. The void coefficient was determined at the same cycle exposure.

The LOFH events are assumed to begin from both startup conditions (xenon free) and conditions which result from a flow reduction from rated power. The stability analysis points (final conditions) are shown in Figure 5-20. The decay ratios are summarized in Table 5-4 and are compared in Figure 5-21 to the stability criteria of Section 5.1. All cases meet the stability criteria of Section 5.1.

5.4.4 Summary

The Region Boundary Definition Procedure described in Section 5.2 was developed to provide a conservative estimate of the region of the power/flow map that has the potential for thermal-hydraulic oscillations. This was accomplished by choosing a combination of best estimate and conservative inputs that, in many cases, were mutually exclusive. Comparisons to previous stability tests and events demonstrated that the procedure consistently predicted larger decay ratios when compared to predictions which used inputs based on actual plant conditions. To further confirm that the procedure provides a conservative estimate of reactor stability, calculations were performed for a wide variety of conditions that were representative of actual plant operation. In addition, conditions resulting from unplanned core flow reductions and feedwater temperature reductions were evaluated (Sections 5.4.1, 5.4.2, and 5.4.3). For all cases analyzed, the core and channel decay ratios were less than or equal to the stability criteria defined in Section 5.1. When combined with the previous calculations of instability events and tests, these calculations demonstrate that the Region Boundary Definition Procedure provides adequate conservatism for the expected range of plant operating conditions.

5.5 PLANT- AND CYCLE-SPECIFIC APPLICATION OF GENERIC REGION BOUNDARIES

The application of the Region Boundary Definition Procedure to a specific BWR/6 plant, Perry, is summarized in Section 5.3. Although this procedure could be individually applied to any plant, this analysis is intended to define a generic region boundary that similar plants can use. To confirm that the region boundary is applicable to another plant, a comparison of the major parameters (including core and fuel design) affecting stability must be made. If all parameters are within a defined range, the region boundary can be applied to the specific plant. If any parameter is outside the defined range, the region boundary can be modified to account for the variability of stability characteristics for the particular parameter.

This approach of defining a generic region can be applied to each plant group where a plant group is defined as a set of plants which are similar with respect to parameters important to thermal-hydraulic instability. Sensitivity studies can then be performed for a plant group, to define the range of acceptable parameters and determine any necessary modifications to the generic region boundary. The parameters that have been identified as key stability parameters which must be evaluated to determine the applicability of a generic region boundary are discussed in this section. The parameters are separated into those that are functions of fuel and core design, and those that are functions of plant design and operational strategies.

5.5.1 Fuel and Core Design Parameters

The effect of fuel and core designs on stability is described by the core and channel decay ratios. Additionally, the parameters which directly affect the decay ratios (e.g., channel pressure drop characteristics, fuel thermal time constant, moderator void reactivity coefficient, etc.) can be independently evaluated for their impact on stability. The following sections describe the procedures that may be used to confirm the acceptability of a design.

(1) Channel Decay Ratio

Use of the channel decay ratio as a parameter allows the consideration of many separate parameters to be combined, thereby simplifying the evaluation process. In addition, the channel decay ratio is a direct variable in the stability criteria of Section 5.1, and any change can be directly related to the criteria. The generic region boundaries are developed with a base fuel type with known channel decay ratio that is representative of current fuel designs. The characteristics of the base fuel type can also be used to define a set of fuel types whose stability characteristics are bounded by the base design. The generic region boundary will be applicable to fuel designs whose stability characteristics have been demonstrated to be bounded by the base design. Any future fuel design can be added to the range of acceptable designs by performing specific channel decay ratio calculations which demonstrate comparable stability performance.

For fuel designs that are not bounded by the base design or that do not have a channel decay ratio calculated for comparison to the base design, other alternatives are available to assess the relative stability characteristics. Sensitivity studies can be performed which vary the important fuel design parameters that affect channel decay ratio (e.g., two-phase to single-phase pressure drop ratio, number of fuel rods, fuel rod diameter, etc.). A response surface can be constructed from these results that describes the change in channel decay ratio as a function of the independent variables. The specific parameters for a design can be input into the response surface to determine the relative change in the channel decay ratio from the base design. This method is useful for designs with similar basic characteristics and may require updating to include sufficient variation in parameters to accommodate new fuel designs.

The response surface can be generated for several points along the exclusion region boundary (i.e., at the intersection of the maximum core flow and maximum rod line, and at natural circulation) and sensitivity studies can be performed to determine the necessary change in the location of the exclusion region boundary for a given change in channel decay ratio

(e.g., for an increase in channel decay ratio of 0.10, the exclusion region boundary must be increased by 5% of rated core flow at the maximum rod line, and reduced by 5% rod line at natural circulation flow). These sensitivity studies can also be used to calculate the required change in the exclusion region boundary when direct calculations result in an increase in the channel decay ratio for a fuel design.

(2) Core Decay Ratio

The core decay ratio is affected by the fuel thermal time constant since changes in neutron flux affect the local void content through the fuel rod surface heat flux. The fuel thermal time constant is dependent primarily on the fuel rod diameter and the gap conductance. The timing and magnitude of this feedback are dependent on the fuel thermal time constant. The resulting magnitude of the void feedback is directly proportional to the moderator density reactivity coefficient which also has a direct effect on the core decay ratio. The base fuel and core design used to develop the generic exclusion region boundaries will define the base fuel thermal time constant and moderator density reactivity coefficient. Since these effect the core decay ratio, calculations of the core decay ratio for fuel designs other than the base design can be made and compared to the base design to determine their acceptability. Alternatively, as described above, a response surface can be defined that describes the change in core decay ratio as a function of the change in fuel thermal time constant and moderator density reactivity coefficient. Changes from the base fuel thermal time constant and moderator density reactivity coefficient can be translated into a change in core decay ratio, and subsequently the exclusion region boundary can be modified if necessary.

5.5.2 Plant Design and Operating Strategy Parameters

The operation of two different plants with the same fuel types can result in a range of core and channel decay ratios, depending on certain characteristics of the plant (e.g., inlet orificing, recirculation loop resistance, feedwater temperature) and operating strategies (e.g., high radial peaking factors to take advantage of low operating limits).

Therefore, in addition to comparisons of fuel designs, plant and operating strategy comparisons must be made to ensure the applicability of the generic region boundaries. The following parameters have been identified as important to stability and proposed methods for comparison to values used in the generic analysis are discussed.

(1) Power/Flow Ratio

In general, stability results are presented at power and flow states as a percent of the rated conditions. Two plants can be operating with the same power/flow condition as a percent of rated but the individual fuel bundles could be at different absolute values of power/flow ratio.

For operation at the same bundle flow, but higher absolute power, the channel decay ratio will be higher. Therefore, one of the sensitivities between plants within a plant group will be the absolute power/flow ratio. In general, within a plant group, the power/flow ratio will not vary significantly. Sensitivity studies will be performed to determine the change in core and channel decay ratios as a function of the absolute power/flow ratio. The generic region boundary will be defined with a base value that, in general, will bound the expected values for other plants within the group.

(2) Recirculation Loop Resistance

The recirculation loop in the FABLE/BYPSS methodology is modeled by a simple gain and time constant. These parameters are calculated for specific power and flow conditions based on the pressure drop characteristics of the recirculation loop. These characteristics may vary among plants because of differences in jet pump design, steam separator pressure drop characteristics, or external recirculation loop hydraulic differences. In general, these parameters are not expected to vary significantly among plants within a plant group. Sensitivity studies will be performed to determine the change in core and channel decay ratios as a function of these parameters. The generic region boundary will be defined

with a base value that, in general, will bound the expected values for other plants within the group.

(3) Fuel Inlet Orifice Diameter

The largest component of the single-phase pressure drop is from the fuel inlet orifice. A wide variety of inlet orifice diameters exists in plants today. Because of the significant impact of the fuel inlet orifice diameter on channel decay ratios, plant groups will be defined with similar inlet orifice diameters. Within a plant group, the generic region boundary will be defined with a base value that will bound the expected values for the other plants within the group.

(4) Inlet Subcooling

Inlet subcooling affects both the core and channel decay ratio and will vary for plants within a plant group. The inlet subcooling is primarily dependent on the feedwater temperature and, in general, plants fall within one of two categories; those with 420°F rated feedwater temperatures and those with 360-385°F rated feedwater temperatures. The generic exclusion region boundary will be determined for a range of feedwater temperatures expected for the plant group.

In addition to normal feedwater temperatures, plants may also operate with feedwater heaters out-of-service (FWHOS) or with final feedwater temperature reduction (FFWTR). In general, FFWTR is performed at the end of cycle when the plant is operating at or near the rated rod line. Generic analyses have demonstrated that the core and channel decay ratios are not significantly different during these conditions and, therefore, FFWTR does not impact the exclusion region boundary when performed at the end of cycle at or above rated core flow. For FWHOS operation, the results are expected to be similar to those determined from confirmation calculations discussed in Section 5.4.3 for the LOFH event. Therefore, the Region Boundary Definition Procedure adequately covers these modes of operation.

(5) Radial Power Distribution

The radial power distribution directly affects both the core and channel decay ratios. The maximum radial peaking factor determines the hot channel power and the distribution of radial power affects the core decay ratio. Direct sensitivity studies can be performed to evaluate the sensitivity of channel decay ratio to maximum radial peaking factor. The generic region boundary will be generated for a given radial peaking factor that reasonably bounds expected plant operation for a plant group. The impact of any deviations from this base value will be estimated from the results of the sensitivity studies.

The radial power distribution has a more complicated effect on the core decay ratio, primarily through its impact on the total moderator density reactivity feedback. The Region Boundary Definition Procedure assumes the radial power distribution from EOC Haling conditions. Confirmation studies in Section 5.4 demonstrated that the overall conservatism of the procedure and stability criteria of Section 5.1 are sufficient to bound the expected variations in radial power distribution. Therefore, only the impact of the maximum radial peaking factor will be evaluated.

Table 5-1
BWR/6 PLANT GROUPING

	Clinton	Grand Gulf	Perry	River Bend
Power Density (kW/l)	52.4	54.2	54.1	52.4
Power/Flow Ratio (MWth/{Mlbm/hr})	34.2	34.1	34.4	34.2
Rated Feedwater Temperature (°F)	420	420	420	420
Maximum Rod Line (%)	120	120	120	105
Fuel Inlet Orifice Diameter (in)	2.43	2.43	2.43	2.43
Operating Cycle	3	5	3	4
Fuel Type	GE8x8	ANF9x9-5 ANF8x8	GE8x8	GE8x8

Table 5-2

PERRY CYCLE 2 REGION BOUNDARY CALCULATIONS

Point*	Power/Flow (%/%)	Hot Channel Decay Ratio	Core Decay Ratio
HIGH ROD LINE			
1	75.4/50.0	0.52	0.56
2	71.0/45.0	0.62	0.71
MEDIUM ROD LINE			
3	61.7/50.0	0.36	0.48
4	54.7/40.0	0.53	0.72
4A	53.2/38.0	0.58	0.78
4B	51.7/36.0	0.64	0.86
NATURAL CIRCULATION			
5	36.7/30.0	0.49	0.77
6	39.0/30.0	0.55	1.07

* See Figure 5-14 for definition of points.

Table 5-3

PERRY EQUILIBRIUM CYCLE REGION BOUNDARY CALCULATIONS

Point*	Power/Flow (%/%)	Hot Channel Decay Ratio	Core Decay Ratio
HIGH ROD LINE			
7	75.4/50.0	0.49	0.74
8	72.9/47.0	0.55	0.82
9	71.0/45.0	0.59	0.89
MEDIUM ROD LINE			
10	58.3/45.0	0.42	0.76
11	54.7/40.0	0.50	0.89
NATURAL CIRCULATION			
12	31.0/30.0	0.39	0.66
13	35.0/30.0	0.44	0.81
14	36.7/30.0	0.47	0.88

* See Figure 5-17 for definition of points.

Table 5-4

PERRY CONFIRMATION ANALYSIS DECAY RATIOS

Point *	Exposure/Cycle	Final Power/Flow	Hot Channel Decay Ratio	Core Decay Ratio
	PUMP UPSHIFT CONDITIONS			
A1	BOC2	37.0/35.0	0.44	0.22
A2	MOC2 **	39.3/34.0	0.51	0.52
A3	BOEC	38.5/34.0	0.51	0.47
A4	MOEC	45.5/38.0	0.39	0.66
A5	EOEC	41.7/36.0	0.21	0.71
	MAXIMUM POWER DURING STARTUP			
B1	BOC2	74.5/50.0	0.35	0.25
B2	EOC2	74.8/50.0	0.38	0.55
B3	BOEC	75.3/50.0	0.32	0.37
B4	MOEC	75.1/50.0	0.39	0.59
B5	EOEC	62.5/45.0	0.35	0.75
	FLOW REDUCTION EVENTS			
C1	BOC2 ***	63.1/45.0	0.32	0.51
C2	MOC2 ***	68.0/46.0	0.39	0.48
C3	EOC2 ***	68.0/46.0	0.40	0.61
C4	BOEC	60.0/42.0	0.34	0.60
C5	MOEC	72.0/48.0	0.31	0.66
C6	MOEC	64.9/45.5	0.35	0.70
C7	EOEC	59.0/43.5	0.32	0.74
	LOSS OF FEEDWATER HEATER EVENTS			
D1	BOC2	37.3/35.0	0.46	0.26
D2	MOC2	39.3/34.0	0.44	0.47
D3	EOC2	68.5/46.0	0.44	0.62
D4	BOEC	74.8/50.0	0.33	0.36
D5	MOEC	72.4/48.0	0.35	0.65
D6	MOEC	45.5/38.0	0.44	0.71
D7	EOEC	59.5/43.5	0.30	0.73

* See Figure 5-20 for definition of points.
** Actual Cycle 2 conditions.
*** Based on Cycle 2 conditions at full power.

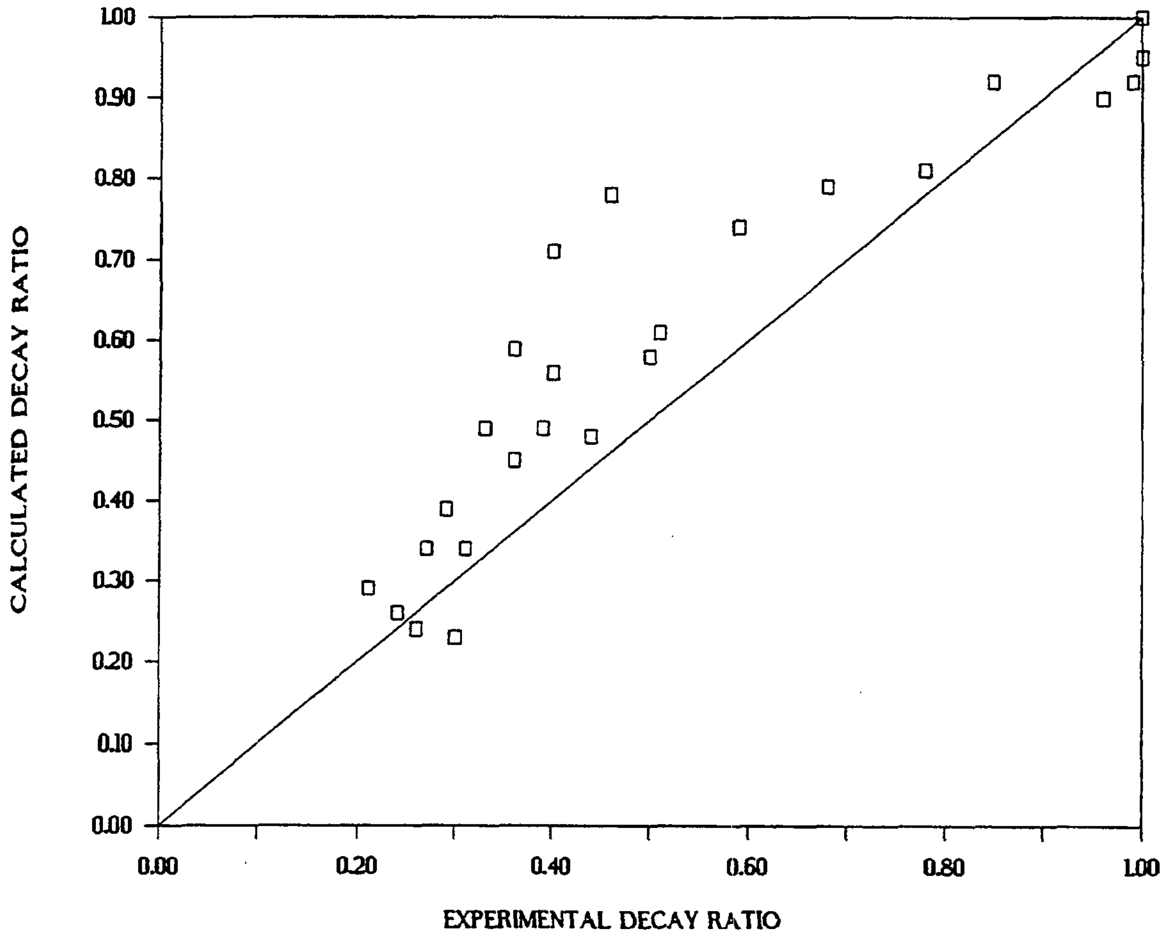


FIGURE 5-1. FABLE/BYPSS COMPARISON TO TEST DATA

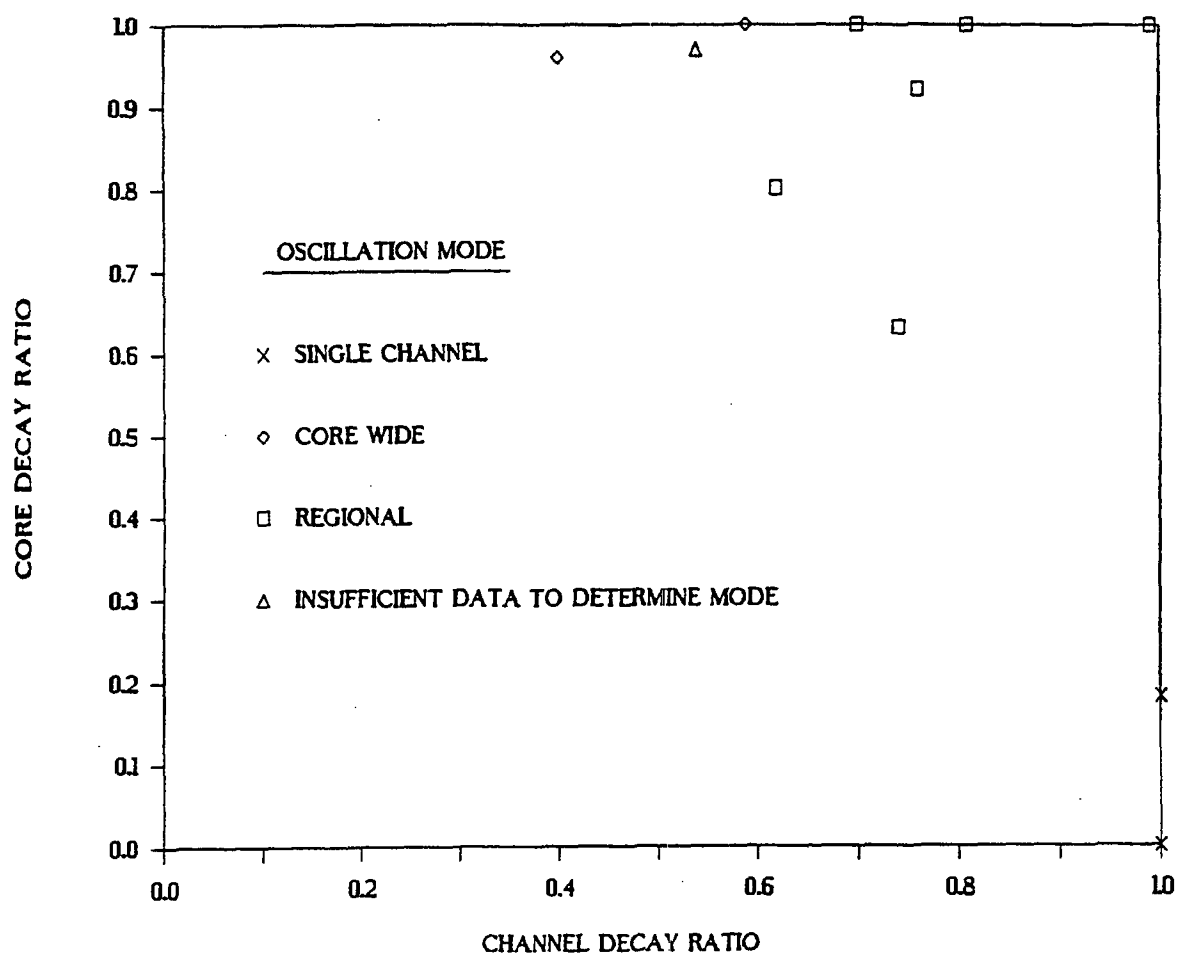


FIGURE 5-2. FABLE/BYPSS COMPARISON TO OSCILLATION MODES

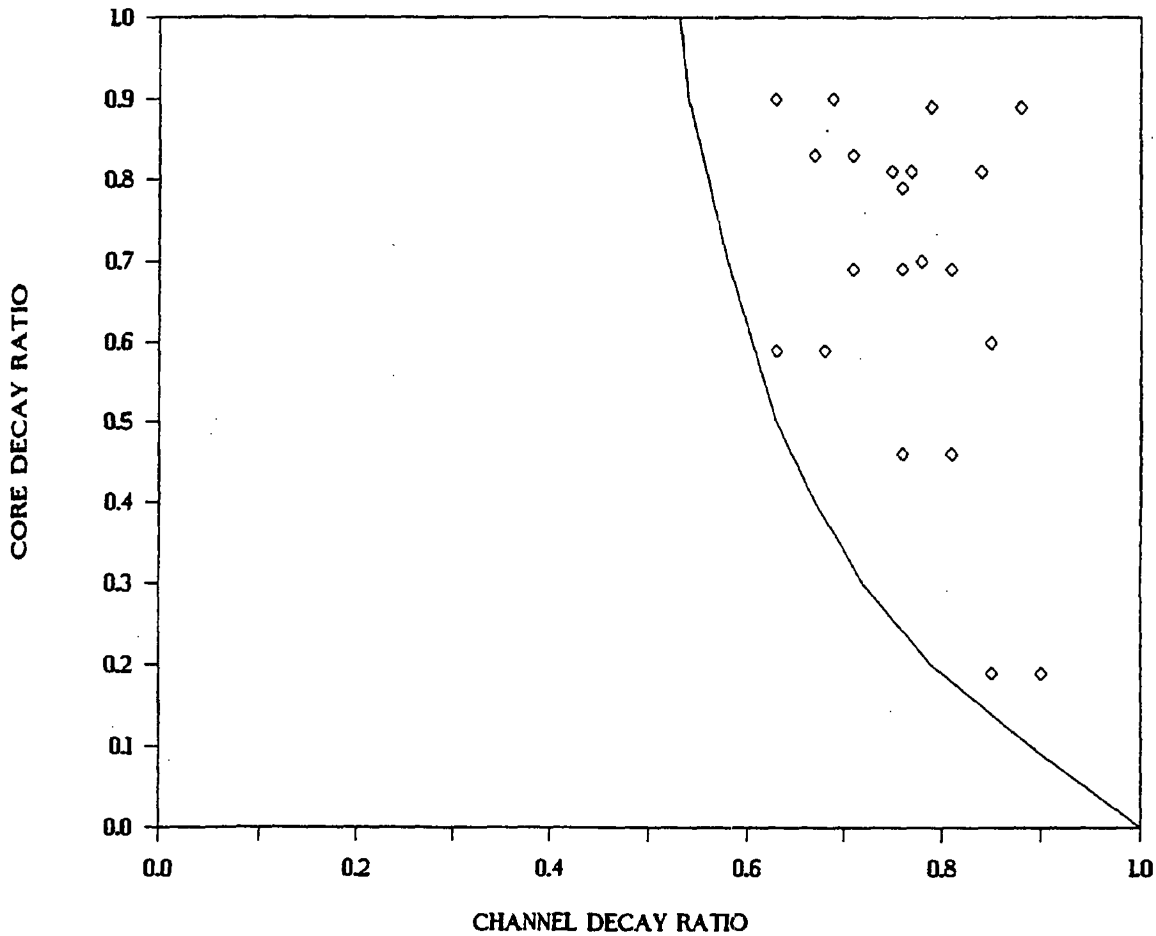


FIGURE 5-3. CORE/CHANNEL DECAY RATIOS RESULTING IN CALCULATED REGIONAL OSCILLATIONS

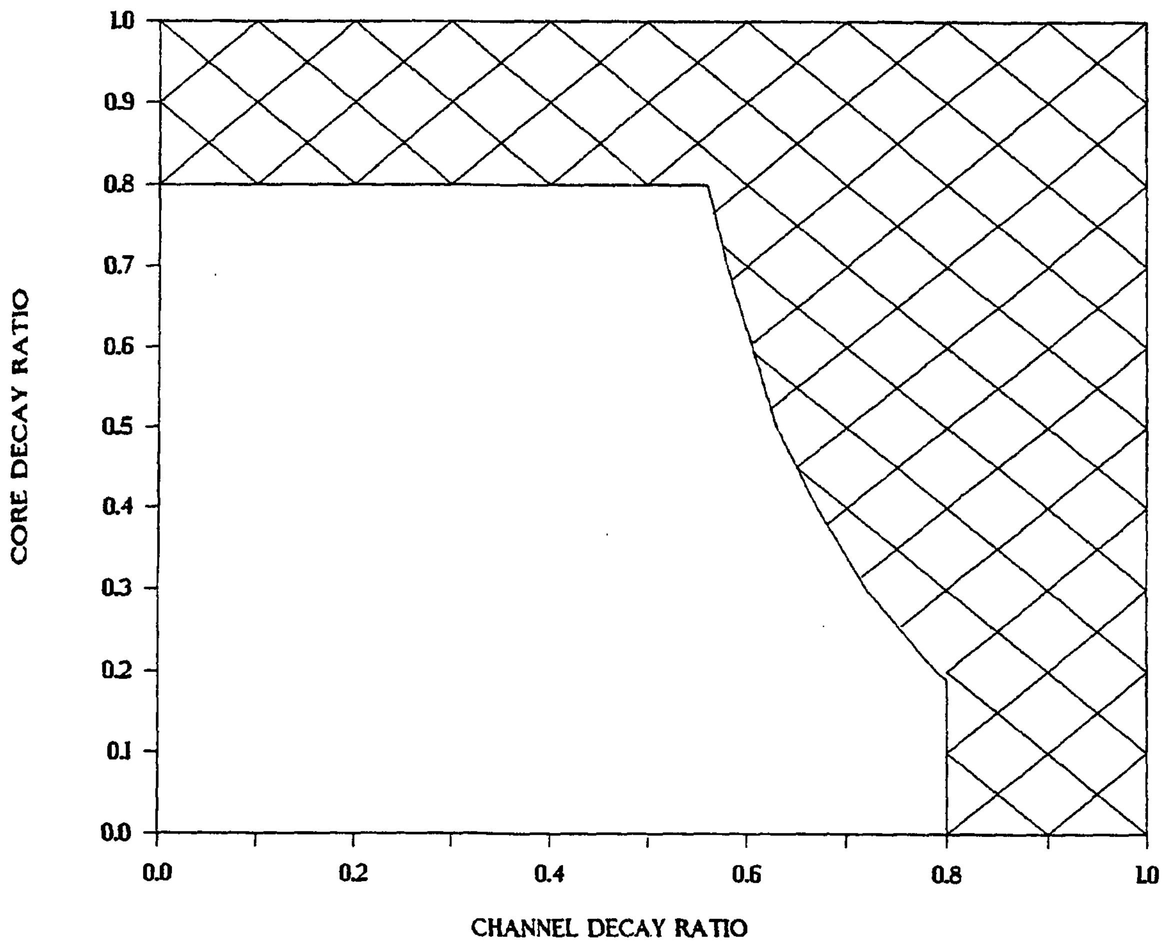


FIGURE 5-4. FABLE/BYPSS STABILITY CRITERIA

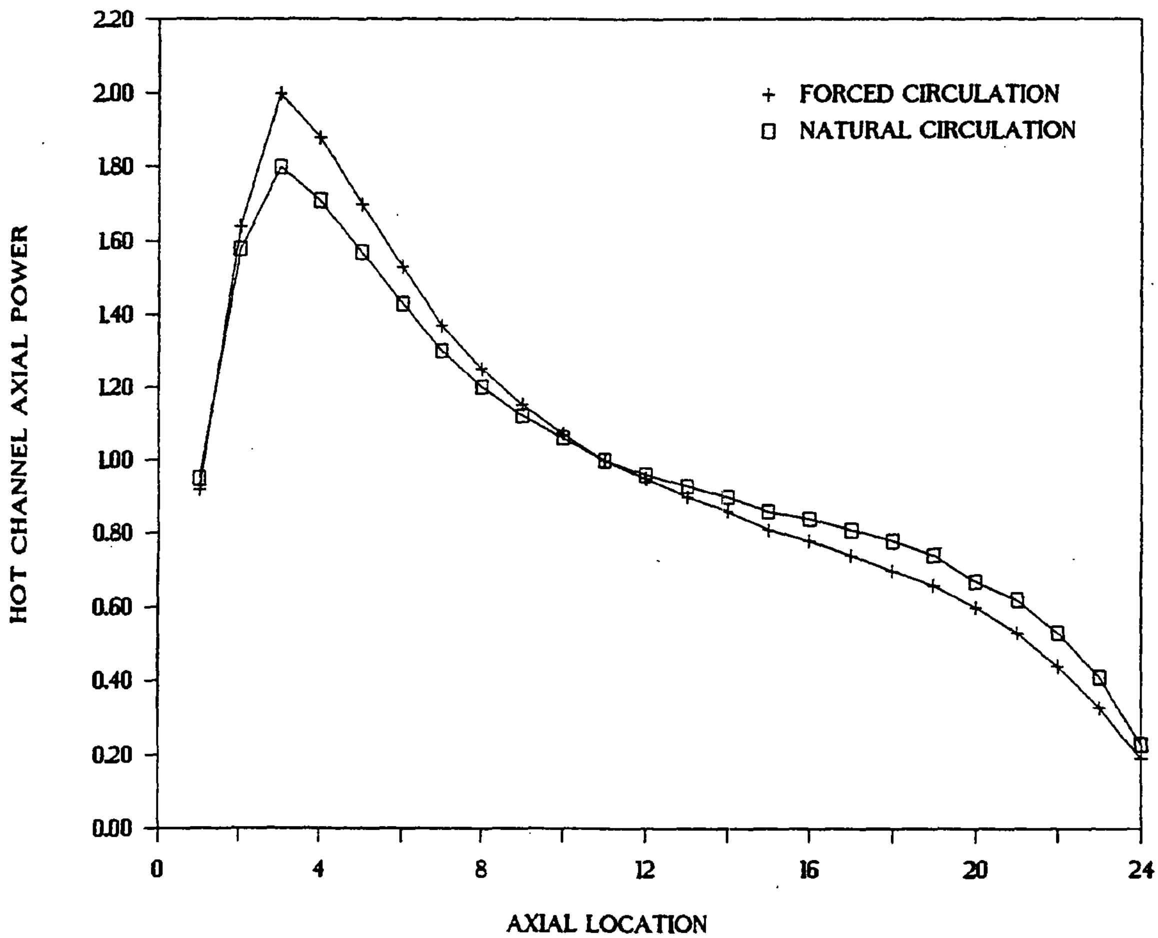


FIGURE 5-5. HOT CHANNEL AXIAL POWER DISTRIBUTION

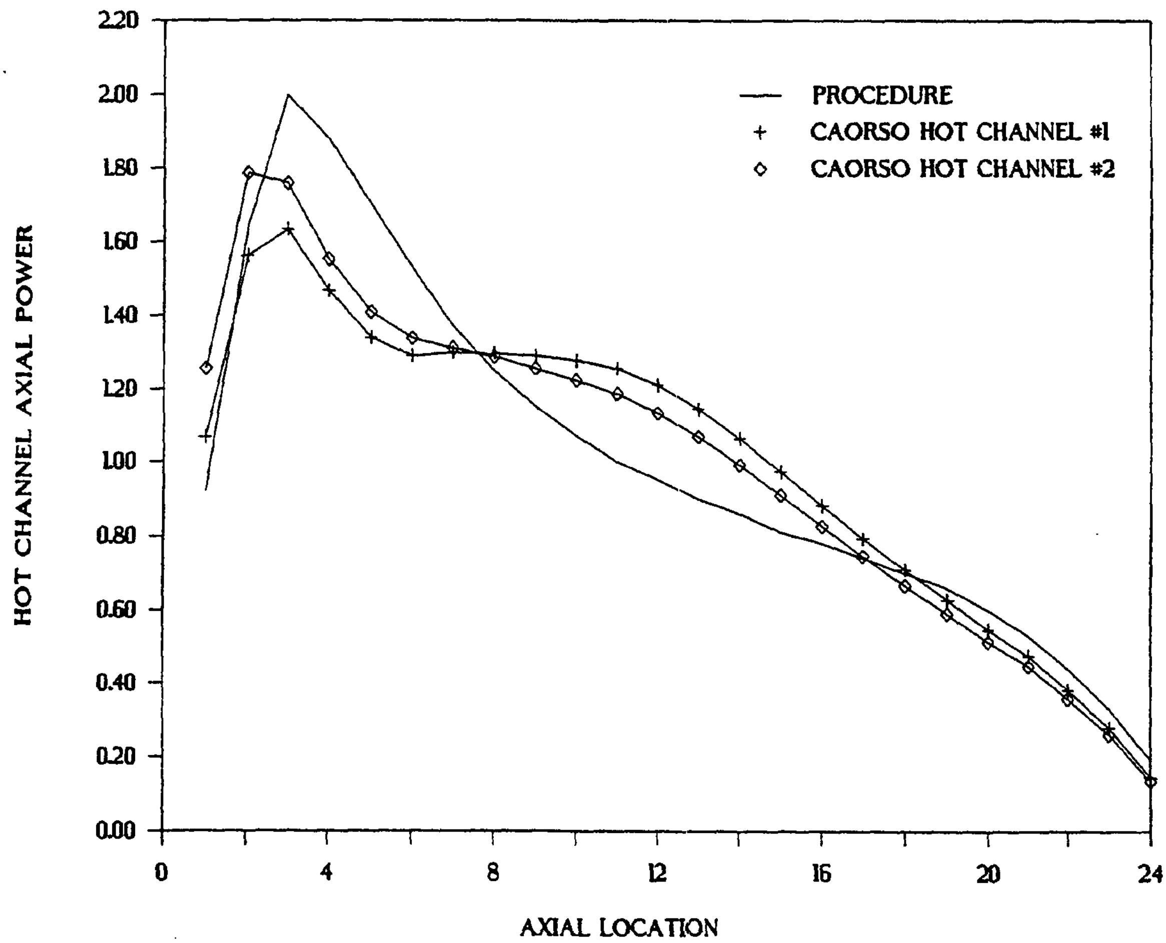


FIGURE 5-6. HOT CHANNEL AXIAL POWER DISTRIBUTION - COMPARISON TO CAORSO EVENT DATA

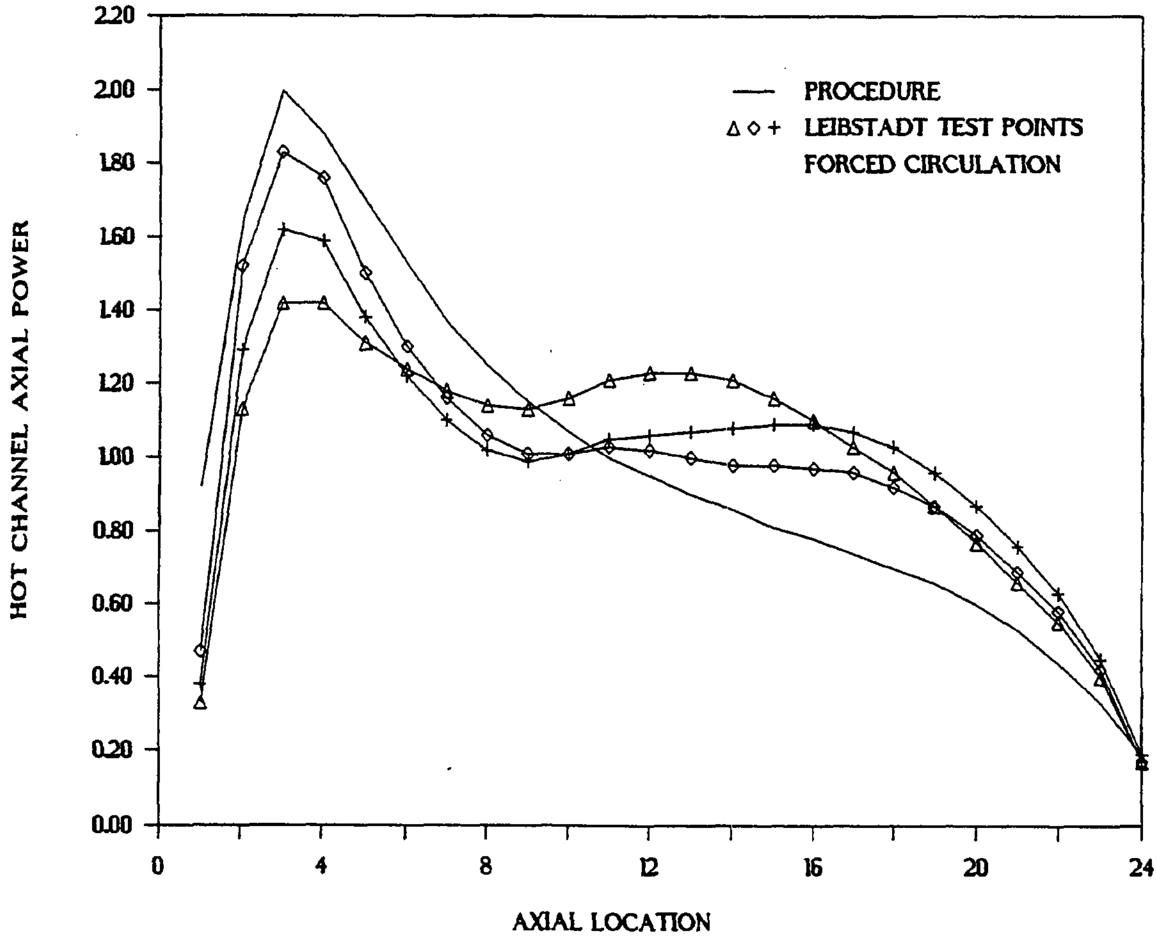


FIGURE 5-7. HOT CHANNEL AXIAL POWER DISTRIBUTION - COMPARISON TO LEIBSTADT TEST DATA

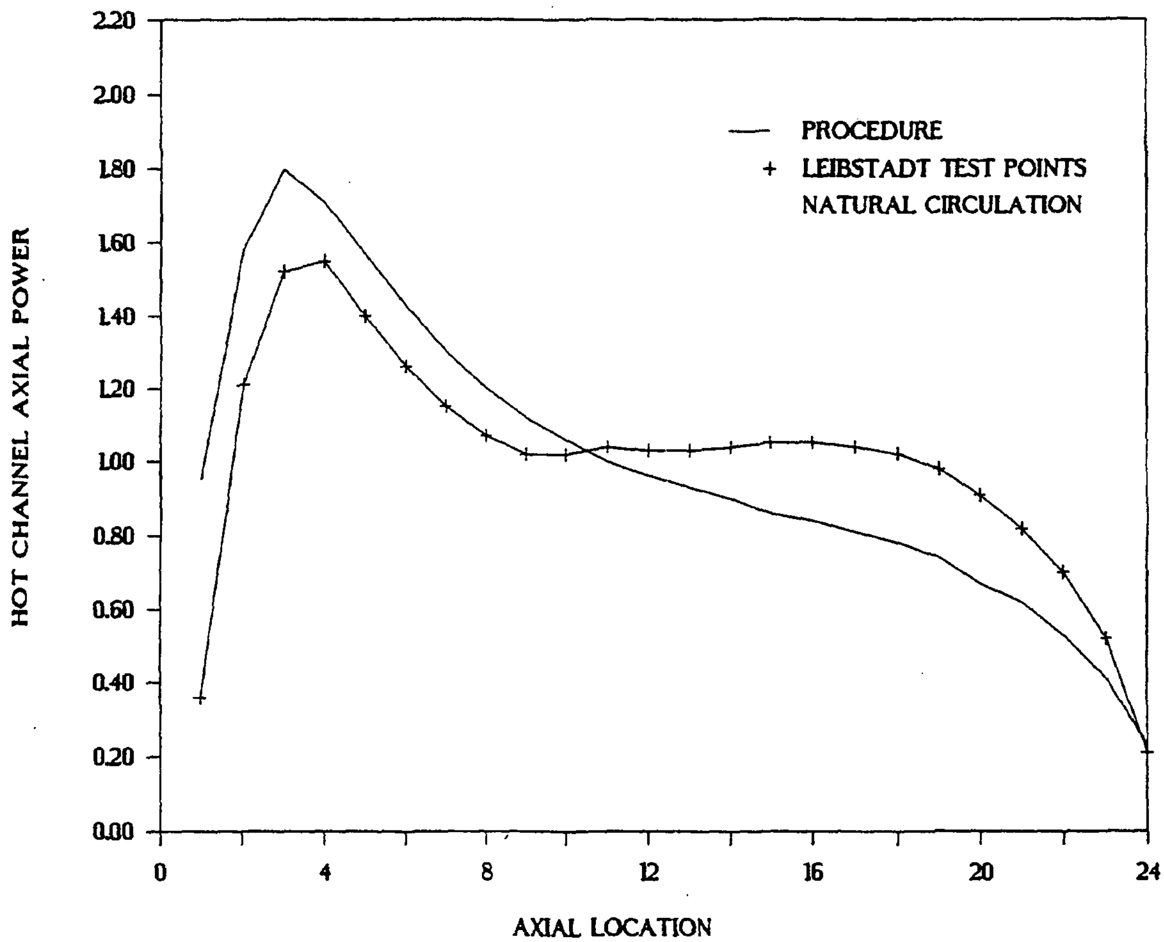


FIGURE 5-8. HOT CHANNEL AXIAL POWER DISTRIBUTION - COMPARISON TO LEIBSTADT TEST DATA

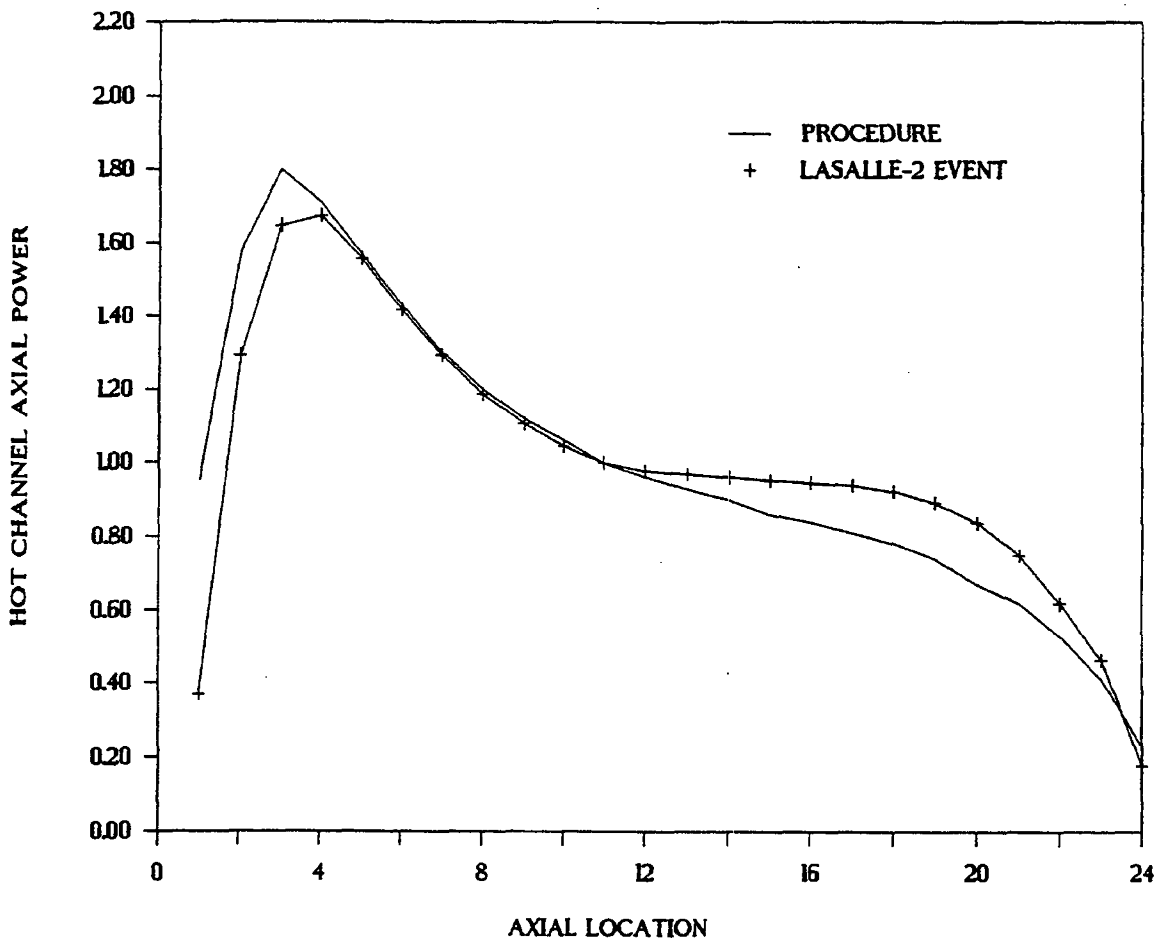


FIGURE 5-9. HOT CHANNEL AXIAL POWER DISTRIBUTION - COMPARISON TO LASALLE EVENT DATA

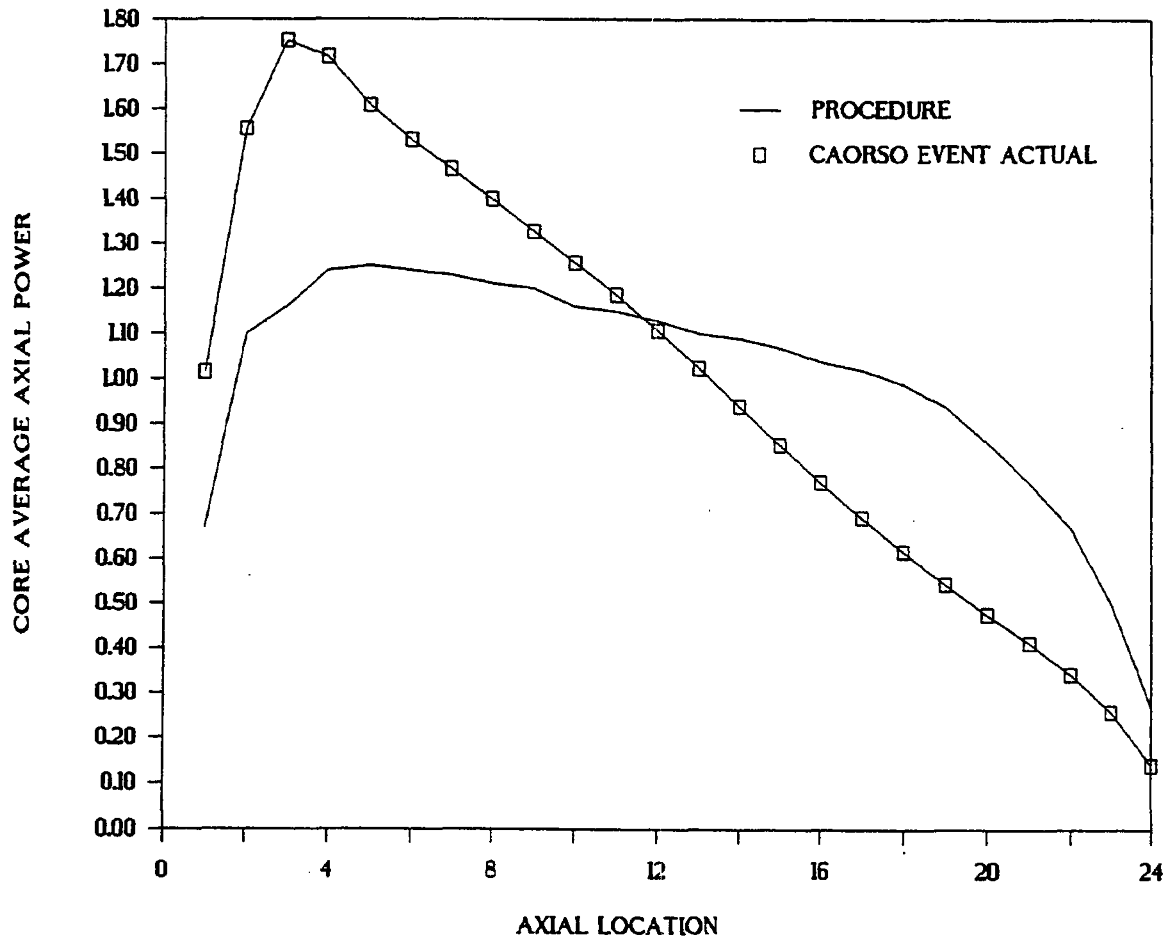


FIGURE 5-10. CORE AVERAGE AXIAL POWER DISTRIBUTION - COMPARISON TO CAORSO EVENT DATA

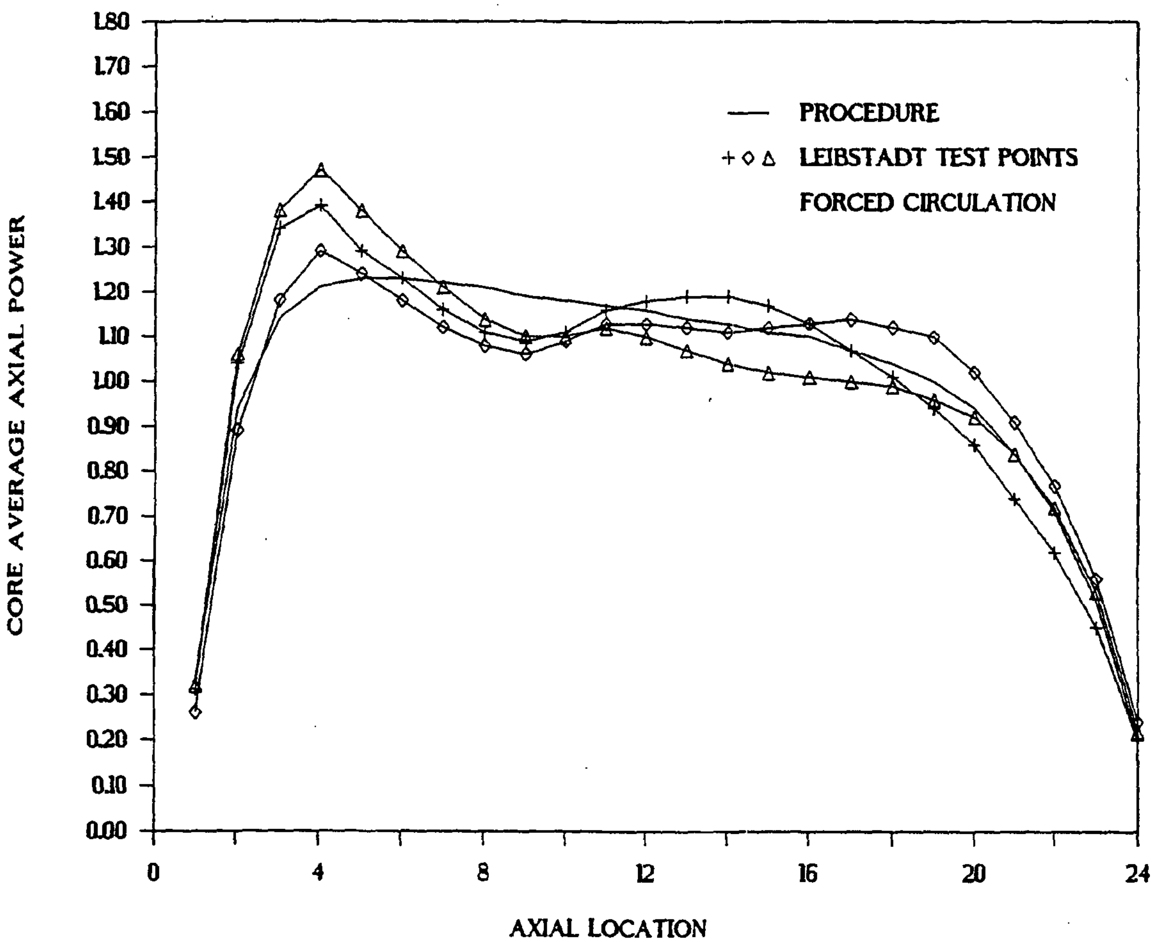


FIGURE 5-11. CORE AVERAGE AXIAL POWER DISTRIBUTION - COMPARISON TO LEIBSTADT TEST DATA

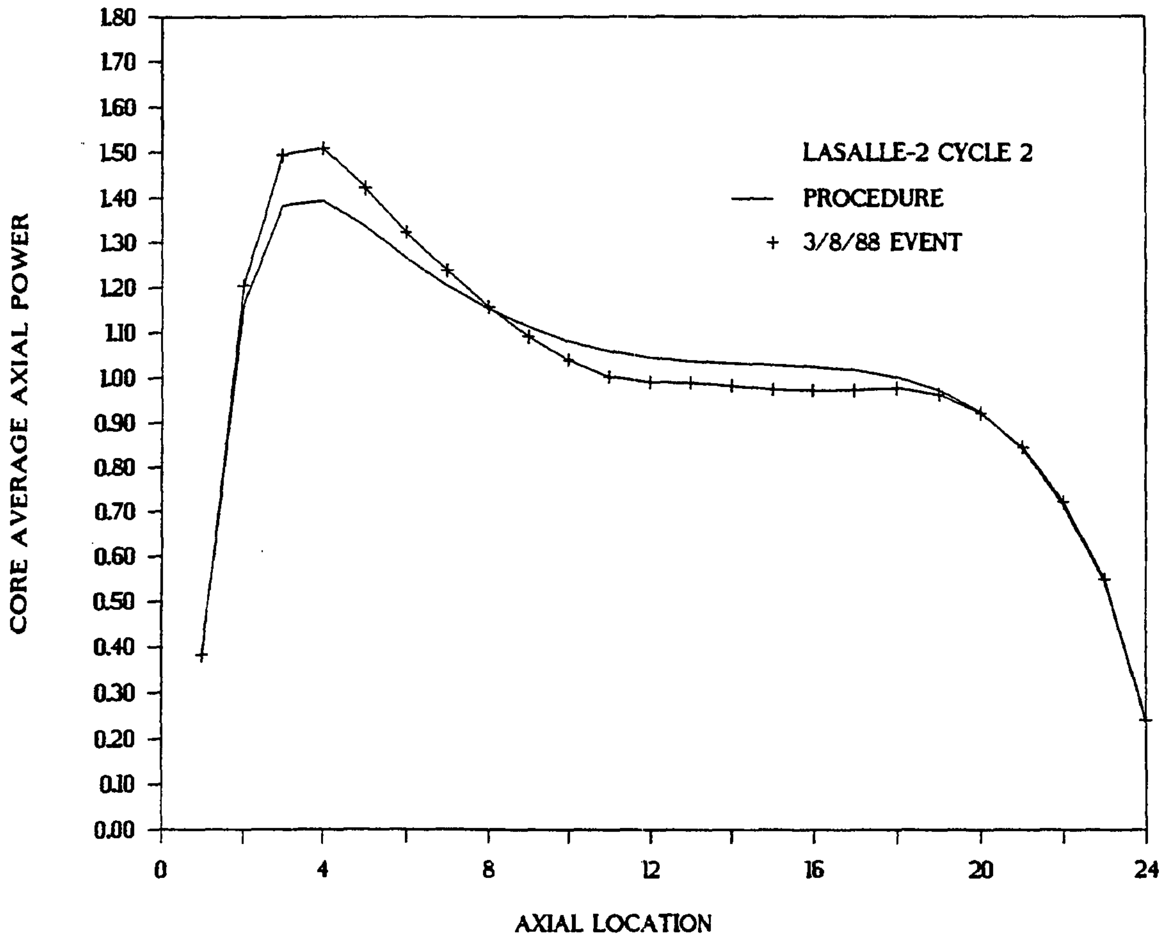


FIGURE 5-12. CORE AVERAGE AXIAL POWER DISTRIBUTION - COMPARISON TO LASALLE EVENT DATA

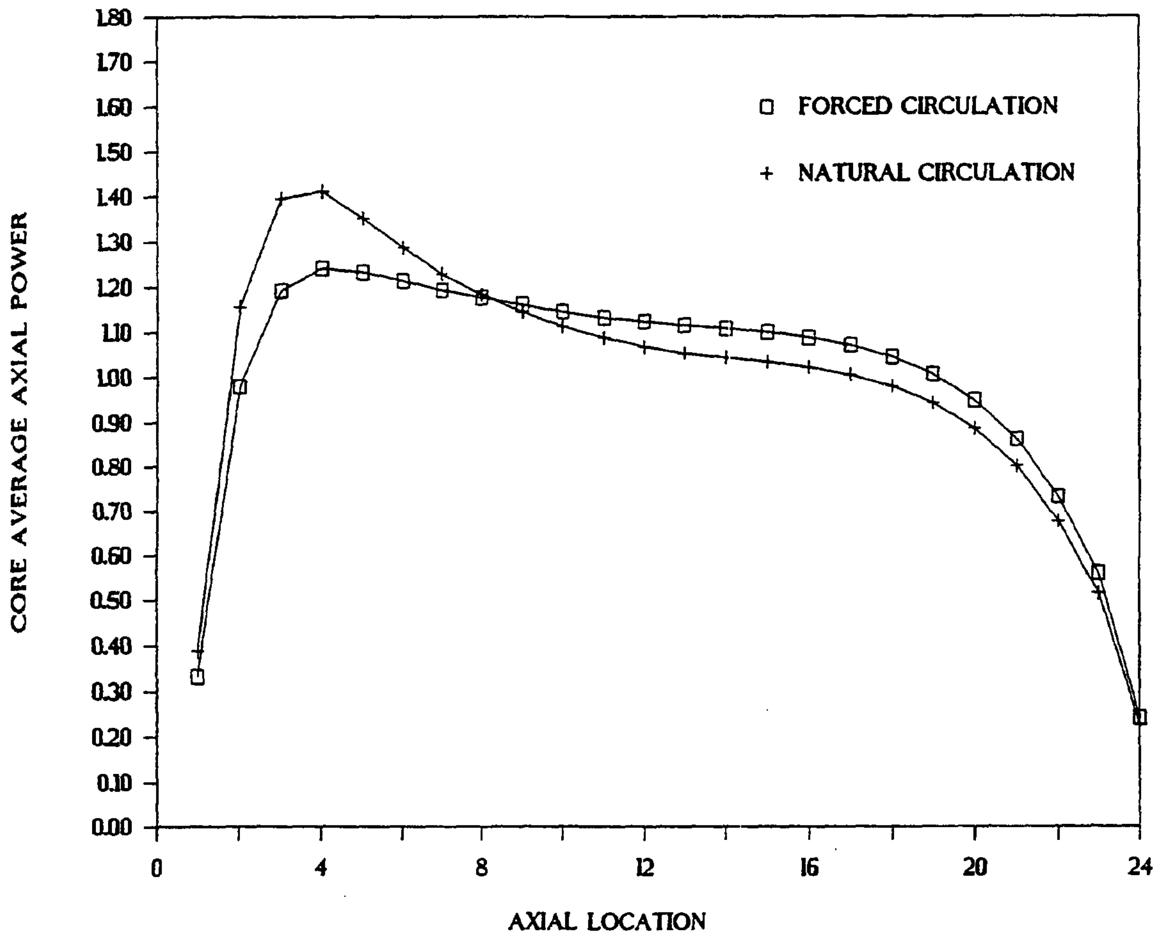


FIGURE 5-13. PERRY CYCLE 2 EOC HALING AXIAL POWER DISTRIBUTIONS

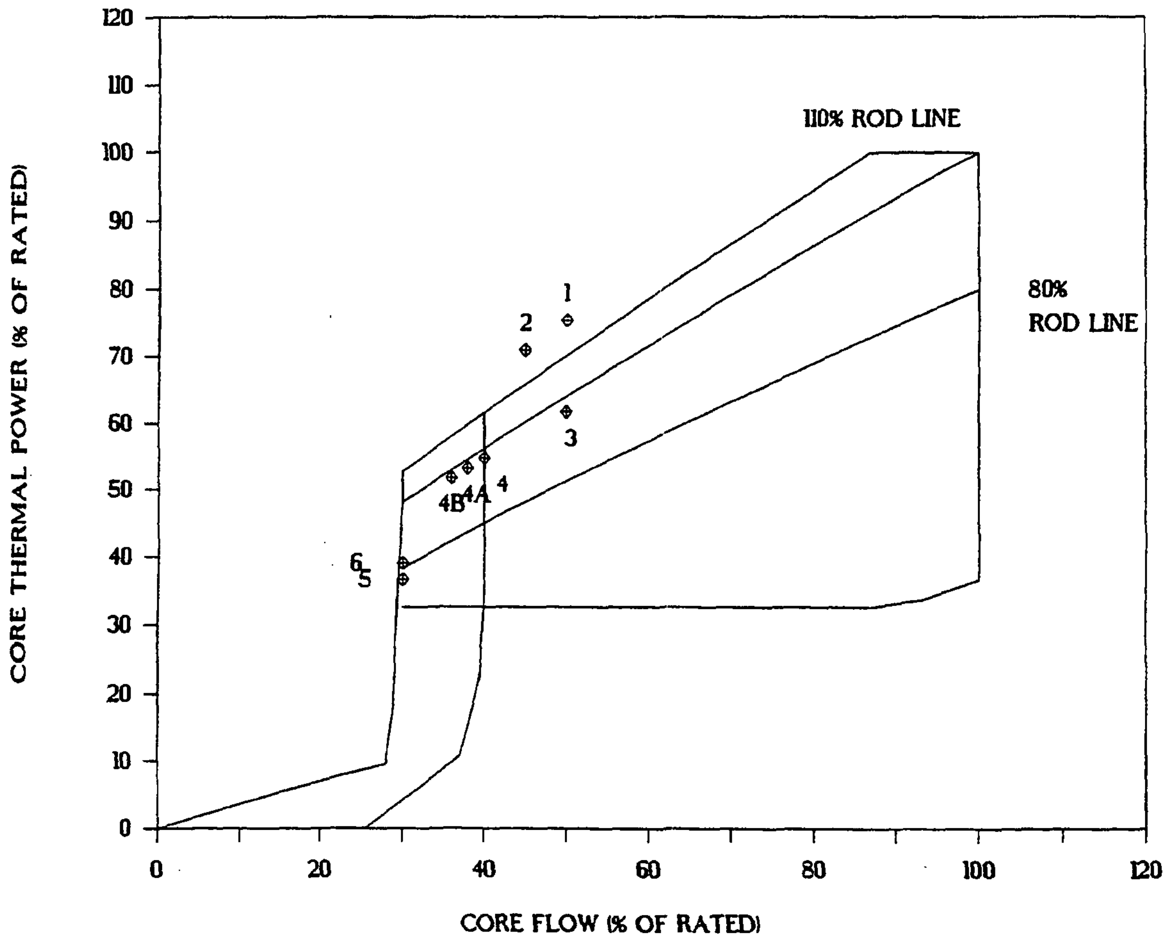


FIGURE 5-14. PERRY CYCLE 2 REGION BOUNDARY DEFINITION ANALYSIS POINTS

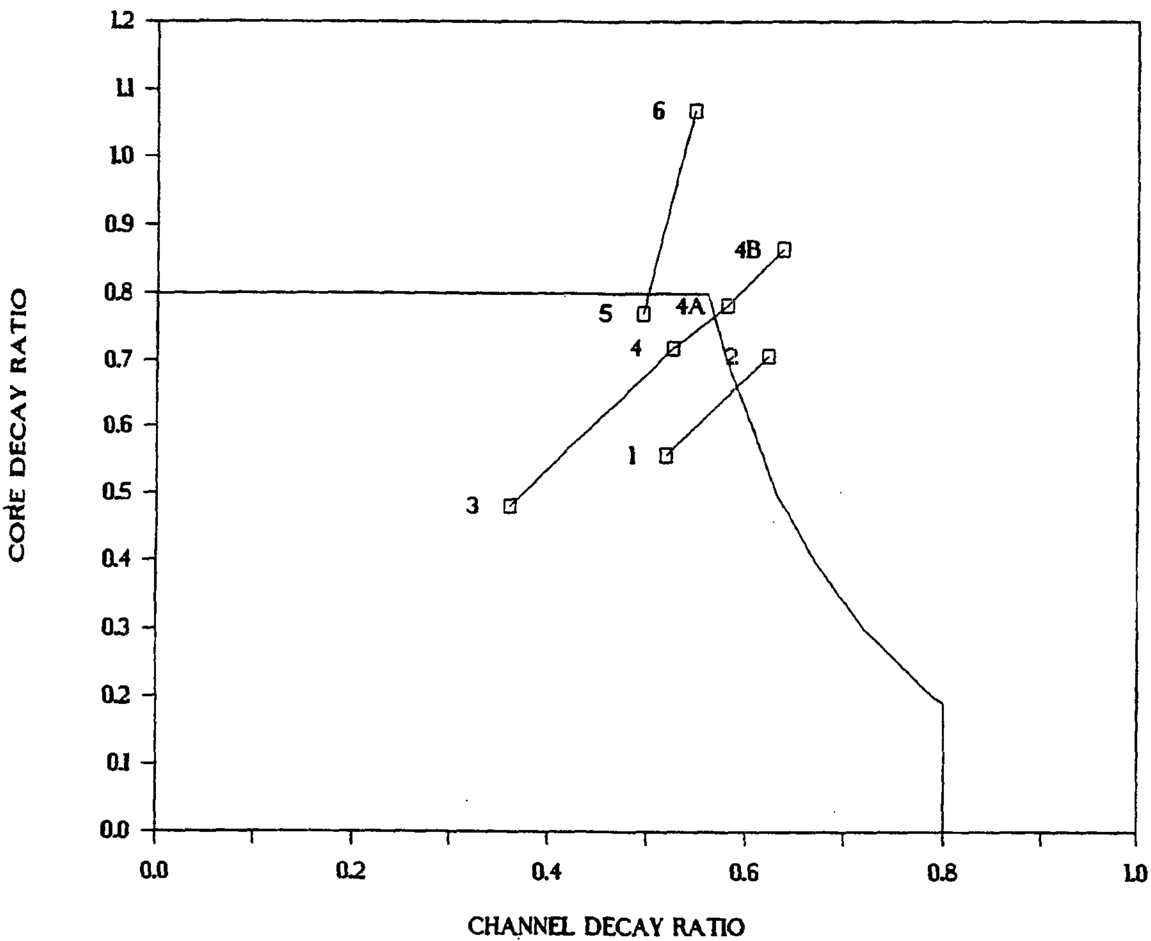


FIGURE 5-15. PERRY CYCLE 2 DECAY RATIOS

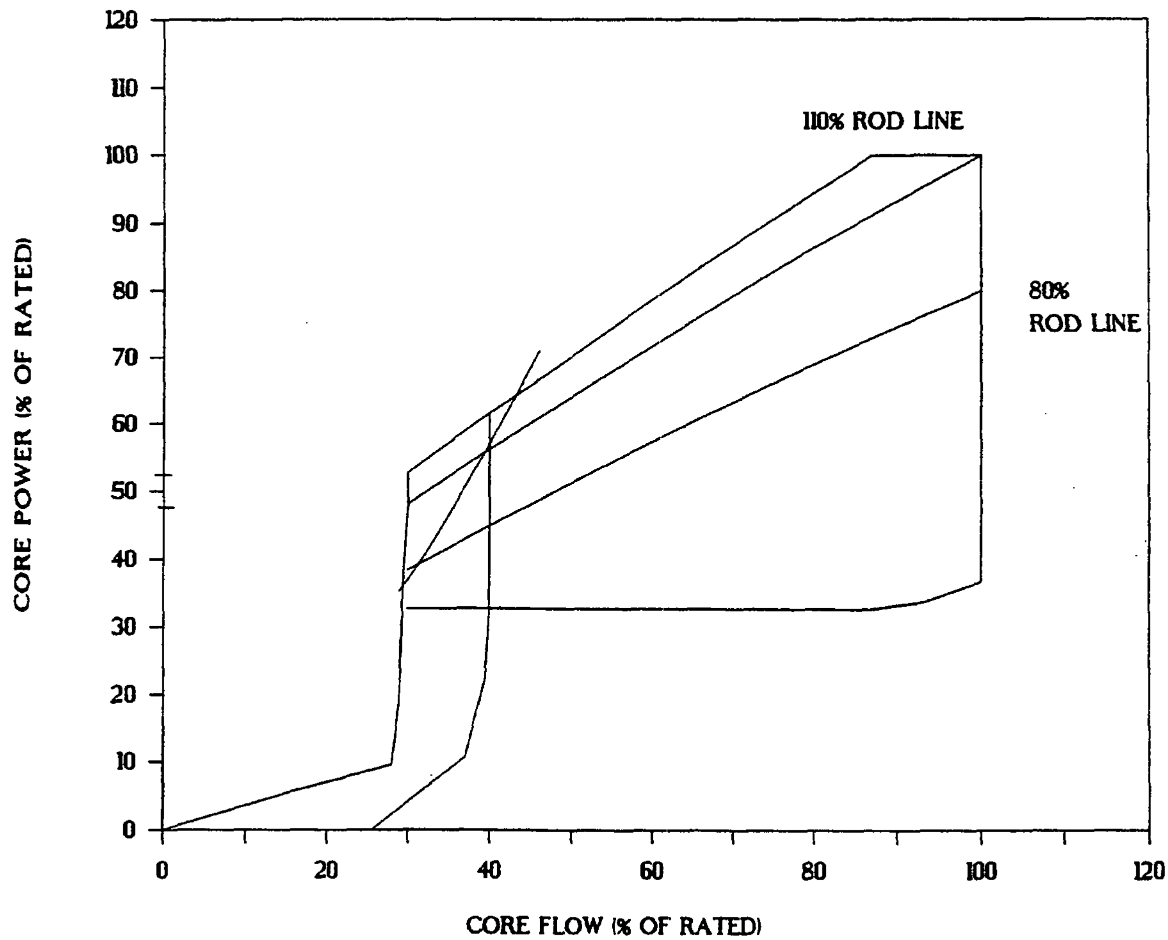


FIGURE 5-16. PERRY CYCLE 2 REGION BOUNDARY DEFINITION

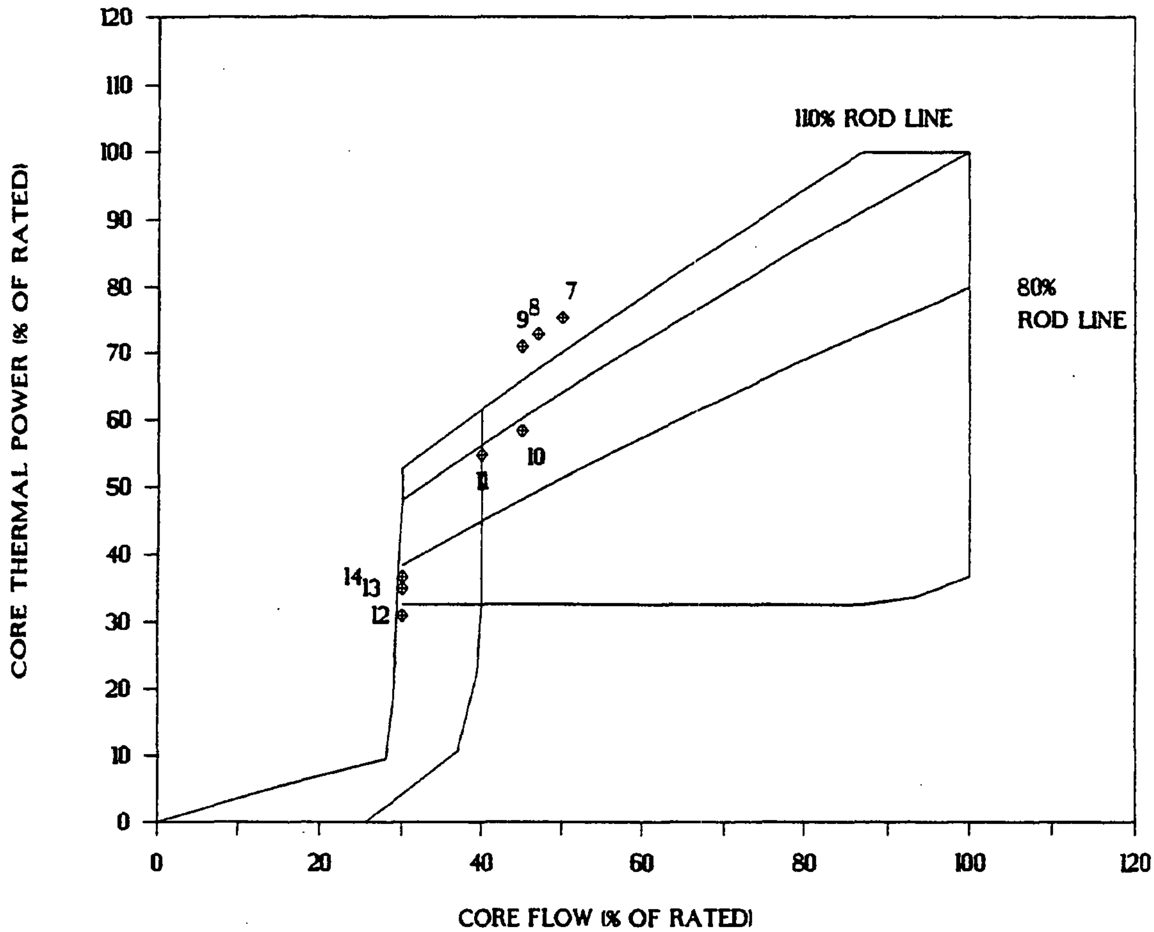


FIGURE 5-17. PERRY EQUILIBRIUM CYCLE REGION BOUNDARY DEFINITION ANALYSIS POINTS

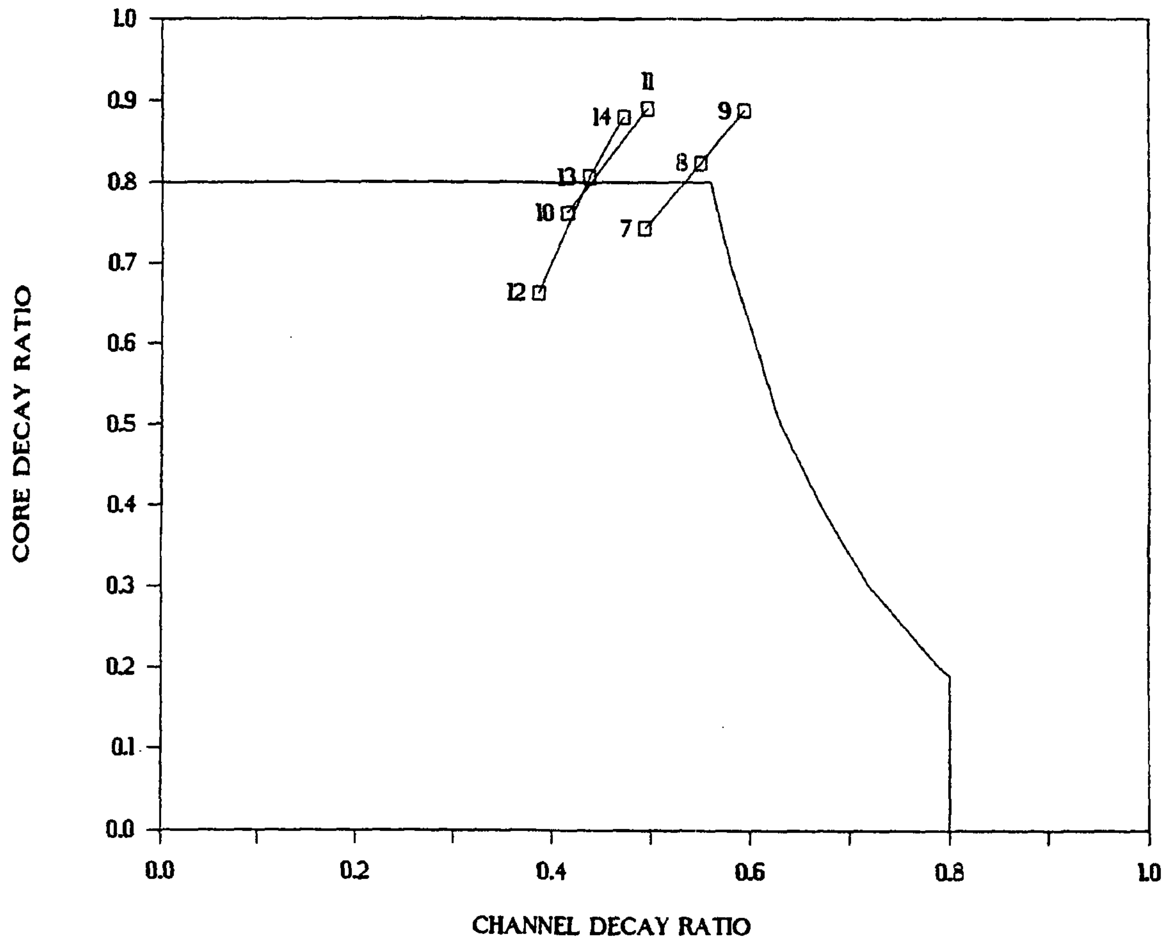


FIGURE 5-18. PERRY EQUILIBRIUM CYCLE DECAY RATIOS

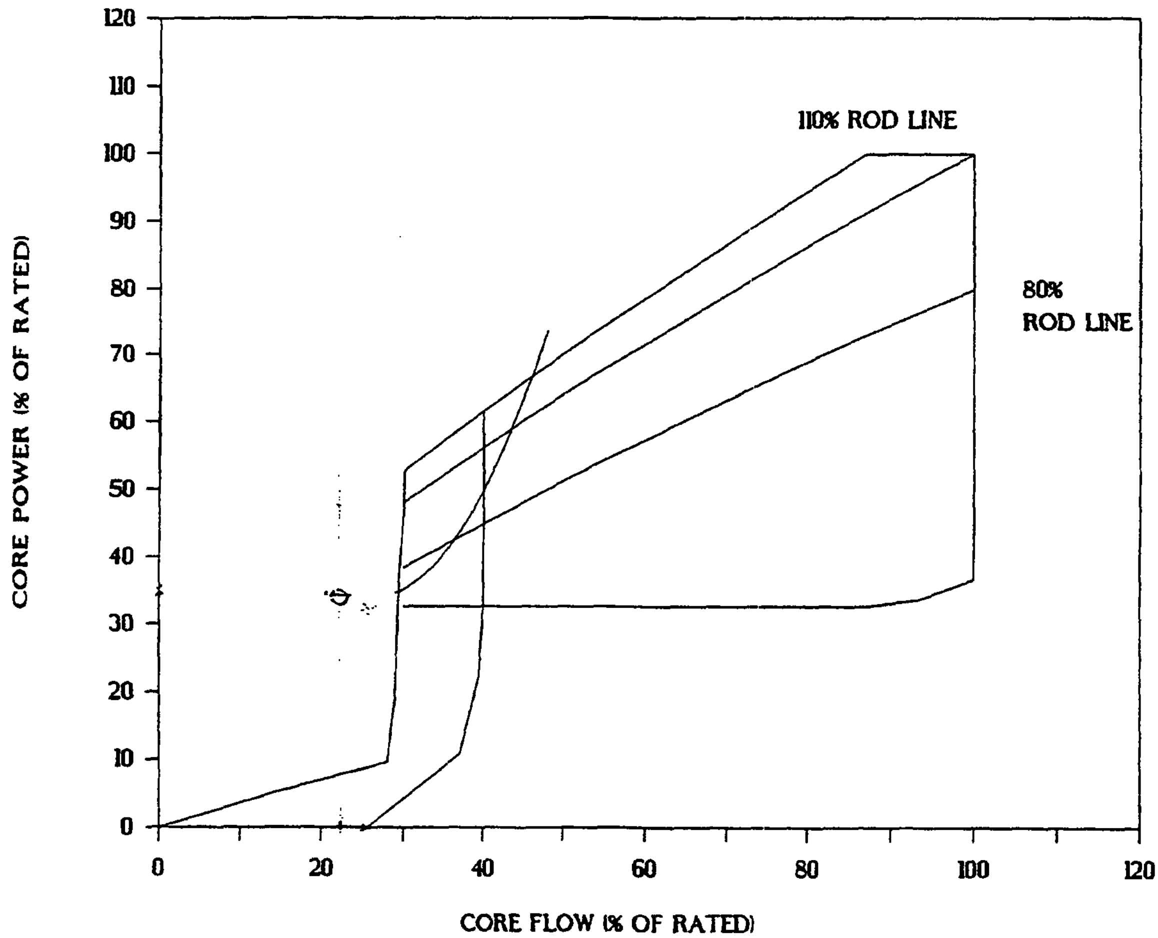


FIGURE 5-19. PERRY EQUILIBRIUM CYCLE REGION BOUNDARY DEFINITION

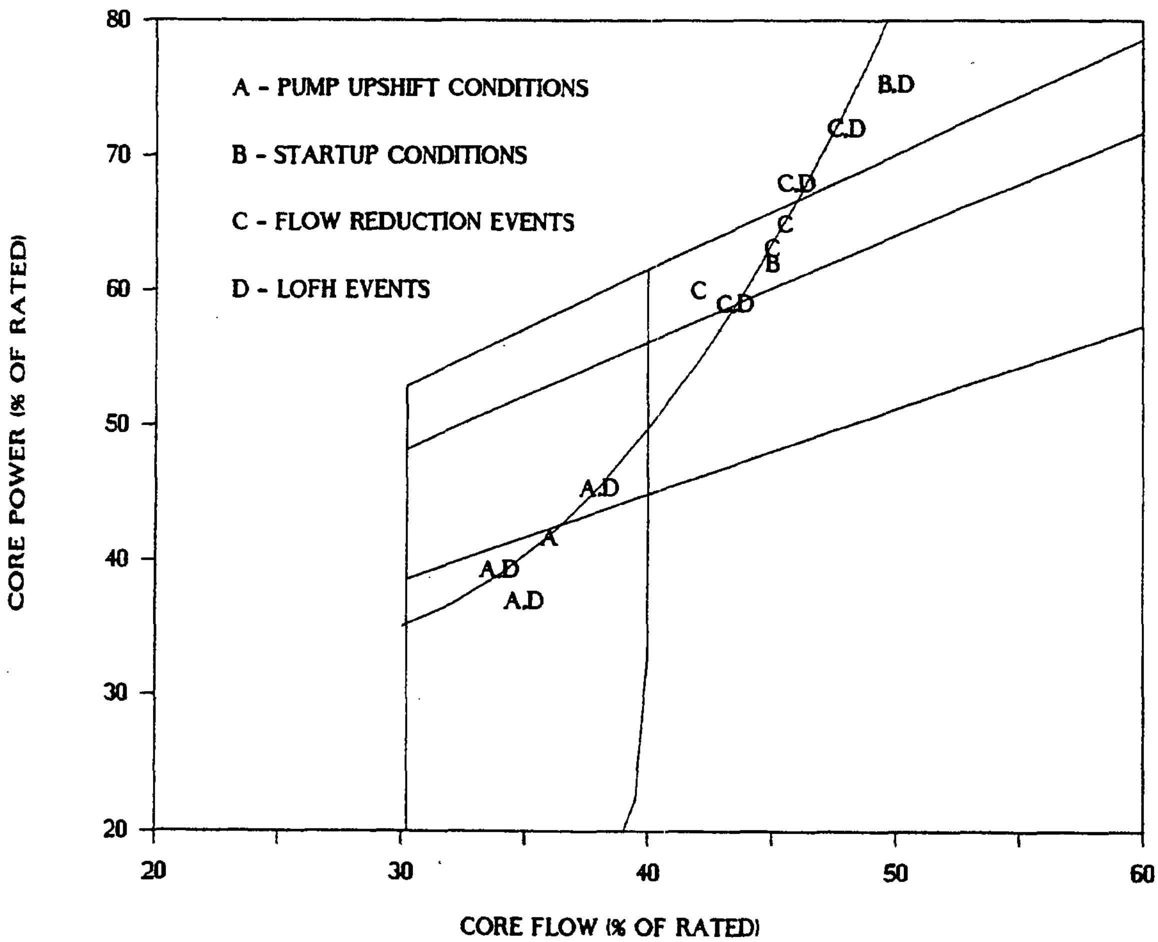


FIGURE 5-20. PERRY EXCLUSION REGION CONFIRMATION ANALYSIS POINTS

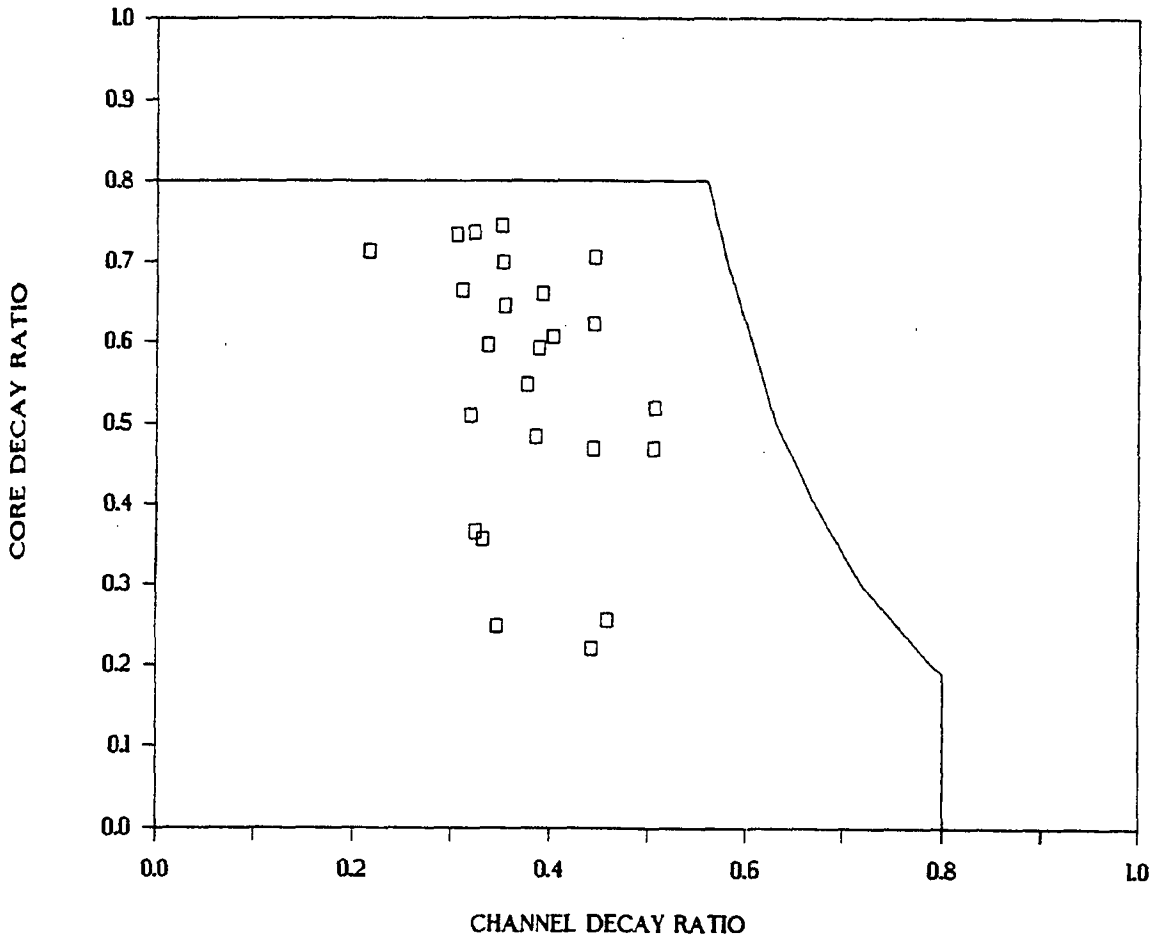


FIGURE 5-21. PERRY CONFIRMATION ANALYSIS DECAY RATIOS

6.0 DETECTION AND SUPPRESSION LICENSING METHODOLOGY

Detection and suppression systems will be designed to automatically detect and suppress stability-related neutron flux oscillations which could potentially result in conditions exceeding the MCPR Safety Limit. As discussed below, analytical MCPR Safety Limit compliance will be demonstrated for all expected modes of GE BWR thermal-hydraulic neutron flux oscillations.

6.1 EXPECTED OSCILLATION MODES

The licensing basis of detection and suppression systems is to generate a trip signal during oscillations of sufficiently low amplitude to provide margin to the MCPR Safety Limit for all expected modes of BWR oscillations.

The potential oscillation modes are as follows:

- <u>Core-wide</u> - the average neutron flux in all fuel assemblies oscillates in phase.

- <u>First Order Side-by-Side</u> - Also called a regional oscillation, where the neutron flux on one side of the reactor oscillates 180° out of phase with the flux on the other side. The axis of zero oscillation magnitude may be at any core azimuth.

- <u>First Order Precessing</u> - A regional oscillation where the axis of zero oscillation amplitude rotates azimuthally, or the two reactor regions of peak oscillation amplitude shift from one location to another at a frequency lower than the oscillation frequency.

- <u>Higher Order Modes</u> - Oscillations with more than one line of symmetry, such as a quarter core oscillation, where the neutron flux in one quadrant of the core oscillates out-of-phase with the neutron flux in the two adjacent quadrants, but in phase with the neutron flux in the diagonal quadrant.

- Single channel - Thermal-hydraulic oscillations in a single fuel assembly. Characteristic of idealized conditions in which a single fuel assembly operates under an imposed constant plenum-to-plenum pressure drop and constant heat flux.

In GE BWRs, only core-wide and first-order regional oscillations have been observed. In general, these are defined as the expected modes of oscillation for GE BWRs. For one specific class of plants, the expected modes of oscillation are further defined to include only the core-wide mode of oscillation with appropriate justification provided (Appendix A). The following discussion as to why higher order modes of oscillation and single channel oscillations are not expected modes is applicable to all GE BWRs.

6.1.1 Higher Order Oscillation Modes

The existence of regional oscillations in a BWR is the result of the excitation of radial harmonics in the neutron flux. The neutron flux distribution in a BWR is analogous to the solution of a standard wave equation in which a fundamental mode and an infinite number of harmonic modes all satisfy the necessary conditions of the steady-state equations. The primary boundary condition in a BWR is that the neutron flux vanishes near the edges of the core. In the radial plane, this is ideally satisfied by a cosine type shape, symmetric about the center of the core. Higher order cosine functions can also satisfy the edge boundary conditions. However, the geometric buckling associated with these harmonic modes is larger than that of the fundamental mode and results in the harmonic modal eigenvalues being less than that of the fundamental mode. This difference in eigenvalues is commonly referred to as the eigenvalue separation of a harmonic and is a measure of the separation between the fundamental and the harmonic mode.

At steady-state critical conditions, the eigenvalue of the fundamental is equal to 1.0 (critical) and, therefore, all harmonic modes must be subcritical (i.e., eigenvalues < 1.0). Subcritical modes will decay in time and will therefore not be present in general at steady-state conditions. For a subcritical mode to be sustained, sufficient spatial reactivity feedback must be provided to overcome the eigenvalue separation. Thermal-hydraulic

oscillations in which changes in channel inlet flows are allowed to oscillate out-of-phase (e.g., inlet flow is increasing in one half of the core while it is decreasing in the other half) are capable of providing this spatial reactivity component to excite the harmonics. Hydraulically, the reactor system tends toward a side-by-side oscillation instead of a core-wide oscillation, since the total core flow is essentially unaffected and the recirculation loop resistance to flow change does not have to be overcome. However, the flow oscillation must have sufficient spatial reactivity feedback to overcome the eigenvalue separation of the harmonics.

The most common regional oscillation that has been observed in BWRs is the first-order side-by-side oscillation. This is the first harmonic in the azimuthal direction for standard cylindrical coordinates. The mode is called first order, since there is only one line of symmetry across which the neutron flux associated with the harmonic mode changes from positive to negative. Higher order harmonics, next being the second-order harmonics (i.e., two lines of symmetry), have even higher eigenvalue separations than most first-order harmonics. An easily recognized second-order harmonic is a quarter core oscillation where the harmonic mode neutron flux is positive or negative in every other quadrant. An example of the eigenvalue separation for several harmonics from a condition that resulted in a north-to-south orientation of a side-by-side regional oscillation is shown in Figure 6-1.

For a higher order mode to be excited in a BWR, sufficient spatial reactivity feedback must be provided to overcome the eigenvalue separation. In general, any spatial reactivity perturbation expected in a BWR that would excite a higher-order mode will also excite the first-order mode in the same spatial coordinate. Since the first-order mode has significantly lower eigenvalue separation, the resulting response will be detected in the first-order mode. If the spatial feedback continues to increase, the higher order modes could theoretically become excited; however, this would be expected to occur well after the detection and suppression system has detected the first- order mode and initiated suppression of the oscillation.

A similar argument also exists for the first-order harmonic in the radial direction, referred to as an inside-outside or "donut" mode. Geometrically,

this mode has a much larger eigenvalue separation than the first-order side-by-side modes and is further precluded by operation of BWRs in quarter core and octant symmetric control rod patterns and fuel loading schemes. These last two factors result in a very heterogeneous condition azimuthally, which precludes a pure radial first-order harmonic. In general, the inside-outside mode would only be expected to occur superimposed on a second-order azimuthal harmonic mode. This further increases the eigenvalue separation and reduces the probability of sustaining the harmonic.

Based on the above discussions, it is concluded that the expected modes of oscillation in a BWR are the core-wide (in-phase mode) and the first-order side-by-side and precessing modes (see Appendix A for specific analyses which justify the core-wide mode as the expected mode of oscillation for a specific class of plants). Higher order modes and the first-order radial mode (inside-outside) are not expected to occur because of their large eigenvalue separation and the capability of the detection and suppression system to detect the first-order mode prior to sufficient reactivity feedback to excite the higher order modes. This conclusion is consistent with stability experience to date, even for the large amplitude oscillations observed at several plants.

6.1.2 <u>Single Channel Hydrodynamic Stability</u>

(1) <u>Operating and Analytical Experience</u>

Pure channel hydrodynamic oscillations (instabilities without nuclear feedback) are not an expected mode of BWR oscillations. Early reactor tests, instabilities at the Caorso plant, and several European BWR tests have demonstrated the nuclear coupling of the reactor in response to local reactivity insertion (i.e., control rod withdrawal). At least six regional oscillation tests or events have occurred in European reactors, which also demonstrated the nuclear coupling phenomenon and the fact that individual channel flow responses are driven by the regional core response.

A plot of known GE-BWR instability experiences, including the six European events, is provided in Figure 6-2. The core and channel decay

ratios were calculated after the event using the FABLE/BYPSS methodology. As can be seen in this figure, all events with high single channel decay ratios actually resulted in regional or core wide oscillations, and <u>not</u> single channel oscillations for these cores consisting of hydraulically compatible fuel. The single channel event at 0.2 core decay ratio and 1.0 channel decay ratio occurred under extreme conditions at Garigliano where a special test assembly was outfitted with a turbine device at the top of the assembly. The oscillations occurred when the device malfunctioned, creating an unusually large two-phase pressure drop.

A high channel decay ratio requires high bundle power and low core flows, both of which cause the core decay ratio to increase and result in conditions conducive to a regional or core wide oscillation.

Further evidence that single channel oscillation is not an expected oscillation mode was obtained in a TRAC-G analysis of the spatial neutronic response to a postulated single channel oscillation. A fuel assembly was modeled with artificially modified pressure drop components (i.e., it was not hydraulically compatible with the other fuel) such that without neutronic feedback, the assembly was unstable. When the assembly was introduced into the 3-D coupled neutronic thermal-hydraulic TRAC-G model, the neutronic feedback dampened the single channel response such that the channel was stabilized. Further modification of the single assembly pressure drop conditions had to be performed before the assembly would remain unstable in the coupled reactor.

This analysis demonstrates the importance of neutronic coupling and the tendency for the neutronics to prefer the core-wide mode. For a single channel to independently oscillate, significant local reactivity feedback must be present to sustain the oscillation and overcome the eigenvalue separation between the fundamental mode and the single channel oscillation conditions. For the cases analyzed with TRAC-G, it was concluded that unrealistic modifications to the channel's hydraulic characteristics were required to provide this local reactivity feedback. For cores with hydraulically compatible fuel designs, these conditions do not occur.

Additionally, BWR reactor reload design practices and the fact that reactors are typically operated in quarter core or octant symmetry assure that there are at least four to eight bundles with very similar operating conditions. This reinforces the tendency of a regional or core-wide oscillation to develop preferentially to a single channel oscillation.

(2) Single Channel Oscillation Probability and Consequences

If a single channel oscillation should occur, it is unlikely that the MCPR Safety Limit basis of 99.9% of the fuel pins avoiding boiling transition would be exceeded. For this to occur, all of the following must occur:

(a) A low core flow condition is reached where the single channel decay ratio is high, but the core decay ratio is very low. It is extremely unlikely to have a wide disparity in decay ratios and not have a regional oscillation for which automatic suppression occurs.

(b) The oscillation would have to go undetected with no operator actions taken to suppress the oscillations. This is extremely unlikely, because the nearest LPRM will undergo very large oscillations. The conditions needed to put one assembly at much higher peaking than its symmetric counterparts are not likely to occur; if postulated, these conditions would be more likely during control rod withdrawal when the operator routinely has the LPRMs displayed.

(c) The single channel oscillation must grow to a magnitude large enough to reach the MCPR Safety Limit.

(d) The oscillation reaches a magnitude sufficient to put 0.1% of the reactor fuel pins (e.g., 47 pins in a 764 fuel assembly core) in boiling transition. This magnitude depends on the number of pins required to reach 0.1% and the pin-to-pin power distribution of the assembly. At this higher magnitude, detection is even more likely.

(3) Consequences

Even if all of the above occurs, the consequences of such an event are minimal, because the fuel pin will be repeatedly quenched during the oscillations and fuel failure is not expected due to transition boiling. Considering the small probability of a single channel oscillation exceeding the MCPR Safety Limit, and the associated minimal consequences, it is concluded that automatic suppression of a single channel oscillation is not required.

6.2 OSCILLATION METHODOLOGY DESCRIPTION

Detection and suppression systems have a common element in the use of LPRMs for the detection of variations in neutron flux that are indicative of stability-related oscillations. Complex codes such as TRAC-G are capable of simulating the spatial dependence of neutron flux oscillations such that the LPRM response for a given oscillation can be determined. However, analysis using codes such as TRAC-G is very resource intensive and does not allow the evaluation of a wide range of potential oscillation characteristics, since the code will calculate a given response for a specified set of initial conditions. To demonstrate that a detection and suppression system can reliably mitigate oscillations prior to exceeding the MCPR Safety Limit, a methodology is required that will relate the MCPR performance of a limiting bundle with the expected response of LPRMs throughout the core. When this response is combined with a specific trip system configuration, confirmation of that system's setpoints can be performed. A methodology has been developed that enables the simulation of a wide variety of oscillation characteristics.

The oscillation methodology shown in Figure 6-4 has three basic components: (1) the basic oscillation model which describes the time dependent behavior of the oscillations; (2) the GE 3D BWR Simulator, which calculates the fundamental and harmonic power distributions; and (3) the MCPR performance correlation, which relates MCPR to bundle oscillation magnitude.

The oscillation methodology uses a simple model that generates an oscillating signal with a specified frequency, phase lag relative to a

reference point, relative oscillation magnitude, and average signal value (Figure 6-3). This basic oscillation model simulates the response of the peak bundle in the core or simulates the response of any of the LPRMs within the core. The GE 3D BWR Simulator is used to predict the spatial dependence of oscillations. This relies on the prediction of the neutron flux distribution in the fundamental and harmonic modes. The fundamental power distribution is used to determine the average steady-state LPRM values throughout the core (A_{ijk}). The harmonic power distribution is used to predict the relative oscillation magnitude ($[\{P-M\}/A]_{ijk}$) throughout the core, in particular, at LPRM locations. Results from TRAC-G simulations are used to relate the peak bundle oscillation magnitude to a change in CPR during the oscillation. For a given absolute magnitude oscillation in a peak bundle, this methodology is used to determine the resulting LPRM signals as a function of MCPR (Figure 6-4). These three components are discussed in more detail in the following sections.

6.2.1 Basic Oscillation Model

Based on testing and operational data, limit cycle oscillations as measured on the LPRMs resemble sine waves at low magnitude and become increasingly non-linear as the oscillation magnitude increases. This results from the fact that the minimum of the oscillation cannot go below zero and eventually most of the increase in magnitude must be realized above the average. An example of a typical non-linear flux response as measured by an LPRM is shown in Figure 6-5. This non-linear behavior is readily simulated by simplified equations describing the phenomenon (Reference 9) and can be represented by a Fourier series, where the non-linear behavior is characterized by an increase in the magnitude of the signal power at frequencies which are multiples of the fundamental oscillation frequency.

Previous studies of oscillations (Reference 2) have demonstrated the adequacy of using higher order sine waves to model the non-linear behavior of the neutron flux. As the oscillation magnitude becomes large, the non-linear effects are more pronounced and the shape of the oscillations more closely resemble higher order sine waves.

Any oscillation (bundle or LPRM) is represented in the basic oscillation model as,

$$T(t) = f(\{P-M\}/A,\ A,\ \omega,\ \theta) \qquad (6\text{-}1)$$

where

T = Oscillating parameter (peak bundle power or LPRM signal)

P = Peak value during oscillations

M = Minimum value during oscillations

A = Average value during oscillations

$\{P-M\}/A$ = Relative oscillation magnitude

ω = Frequency of oscillation

t = Time (sec)

θ = Phase angle relative to a reference point in the core

A comparison of the basic oscillation model to plant LPRM data is shown in Figure 6-6. This shows the good agreement between the oscillation model and plant data in the range of interest. Equation 6-1 is used to simulate the peak oscillation magnitude for a limiting fuel bundle. The relationship between the peak bundle oscillation magnitude and the LPRM oscillation magnitudes is determined by the GE 3D BWR Simulator as discussed in Section 6.2.2.

Since the LPRMs are distributed radially and axially within the core, their relative phase lag must be known to ensure that combinations of LPRM signals are correctly modeled. The radial component is determined by the expected oscillation mode and is discussed in Section 6.2.2. The axial phase lag is primarily influenced by thermal-hydraulics. General stability theory

(Reference 10) relates the axial phase lag to the speed of the traveling density wave. This speed is associated with the void transit time. The void transit time is directly related to the local void fraction and coolant velocity. These parameters are, in turn, dependent on subcooling and power distribution. The measured phase lags between LPRMs from a single string during testing at several BWRs is shown in Figure 6-7. Since the axial phase lags are the result of complex combinations of conditions, conservatively large axial phase lags are used for the simulation.

6.2.2 Oscillation Contours

Simulation of LPRM signals to confirm detect/suppress system setpoints requires knowledge of the oscillation magnitude throughout the core. The oscillation methodology assumes that the oscillation in neutron flux can be expressed as a function separable in space and time, where

$$\phi(\underline{r},t) = \psi(\underline{r})T(t) \qquad (6\text{-}2)$$

$= $ Neutron flux at location $\underline{r}$

$\psi(\underline{r})$ = Spatial shape function

$T(t)$ = Amplitude function

The amplitude function is an oscillatory component that only describes the normalized time dependent behavior of the oscillation (Equation 6-1). The shape function, or "oscillation contour," describes the distribution of oscillations throughout the core. This representation of the oscillation is analogous to the derivation of point kinetics models, which assumes that the basic power distribution remains constant during a transient. For the oscillation model, the oscillation distribution is assumed to remain constant.

A core-wide (in-phase) oscillation is an excitation of the fundamental mode in the radial/azimuthal plane and its "oscillation contour" is constant and equal to the core average relative oscillation magnitude. The "oscillation contour" is defined as the relative distribution of oscillation magnitudes within the core and is expressed in terms of each LPRM's (P-M)/A oscillation magnitude. For regional (out-of-phase) oscillations, a harmonic mode is

excited in the radial/azimuthal plane and results in a distribution of oscillation magnitudes radially across the core.

One source of oscillation contours for regional oscillations is from plant data obtained during oscillations. The oscillation contour for the Caorso Cycle 2 test condition (Reference 11) is shown in Figure 6-8, where the z-axis represents the oscillation magnitude for a given LPRM string, expressed as (P-M)/A. The general shape of the oscillation contour is similar to the neutron flux distribution of the first harmonic of the neutron diffusion equations. The data suggest that the relative oscillation magnitude (i..e, shape function) is characterized by the harmonic flux distribution. Modal synthesis methods (Reference 12) are commonly used to describe the neutron flux during spatial transients. These methods assume that the transient neutron flux can be expressed as a function of the steady-state modal flux distributions and associated amplitude functions. For regional oscillations, the fundamental and first harmonic mode neutron flux distributions in the radial plane are sufficient to describe the spatial behavior of the oscillations.

To evaluate the validity of this assumption, the GE 3D BWR Simulator code (Reference 8) was modified to calculate the harmonic mode of the neutron flux (Reference 13). The method has been validated by solving a benchmark problem consisting of a homogeneous rectangular parallelepiped reactor. The analytical solution to this problem is known and is accurately calculated by the modified GE 3D BWR Simulator (Table 6-1).

To calculate the radial harmonic modes with the GE 3D BWR Simulator, a standard calculation of the fundamental power distribution is performed for a specified set of initial conditions (core and fuel design, power, flow, control rod pattern, etc.). Using the modified version of the code, the power distribution in a number of harmonic modes is then calculated. The GE 3D BWR Simulator explicitly determines the potential harmonic modes based solely on the constraints of the neutronics. These are assumed to be the potential oscillation modes for the given initial conditions. Although these modes would not necessarily be sustained for the actual conditions of the plant when thermal-hydraulic feedback is included, the calculations represent the

possible oscillation modes should an instability occur. In general, only two oscillation modes are calculated, both of which are first order regional oscillations. Higher order modes can be calculated by the Simulator but are not expected modes because of the very high eigenvalue separation (see Section 6.1). For quarter core symmetric core loadings and control rod patterns, the two first-order radial modes calculated are orthogonal modes in which the oscillation axes of symmetry calculated by the Simulator are perpendicular for the two modes (e.g., if one mode has an axis that runs north-to-south, the second mode's axis runs east-to-west). For each of the radial harmonic modes, the oscillation contour (i.e., shape function) is calculated as follows:

$$\psi^n_{ij} = P^n_{ij}/P^n_{max} \tag{6-3}$$

= Normalized azimuthal oscillation contour at location i,j (normalized to maximum relative bundle power)

P^n_{ij} = Relative bundle power at location i,j for harmonic n (normalized to an average value of 1.0)

P^n_{max} = Maximum relative bundle power for harmonic n

The above expression is calculated for each bundle in one half of the core relative to the oscillation axis of symmetry. Oscillation contour values for bundles in the other half of the core are equal to the value for their symmetric bundle. For bundles in the other half of the core, a radial phase lag of 180° is also needed. The contour has been specifically normalized to the oscillation magnitude of the peak bundle, since this is the limiting oscillation in the core. The methodology is centered around first specifying the oscillation magnitude for the peak bundle, and then determining the response of the LPRMs relative to the peak bundle. An individual LPRM detector reading is assumed to be proportional to the response of the surrounding eight fuel nodes (two axial nodes from each of the surrounding four fuel bundles at the appropriate elevation). An example of an oscillation contour is shown in Figure 6-9, where the values represent the oscillation magnitude for a specific LPRM relative to the oscillation magnitude for the peak bundle. For example, the LPRM at 16-09 is calculated to oscillate at 85% of the oscillation magnitude of the peak bundle.

As discussed above, this shape function is assumed to represent the oscillation magnitude at any radial location within the core. Because the harmonic flux distribution is relative to the fundamental distribution (i.e., the harmonic flux distribution describes the change in neutron flux relative to the fundamental or steady state flux distribution), it is assumed to be a representation of the (P-M)/A oscillation magnitude distribution. Using this calculation of the oscillation contour, the method has been compared to stability data taken at an operating BWR. Comparison of the (P-M)/A for LPRM readings during a regional oscillation with a contour plot of the first harmonic of the neutron flux from the GE 3D BWR simulator shows good agreement (Figure 6-10).

Once the normalized azimuthal oscillation contour, ψ^n_{ij}, is determined, the oscillation magnitude expressed as (P-M)/A at any location (i,j) can be determined by first choosing a value for the peak oscillation magnitude. At each location (i,j), the oscillation magnitude is then determined as

$$\{(P\text{-}M)/A\}_{ij} = \psi^n_{ij} * \{(P\text{-}M)/A\}_{max} \tag{6-4}$$

This model conservatively assumes the oscillation magnitude is constant axially within a bundle or LPRM string. Therefore,

$$\{(P\text{-}M)/A\}_{ijk} = \{(P\text{-}M)/A\}_{ij}. \tag{6-5}$$

6.2.3 <u>MCPR Performance</u>

The previous sections have identified the methodology for describing the spatial and temporal dependence of oscillations for the peak bundle and for each LPRM in the core. Since the purpose of the methodology is to confirm that a specific trip system protects the MCPR Safety Limit, the final piece of the methodology is to determine the change in CPR during a specified oscillation. Fully-coupled TRAC-G calculations of core-wide and regional oscillations are performed for a variety of plant and fuel types at various conditions. Each of these cases generates a certain oscillation magnitude and resultant CPR change in the limiting bundle. A correlation is then developed relating the CPR change to the peak bundle oscillation magnitude relative to

the average bundle power. Based on existing TRAC-G evaluations, correlations have been developed for the relative change in CPR during oscillations as a function of the relative oscillation magnitude. Example correlations for the response during regional oscillations are shown in Figure 6-11. The correlations represent the different responses for relatively high channel decay ratios and for more nominal channel decay ratios.

6.2.4 Example Application of Oscillation Methodology

The oscillation methodology used to calculate MCPR response relative to LPRM oscillation magnitudes is summarized in Figure 6-4. To demonstrate the use of the methodology, an example calculation is performed for a 560 bundle BWR/4 plant. The initial conditions prior to the oscillation are natural circulation near the rated rod line, at end of cycle conditions with all control rods withdrawn. These conditions are analyzed using the GE 3D BWR Simulator to calculate the initial LPRM distribution, initial radial peaking factor and MCPR for the hot bundle. The initial conditions are summarized in Table 6-2. The Simulator is also used to calculate the oscillation contour. The first harmonic is calculated to be a regional oscillation with an oscillation axis of symmetry along the northwest-southeast diagonal and the contour is shown in Figure 6-12. The oscillation period is assumed to be 2.0 seconds and the axial phase lags are based on measured data from plant stability tests. The peak bundle oscillation magnitude ((P-M)/A) is assumed to be 1.0, which represents an oscillation magnitude of 100% of the average steady-state value. The oscillation characteristics are also summarized in Table 6-2.

The basic oscillation model is then used to predict the response of the LPRMs and the peak bundle. The LPRMs are also combined to form their respective APRM channels and are combined to form OPRM cells consisting of eight LPRMs each (Appendix A). The results of the simulation are summarized in Table 6-3.

The calculated behavior of the peak bundle power is shown in Figure 6-13. LPRM responses for different locations within the core, as well as for different axial locations within a single LPRM string, are shown in Figures

6-14 and 6-15. The axial and radial phase lags can be seen in the figures. Representative APRM channel responses are shown in Figure 6-16 and several OPRM channel responses are shown in Figure 6-17.

6.3 APPLICATION OF OSCILLATION METHODOLOGY

To confirm that a specific detection and suppression system adequately protects the MCPR Safety Limit, analyses must be performed that cover a representative range of potential initial conditions and oscillation characteristics. In addition, the uncertainties in the modeling and inputs must be considered in the confirmation calculations. The basic oscillation methodology which calculates the MCPR performance as a function of LPRM response for a specific set of initial conditions is described in Section 6.2. To provide the best representation of expected oscillation responses, the most sensitive inputs to the methodology are randomly chosen from distributions which describe the expected variation of the parameters. These randomly selected inputs (Section 6.3.2) are then evaluated using the oscillation methodology to produce a specific MCPR for a given case.

A large number of cases is evaluated and the 95% probability/95% confidence level MCPR ($MCPR_{95/95}$) is determined. The 95% probability/95% confidence level acceptance criterion assures that a measurable and adequate level of conservatism is included. The $MCPR_{95/95}$ value is then compared to the MCPR Safety Limit to confirm that the specific combination of inputs is acceptable.

6.3.1 Design Objective

Standard reload licensing analyses typically calculate the change in CPR for a set of transients to determine the most limiting event. The CPR change during the limiting transient is then added to the MCPR Safety Limit to establish the MCPR operating limit. This technique is established to ensure that, if the limiting transient were to occur when the plant was operating at the MCPR operating limit, the MCPR Safety Limit would not be exceeded. The design objective for detection and suppression systems is to provide mitigation such that an instability does not have to be considered in the evaluation of limiting events. This design objective is satisfied by

performing analyses which assume that the plant is at the MCPR operating limit established by the standard licensing analysis, and then confirming, with a high degree of confidence, that the final MCPR during the defined instability has a margin to the MCPR Safety Limit of 0.10. The high degree of confidence is provided by the use of Monte Carlo statistical methods which simulate the various uncertainties associated with calculating MCPR as a function of a trip system's response. Use of the MCPR Safety Limit is also widely recognized as a conservative measure of fuel cladding integrity since damage is not likely during brief periods of departure from nucleate boiling followed by the quenching that occurs during density wave oscillations.

In addition to the explicit treatment of known uncertainties by the Monte Carlo analysis, additional conservatisms exist in the methodology. These conservative assumptions include, but are not limited to, the assumption that (1) every two recirculation pump trip event results in an instability, (2) the instability grows at a rate too fast for the operator to take manual actions, (3) the reactor is at MCPR operating limits just prior to the initiating event, and (4) the most responsive RPS channel is failed at the time of the instability event.

The statistical treatment of known uncertainties and the additional conservative assumptions provide flexibility for the initial design and development of detection and suppression systems. Without this margin, future modifications to CPR correlations, fuel designs, trip system features and other factors affecting stability could potentially result in unacceptable restrictions on plant operation or unnecessary reactor trips caused by overly restrictive setpoints.

6.3.2 Analysis Procedure

The objective of the analysis procedure is to determine the final MCPR during an oscillation event given expected probability distributions of input parameters. The inputs describe the initial reactor conditions before the oscillations, the oscillation characteristics, the trip system design features and the bundle MCPR performance during the oscillations. Figure 6-18 shows a flowchart of the analysis procedure. The initial conditions determine the

initial MCPR prior to oscillations. The important oscillation characteristics are the oscillation contour and the rate at which the oscillation magnitude is growing. The trip system design features describe how the LPRMs are combined and evaluated relative to a trip algorithm and setpoints. The trip algorithm, setpoints, and oscillation growth rate determine the maximum amplitude of the oscillations before the ASF suppresses the oscillation. The following sections discuss the different input assumptions and how they are represented in the analysis.

6.3.2.1 Initial Conditions

The initial reactor operating conditions are important in determining the conditions just prior to the onset of oscillations. This is the primary factor in determining the initial MCPR at the onset of oscillations. The initial conditions are separated into two categories: (1) steady-state operating conditions and (2) an initiating event that is assumed to result in the reactor operating at potentially unstable conditions. The potential initial conditions are limited to those that are expected to result in an instability. Oscillations are most likely to occur in the low flow/high power region of the operating domain. This region can be entered as the result of core flow reduction or power increase. In general, power increases are not performed in the region of potential instability. Possible power increases near this region due to control rod withdrawal or feedwater heating loss are relatively slow increases during which the probability of operator intervention, should the onset of oscillations be detected, is very high.

Core flow reductions (e.g., two recirculation pump trips), however, have the potential to result in very large changes in operating conditions in a relatively short time and the operator has less opportunity to intervene during the transient plant conditions. Because of the large change in operating conditions, it is also possible that operation could occur beyond the point of oscillation inception. This is more likely to lead to oscillations that grow to magnitudes which require mitigation when compared to a gradual approach to the instability such as would be experienced during a power increase. The limiting flow reduction event is therefore the two recirculation pump trip event, which results in operation at natural

circulation flow. A two recirculation pump trip (RPT) is assumed to be the initiating event for the analysis procedure.

The initial power level prior to the RPT is also an important factor in determining the probability of oscillations and the MCPR prior to the onset of oscillations. In general, operation at higher rod lines results in less stable conditions following an RPT. Therefore, the initiating event is assumed to start from the highest licensed rod line for a particular plant. For a specific initial power and flow condition along the maximum allowed rod line, the plant Technical Specifications require the MCPR to be greater than the MCPR Operating Limit. For the oscillation analysis, it is conservatively assumed that the plant is operating at the MCPR Operating Limit.

Since the majority of the operating cycle is spent at full power operation, the analysis procedure assumes that 95% of the cases begin at full power conditions. The remaining cases are assumed to start from conditions that are representative of startup conditions. During a typical plant startup, control rods are withdrawn along a minimum core flow line (approximately 40% of rated core flow) until the full power control rod pattern is attained, the maximum allowed rod line is reached, or thermal limits are reached. These startup cases also assume an RPT, since the potential for an instability following an RPT is the greatest. For these conditions, the appropriate MCPR Operating Limit at the power/flow conditions specified is assumed.

During the evolution of the RPT, the MCPR increases as core flow decreases to natural circulation conditions. Figure 6-19 shows the MCPR just prior to an RPT and after equilibrium conditions have been reached following the flow reduction. The data points represent the MCPRs calculated by the plant online process computer for various flow reduction events performed during startup test programs at a number of BWRs. The flow reduction events include the trip of both recirculation pumps, recirculation pump runbacks, and single recirculation pump trip events. From the figure, it can be seen that the slopes of the curves which represent the MCPR increase during the flow reduction is relatively constant. This is expected, since the major parameter which determines the final MCPR following the flow reduction is the final power. The final power is dependent on the void reactivity coefficient, axial

power distribution, and change in core flow during the flow reduction. Because these parameters will vary during a cycle and from plant to plant, a variation in the final power level at a given core flow along a particular rod line is expected. This variation is described by a distribution of a parameter that represents the change in CPR during the flow reduction. Based on the plant data, the following parameter is defined:

$$\text{DIDW} = \frac{(\text{MCPR}_2 - \text{MCPR}_1)/\text{MCPR}_1}{(W_1 - W_2)}, \tag{6-6}$$

where,

MCPR_1 = Initial MCPR prior to RPT
MCPR_2 = Equilibrium MCPR following RPT
W_1 = Initial core flow (% of rated)
W_2 = Final core flow (% of rated).

Since only limited data exist from plant flow reduction events, the GE 3D BWR Simulator is used to generate a representative set of data for the parameter DIDW. Analyses are performed for a wide variety of plant and fuel types, power distributions and initial power/flow conditions. To verify that the GE 3D BWR Simulator provides an accurate prediction of MCPR along the flow control lines, a comparison with startup test data was performed. The results for two flow reduction events are shown in Figure 6-20, which demonstrate the good agreement between the GE 3D BWR Simulator and the process computer calculations during actual plant tests. An appropriate distribution is generated from the results of the Simulator calculations and, for each case, a random sample is chosen to represent the change in MCPR during the flow reduction portion of the analysis. When combined with the randomly selected initial power/flow condition (full power or startup conditions) and the flow reduction to natural circulation, the initial MCPR prior to the assumed onset of oscillations is determined. The initial MCPR is calculated as:

$$\text{IMCPR} = \text{TSMCPR} * \{1.0 + \text{DIDW} * (W_1 - W_2)\}, \tag{6-7}$$

where,

IMCPR = Initial MCPR prior to the onset of oscillations at equilibrium conditions reached following the RPT

TSMCPR = MCPR Operating Limit at power/flow conditions prior to the RPT

DIDW = Relative change in CPR during the flow coastdown (equation 6-6)

W_1 = Initial core flow (% of rated)

W_2 = Final core flow, assumed to be natural circulation (% of rated).

In addition to the IMCPR, the radial peaking factor must be determined since the calculation of the peak bundle oscillation magnitude is dependent on the initial bundle power. The radial peaking factor is primarily dependent on the control rod pattern and fuel loading and therefore is expected to vary with cycle exposure and core and fuel design. The GE 3D BWR Simulator is used to calculate the radial peaking factor for a wide range of core and fuel designs and control rod patterns. These results are used to generate a distribution of radial peaking factors from which random samples are taken for each specific case.

6.3.2.2 Oscillation Characteristics

(1) Oscillation Contours

The previous section discussed the initial conditions and initiating event that result in conditions just prior to the assumed onset of oscillations. It is conservatively assumed that every RPT from the maximum allowed rod line results in an oscillation that eventually grows to a magnitude sufficient to exceed the trip system setpoints without any operator intervention. For the majority of plants, the expected modes of

oscillation have been defined as the core-wide oscillation mode and the first-order harmonic mode. For core-wide oscillations, all fuel nodes at the same axial location oscillate in phase, resulting in the maximum LPRM, OPRM cell, and APRM responses for a given peak bundle oscillation magnitude. As discussed in Section 6.2.2, the oscillation contour for core-wide oscillations is constant in the x-y plane everywhere in the core and is equal to the magnitude of the peak bundle oscillation. For the regional oscillations (first-order harmonic mode), the oscillation contour is calculated by the GE 3D BWR Simulator, and varies across the core as a function of the distance from the oscillation line of symmetry. Although the expected oscillation modes for an individual plant would include both the core-wide and regional oscillation modes, regional oscillations will be conservatively assumed as the only oscillation mode.

To evaluate the impact of different plant conditions on the oscillation contour, oscillation contour calculations were performed for a variety of plant sizes, operating cycles, control rod patterns, cycle exposure, and Xenon concentrations. The conditions that have been analyzed are summarized in Table 6-4. In general, the oscillation contours are relatively insensitive to most parameters and are determined mainly by geometry. To a lesser extent, the radial power distribution affects the oscillation contour. This is mainly seen in areas near inserted control rods and near the core periphery. Control rod pattern symmetry and fuel loading symmetry play the dominant role in determining the axis of symmetry for the first order azimuthal harmonics.

A portion of the oscillation contour for six different 560 bundle core conditions is shown in Figure 6-21. Each of the oscillations has a NWSE axis of oscillation symmetry. The oscillation contours in Figure 6-21 are based on LPRMs that lie on a diagonal that is perpendicular to the axis of oscillation symmetry (i.e., the NESW diagonal). These LPRMs are therefore at an azimuthal angle of 90° relative to the oscillation axis of symmetry, which represents the maximum oscillation magnitudes for a given radius from the core center. The contours in Figure 6-21 show the same characteristic shape as observed from plant instability data, with some variation from case to case. Of particular interest is the relative oscillation

magnitude, for the LPRMs located near the point of peak oscillation magnitude, since these LPRMs provide the maximum signal response. At these locations, the contours do not show a significant scatter. This trend is also true for other core sizes.

Because of the different number of LPRMs for different core sizes, the response of trip systems may vary as a function of core size and number of LPRMs. Therefore, the final setpoint analyses will be performed for each specific LPRM configuration. Because of the relative consistency among the contours, the resulting LPRM response as a function of peak bundle oscillation magnitude is not expected to vary significantly for different contours. Therefore, only a limited number of contours need to be generated for a given core size.

A more important feature of the oscillation contour is the axis of oscillation symmetry. Because of the LPRM symmetry within the core, the trip system responses vary as a function of the oscillation axis of symmetry. For most plants, the NESW and NWSE axes of symmetry are the dominant axes because the plants are only operated in an A2 sequence (i.e., control cell core operating strategy), which typically results in octant symmetric control rod patterns. For plants that operate with conventional cores, operation in the B sequence can result in power distributions that are not symmetric about the diagonals and therefore result in EW and NS axes of symmetry. Therefore, these cores can oscillate about one of the four possible oscillation contour axes. Evaluations have been performed of the system responses as a function of the oscillation axis of symmetry and it has been found that the NESW and NWSE axes of symmetry result in the limiting (i.e., lowest) system responses. Therefore, only these modes of oscillation are assumed in the analysis. This is a conservative assumption for plants that operate with conventional cores (i.e., A and B sequences).

(2) <u>Oscillation Growth Rate</u>

Following the RPT, it is assumed that oscillations begin to grow and eventually reach a limit cycle oscillation. During the time when the oscillation magnitude is growing, the specific trip system will detect the

oscillations and initiate an ASF, thereby mitigating the effects of the oscillation. The oscillation model described in Section 6.2 evaluates only one cycle of the oscillation. This cycle is the limiting cycle that results in the peak oscillation magnitude. When the trip system signal exceeds the trip setpoint, the oscillation continues to grow until control rod motion from the ASF begins to mitigate the oscillation.

This effect is illustrated in Figure 6-22 for a growing oscillation that exceeds an absolute magnitude trip setpoint. If the timing of the ASF is such that the oscillation is mitigated before the next cycle begins, the peak oscillation magnitude is represented by the oscillation peak during the cycle that exceeded the trip setpoint ("First Peak" as labeled in Figure 6-22). If the initiation and mitigative action of the ASF is slow relative to the oscillation frequency, it is possible that the "second peak" (as labeled in Figure 6-22) represents the peak oscillation magnitude. These peak oscillation magnitudes are called the signal overshoot, since they represent the amount that the monitored signal overshoots the trip setpoint. Since the oscillation model is based on specifying the peak bundle oscillation magnitude and then calculating the resultant trip system response, the signal overshoot is added to the peak bundle oscillation magnitude prior to determination of the MCPR response. The amount of overshoot is dependent on the growth rate of the oscillation, the amount of inherent noise superimposed on the trip system signal, and the trip system algorithm and setpoint design.

A review of plant stability data shows a range of oscillation growth rates from 1.0 to 1.3. The plant data also show the presence of background noise levels that are always present in BWRs during power operation. Noise components that affect the oscillation response are those with a frequency similar to the fundamental oscillation frequency.

Electronic noise that is always present in the Neutron Monitoring System (NMS) does not significantly affect the trip system responses, since the frequency is much higher than the oscillation frequency and will in general, be filtered out by the detection and suppression systems. An example of measured plant LPRM data during an oscillation with increasing

magnitude is shown in Figure 6-23 which also shows a simulated oscillation with no background noise component. The oscillation with no background noise shows an exponentially increasing envelope that can be characterized by a constant growth rate during most of the increase. The plant data show the same general trend but with an additional noise component superimposed on the basic growing oscillation. This noise component results in a widely varying growth rate when successive oscillation peaks are evaluated and affects the calculated overshoot.

A point kinetics model which includes simplified equations representing the void and Doppler reactivity feedback components (Reference 9) is used to generate a range of oscillation growth rates. Background noise is simulated by introducing a random noise source to the fuel temperature term in the model. This random noise source results in the characteristic neutron flux noise behavior experienced at operating BWRs. An example of a simulation for reactor conditions which correspond to a decay ratio of 0.48 (stable) is shown in Figure 6-24. The steady-state noise levels that are characteristic of BWRs are well represented by the point model. Examples of oscillations generated by the point model are shown in Figures 6-25 and 6-26. These oscillations represent growth rates of approximately 1.05 and 1.30, respectively. These oscillation "scenarios" are used to determine the expected trip signal overshoot for a specific trip system algorithm design and setpoints (Section 6.3.2.3). Scenarios have been generated with growth rates ranging from 1.05 to 1.60, with the majority of the growth rates near 1.30. A growth rate of 1.30 was chosen as the dominant growth rate, since this bounds available plant stability data.

For the analysis procedure, the generated scenarios are evaluated against a specific trip system algorithm and corresponding setpoints to determine the distribution of the signal overshoot. An example distribution of setpoint overshoot for the OPRM system described in Appendix A is shown in Figure 6-27. The overshoot distribution corresponds to the distribution of "second peaks" as defined in Figure 6-22. The "second peak" is conservatively chosen for BWR/3-5 plants, since the Technical Specification scram times are of the same magnitude as the expected oscillation frequency (2.0 seconds to 50% insertion). For BWR/6

plants, the Technical Specification scram times are shorter than the expected oscillation period and, therefore, the ASF mitigates the oscillation prior to the second peak. The first peak is used to represent the overshoot distribution for BWR/6 plants.

For the OPRM system, the trip system signal and setpoints are expressed in terms of the signal value relative to a time-averaged value. For the evaluation of a specific case, a randomly selected overshoot is chosen from the specified distribution. This represents the maximum value that the trip system signal reaches during the oscillation. Based on this input, the peak bundle oscillation magnitude that results in this signal magnitude is then determined. A nearly identical procedure is used to generate the overshoot distribution for evaluating the APRM response for Options I-C and I-D, and the LPRM response for Option III-A.

6.3.2.3 Trip System Definition

Once the initial conditions, initiating event, oscillation contour, and oscillation growth rate are defined for a particular case, the oscillation methodology is used to simulate the response of the LPRMs and peak bundle. To determine the peak bundle oscillation magnitude that results in the specified setpoint overshoot, the trip system design must be specified. This design includes the assignment of LPRMs to RPS trip channels, any averaging of LPRMs (e.g., APRMs, OPRMs), evaluation of the trip system signal using algorithms designed to detect oscillations (see Appendix B), and the trip system setpoints. The trip system stability detection algorithm and trip setpoints are important in determining the overshoot distribution. The assignment of LPRMs to RPS trip channels and any averaging of LPRMs is simulated by the oscillation methodology so that the trip system response is accurately represented. Appendix A provides examples of trip system designs. Since the channels of the trip system will not all have the same response during the oscillations, the most responsive channel is conservatively assumed to fail, and the next most responsive channel is assumed to initiate the trip during the oscillations. This channel's response is combined with the required signal overshoot to determine the peak bundle oscillation magnitude.

6.3.2.4 LPRM Failures

Because the detection and suppression systems are based on LPRMs, failed LPRMs that are out-of-service may affect the responsiveness of the systems. To account for this, each case is assumed to have a fraction of the LPRMs failed and bypassed in the respective trip systems. The number of failed LPRMs is based on a random sample from a distribution which represents the expected LPRM failure rate. Plant data have been evaluated to determine the expected LPRM failure rates. Based on a survey of eight plant designs covering a wide range of plants and operating cycles, failure rate distributions have been developed for beginning of cycle, middle of cycle and end of cycle conditions. As expected, the failure rate for the end of cycle is the highest. The distributions do not show any bias relative to plant size, and show a downward trend with time (i.e., reduction in LPRM failures over the last few years). The distribution of composite LPRM failure rates from the plant data is shown in Figure 6-28.

Once the LPRM failure rate has been randomly selected for a specific case, the location of the failed LPRMs is determined by assuming each LPRM has an equal probability of failure. This results in a random distribution of failed LPRMs. This is also consistent with plant data which showed no particular bias toward LPRM axial or radial location. Failed LPRMs are assumed to be bypassed in the respective trip system channel and therefore do not contribute to the channel response.

6.3.2.5 MCPR Performance

The TRAC-G model is used to generate a correlation for CPR change as a function of peak bundle oscillation magnitude. This correlation is generated with a sufficient number of cases to adequately characterize the CPR performance for various plant and fuel designs. In particular, the correlation is evaluated for its sensitivity to channel hydrodynamic decay ratio. Examples of correlations that have been developed based on previous calculations using the TRAC-G model are shown in Figure 6-11. For a particular plant and fuel design, a representative correlation will be specified, where the correlation may be described by a distribution of CPR

changes for a given peak bundle oscillation magnitude. For a given case being analyzed by the oscillation model, the change in CPR during the oscillations will be determined from the appropriate correlation using the peak bundle oscillation magnitude including the effects of signal overshoot. The final MCPR for the specific case can then be determined by:

$$FMCPR = IMCPR * (1.0 + \Delta/IMCPR) \qquad (6\text{-}8)$$

where

FMCPR = Minimum CPR during the oscillations

IMCPR = Initial MCPR prior to the onset of oscillations

Δ/IMCPR = Relative change in CPR during oscillations.

6.3.3 Examples of Monte Carlo Analysis

To demonstrate the application of the detection and suppression methodology, sample calculations are performed for a 560 bundle BWR/4 plant. The following inputs are assumed for the analysis.

(1) Initial Conditions

As discussed in Section 6.3.2.1, the reactor is assumed to begin operation at one of two conditions: (1) full power or (2) high power/low flow conditions expected during startup. It is assumed that 95% of the time the reactor is operating at full power and 5% of the time at the startup conditions. The reactor is assumed to be operating at the maximum allowable rod line (e.g., 110% of rated), since this results in the most limiting initial MCPR just prior to oscillation onset. The rated power MCPR operating limit is assumed to be 1.25. For a BWR/4 with a standard K_f curve, the operating limit at 40% core flow (assuming a maximum flow runout of 102.5% of rated) is 1.44. The initiating event is a two recirculation pump trip which results in an increase in MCPR as flow reaches natural circulation. The change in CPR during this flow reduction is simulated by

data from process computer calculations during flow reduction events at operating BWRs (Figure 6-19) which are characterized by a normal distribution. The radial peaking factor is based on GE 3D BWR Simulator calculations that were used to generate the oscillation contours in Table 6-4. The radial peaking factor is also assumed to be normally distributed.

(2) Oscillation Contours

Six oscillation contours (three NESW and three NWSE modes) are chosen from the contours generated for the 560 bundle core size (Table 6-4). The contours for Cycle 9 and Cycle 12 conditions were chosen. The axis of oscillation symmetry is a diagonal for each case (NWSE/NESW modes), which is a conservative assumption for plants that operate in B sequences, since the NS and EW modes (which show improved LPRM response) could also be expected to occur.

(3) Oscillation Growth Rate

The oscillation scenarios described in Section 6.3.2.2 are used to simulate the potential oscillation magnitude growth rates. There are 22 scenarios which range in growth rates from 1.05 to 1.60. Each scenario has a random noise component superimposed on the oscillation which is representative of noise levels expected during operation.

(4) Trip System Definition

An OPRM, (Appendix A), is used to detect the oscillations and initiate the ASF. The LPRM assignments to OPRM channels for the 560 bundle core are shown in Figures 6-29 and 6-30. The high-low-high detection algorithm defined in Appendix B is used with a maximum trip setpoint of 1.20 (peak-to-average). Based on the 22 oscillation scenarios and the specified trip algorithm and setpoints, the distribution of peak OPRM signals prior to mitigation is shown in Figure 6-27. The second peak following ASF initiation is chosen to define the distribution, since the BWR/4 technical

specification scram times are similar to the oscillation period. It is assumed that, during the oscillations, the single worst RPS channel failure occurs.

(5) LPRM Failures

Average LPRM failure rates used in this example have been determined from plant data for beginning, middle and end of cycle as 6%, 8%, and 9% respectively. The distribution of LPRM failure rates is also based on plant operating data and is similar to the distribution shown in Figure 6-28. The failed LPRMs are assumed to be randomly distributed throughout the core.

(6) MCPR Performance

The MCPR performance during oscillations is based on Curve 1 of Figure 6-11. This curve represents a least squares fit of the calculational results from TRAC-G analysis of a BWR/5 with loose inlet orifice diameter (2.43 inches).

(7) Results

The inputs for the example calculations are summarized in Table 6-5. The various distributions are randomly sampled to generate 5000 cases, with the result of each case being a final MCPR. A statistical evaluation of the results is performed to determine the $MCPR_{95/95}$ result. The results of the analysis are summarized in Table 6-6, where the mean value of the MCPR response is 1.404. The $MCPR_{95/95}$ value is determined by applying an appropriate tolerance factor to the standard deviation of the 5000 cases

where

$$MCPR_{95/95} = MCPR_{mean} - k*\sigma \qquad (6\text{-}8)$$

Since only 22 oscillation scenarios are used, it is assumed that the 5000 cases only represent 22 independent cases and, therefore, 2.35σ is

subtracted from the mean to determine the $MCPR_{95/95}$ value of 1.23 for this example.

6.4 PLANT- AND CYCLE-SPECIFIC APPLICATION OF GENERIC ANALYSIS

Section 6.3 summarizes the application of the oscillation methodology for a specific core size and Technical Specification limits. Although this procedure could be applied to any individual plant, generic analyses will be performed for plants with similar characteristics. The primary characteristic that distinguishes a plant group is the core size. Because of the wide range of MCPR operating limits and possible trip system designs and setpoints, it is necessary to perform sensitivity studies to ensure that all plants within a plant group are initially bounded by the results. In general, these sensitivities will be performed for varying MCPR operating limits and trip system algorithms and setpoints. These studies will establish acceptable trip system setpoints as a function of MCPR operating limit. An example of the form in which the generic results will be presented is shown in Figure 6-31 for the OPRM system (Appendix A).

To confirm that the generic results are applicable to a plant within the group, a comparison of the major parameters affecting the system response and CPR performance must be made. If all parameters are within a defined range, the appropriate trip system setpoints can be applied to the specific plant. This section discusses the key parameters that must be evaluated to determine the applicability of the detection and suppression system setpoints.

(1) <u>Initial Conditions</u>

The generic analyses will cover a sufficient range of MCPR operating limits and allowable rod lines to ensure that current and expected core and fuel designs are covered. If a plant's MCPR operating limit changes, the acceptability of existing setpoints can be confirmed from the generic analyses. Other parameters associated with initial conditions are relatively minor contributors to the overall results and are not expected to vary significantly as a function of fuel and core design.

(2) Oscillation Contours

The variation in oscillation contours is primarily a function of neutron leakage and control rod patterns. The variation in oscillation contour for a 560-bundle core is shown in Figure 6-21. A similar result is also found when contours are compared from different size plants. The oscillation contours from a range of core sizes are shown in Figure 6-32. The oscillation magnitude is plotted as a function of the relative distance from the core center for LPRMs that are located along a diagonal which is perpendicular to the axis of oscillation symmetry. These results demonstrate that the variation in oscillation contours as a function of core size is similar to the variation due to power distribution and is primarily controlled by the cylindrical geometry of the core. A generic set of contours will be used for the generic analysis and no additional evaluations are required during plant- and cycle-specific confirmations.

(3) Oscillation Growth Rate

The generic analyses will be performed using the range of oscillation growth rates discussed in Section 6.3.2.2, which covers the range of expected growth rates. These growth rates are not expected to vary significantly as a function of core and fuel design and, therefore, no additional confirmations are required.

(4) Trip System Definition

The generic analyses will explicitly cover the trip system configurations described in Appendix A, detection algorithms, and range of expected setpoints. Any modifications outside the evaluated range will require additional analysis.

(5) LPRM Failures

The LPRM failure rates have been determined from actual plant data and have shown a gradual decrease over time. Restrictions on the number of LPRMs required to be operable for accurate power distribution monitoring

and APRM operability provide sufficient control on the expected failure rates. A relatively large fraction of the cases evaluated in the Monte Carlo simulations (10-15%) result in LPRM failures in excess of those allowed by Technical Specifications. This demonstrates the conservatism in the basic assumptions. In addition, sensitivity studies have been performed which show that the overall impact of failed LPRMs is small for the OPRM and LPRM-based systems described in Appendix A. Therefore, the assumptions in the generic analysis are sufficient to preclude the need for future confirmations.

(6) MCPR Performance

The two most important parameters in determining the final MCPR are the initial MCPR and the change in MCPR during oscillations. Generic correlations will be developed for a range of fuel designs which describe the change in CPR as a function of oscillation magnitude. These correlations will be presented as a function of the important parameters which affect the response. It is expected that the primary parameter will be channel decay ratio, since this is a direct measure of the hydraulic responsiveness of the channel, which dominates the CPR response. The generic analyses will be performed with a range of fuel designs representing the range of expected fuel response for current fuel designs. For future core and fuel designs, parameters will be defined that must be confirmed to be within the defined ranges of the generic analysis. This may simply include the verification that the proposed fuel designs for the reload are within the defined range of acceptable designs.

For new fuel designs which have not been explicitly covered by the generic analysis, the CPR response must be evaluated. Details of the evaluation approach will be provided in a supplement to this licensing topical report.

Table 6-1

CALCULATION OF HARMONIC MODE EIGENVALUES - COMPARISON TO KNOWN ANALYTICAL SOLUTION

	Analytical Eigenvalue	BWR Simulator Eigenvalue
Fundamental Mode, λ_o	1.05471	1.05467
First Harmonic, λ_1	1.04302	1.04284
$\lambda_o-\lambda_1$	0.01169	0.01183
Fourth Harmonic, λ_4	1.03159	1.03115
$\lambda_o-\lambda_4$	0.02312	0.02352

Table 6-2

BWR/4 EXAMPLE - INITIAL CONDITIONS AND OSCILLATION CHARACTERISTICS

Core Thermal Power	50% of rated
Core Flow	30% of rated
Xenon Concentration	Full Power Equilibrium
Control Rod Pattern	All rods out
Cycle Exposure	8100 MWd/T
Radial Peaking Factor	1.37
Initial MCPR	1.84
Oscillation Mode	Regional NWSE Axis of Oscillation
Oscillation Contour	Figure 6-12
Oscillation Period	2.0 sec
Peak Bundle Oscillation Magnitude (P-M)/A	1.0
Axial Phase Lag (relative to A-level)	
B-Level	-23°
C-Level	-59°
D-Level	-82°

Table 6-3

BWR/4 EXAMPLE - RESULTS DURING OSCILLATIONS

Peak Bundle Oscillation Magnitude	110% of rated 161% of initial
Minimum CPR	1.35
Peak LPRM Signal	61% of scale 149% of initial
Peak APRM Signal	52.2% of rated
Peak OPRM Signal	133% of initial

Table 6-4

OSCILLATION CONTOURS

Core Size[a]	Cycle	Cycle[b] Exposure	Power/Flow (%/%)	Xenon[c]	Oscillation Axis
368	10	BOC	46/30	FP	NWSE/NESW
		MOC	46/30	FP	NS/EW
		MOC	47/30	FP	NWSE/NESW
		EOC	47/30	FP	NWSE/NESW
484	12	BOC	48/30	FP	NS/EW
		MOC	49/30	FP	NWSE/NESW
		EOC	53/30	FP	NWSE/NESW
560	12	BOC	50/30	FP	NWSE/NESW
		EOC	50/30	FP	NWSE/NESW
	9	MOC	50/30	FP	NWSE/NESW
	10	BOC	50/30	FP	NWSE/NESW
		MOC	50/30	FP	NWSE/NESW
		EOC	50/30	FP	NWSE/NESW
	10	BOC	50/30	FP	NWSE/NESW
		MOC	50/30	FP	NS/EW
		EOC	50/30	FP	NWSE/NESW
648	1	BOC	46/29	SS	NWSE/NESW
		BOC	56/31	SS	NS/EW
748	2	BOC	32/35	NO	NS/EW
		BOC	48/50	NO	NS/EW
		MOC	39/34	NO	NWSE/NESW
		MOC	50/30	NO	NWSE/NESW
		MOC	76/57	NO	NS/EW
764	2	BOC	45/28	FP	NWSE/NESW
	8	BOC	50/30	SS	NWSE/NESW
		MOC	50/30	SS	NWSE/NESW
		EOC	50/30	SS	NWSE/NESW

[a] Number of fuel bundles

[b] BOC = beginning of cycle
MOC = middle of cycle
EOC = end of cycle

[c] FP = full power equilibrium xenon
SS = equilibrium xenon at power/flow
NO = no xenon

Table 6-5

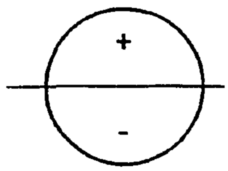
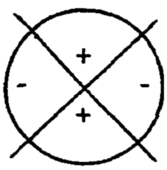
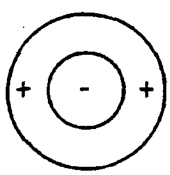
560 BUNDLE EXAMPLE - MONTE CARLO SIMULATION INPUTS

Initial Conditions	100% power/87% flow 62% power/40% flow	(95% probability) (5% probability)
Initial MCPR		
100% Power	1.25	Tech Spec limit
40% Flow	1.44	Tech Spec limit
CPR Change During Flow Reduction, DIDW (equation 6-6)	0.0065 0.0006	Average Standard Deviation
Radial Peaking Factor	1.415 0.083	Average Standard Deviation
Oscillation Contours	560 Bundle BOC/EOC-12 MOC-9	Table 6-4
Oscillation Growth Rates	22 scenarios	Growth rates of 1.05 to 1.60
Trip System Design	High-low-high Detection Algorithm (Appendix B)	Setpoints S1 = 1.10 S2 = 0.92 DR3 = 1.30 Smax = 1.20
Overshoot Distribution	Figure 6-27	
Average LPRM Failure Rate	6%, 8%, 9%	BOC, MOC, EOC
MCPR Performance	Curve 1	Figure 6-11

Table 6-6

560 BUNDLE EXAMPLE - MONTE CARLO SIMULATION RESULTS

Number of Cases	5000
MCPR	
Mean ($\bar{x}$)	1.404
Standard Deviation (σ)	0.073
Tolerance Factor (k) (22 independent cases)	2.350
95/95	1.232

OSCILLATION MODE	EIGENVALUE SEPARATION $(k_n - k_o)$
	$-1.2\ \$$
	$-2.5\ \$$
	$-4.1\ \$$

k_o = fundamental mode eigenvalue

k_n = harmonic mode eigenvalue

FIGURE 6-1. EIGENVALUE SEPARATION OF HARMONIC MODES

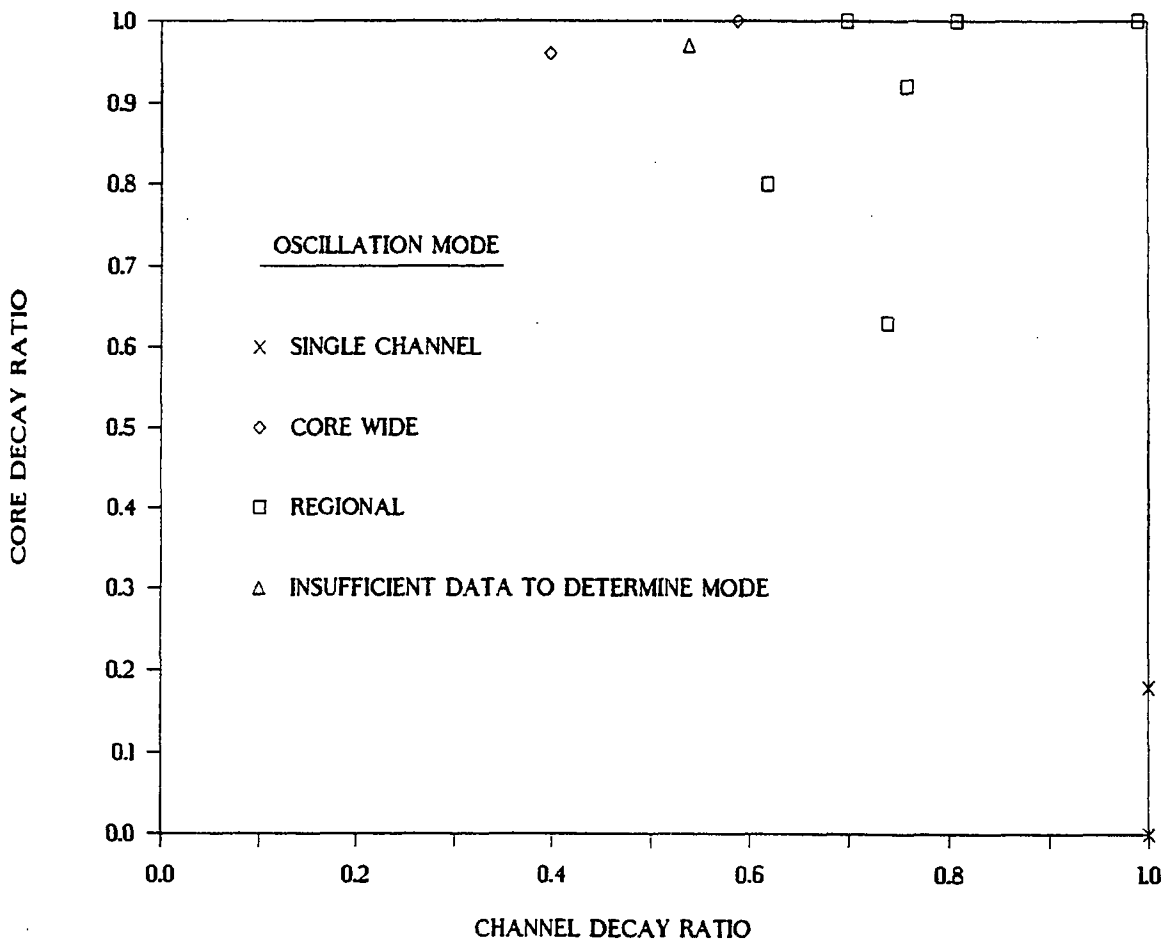


FIGURE 6-2. GE BWR STABILITY EXPERIENCE

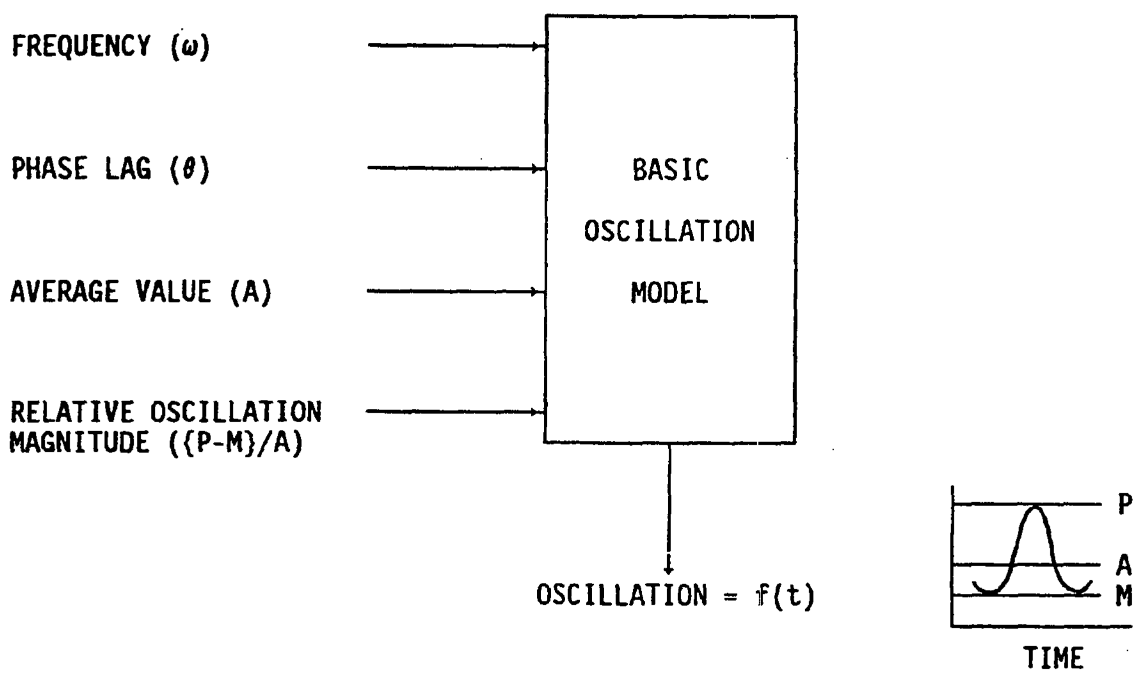


FIGURE 6-3. BASIC OSCILLATION MODEL

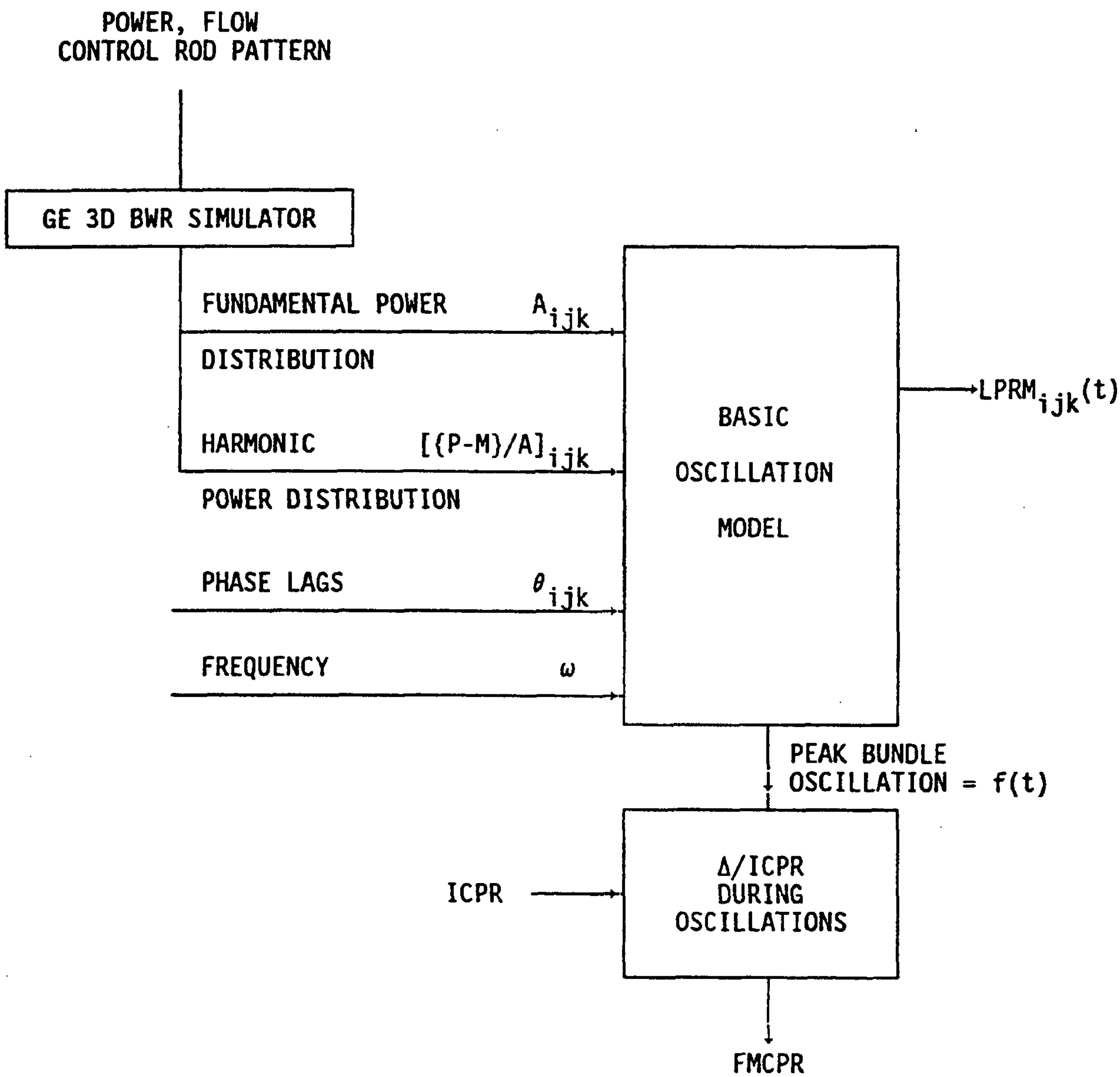


FIGURE 6-4. OSCILLATION METHODOLOGY BLOCK DIAGRAM

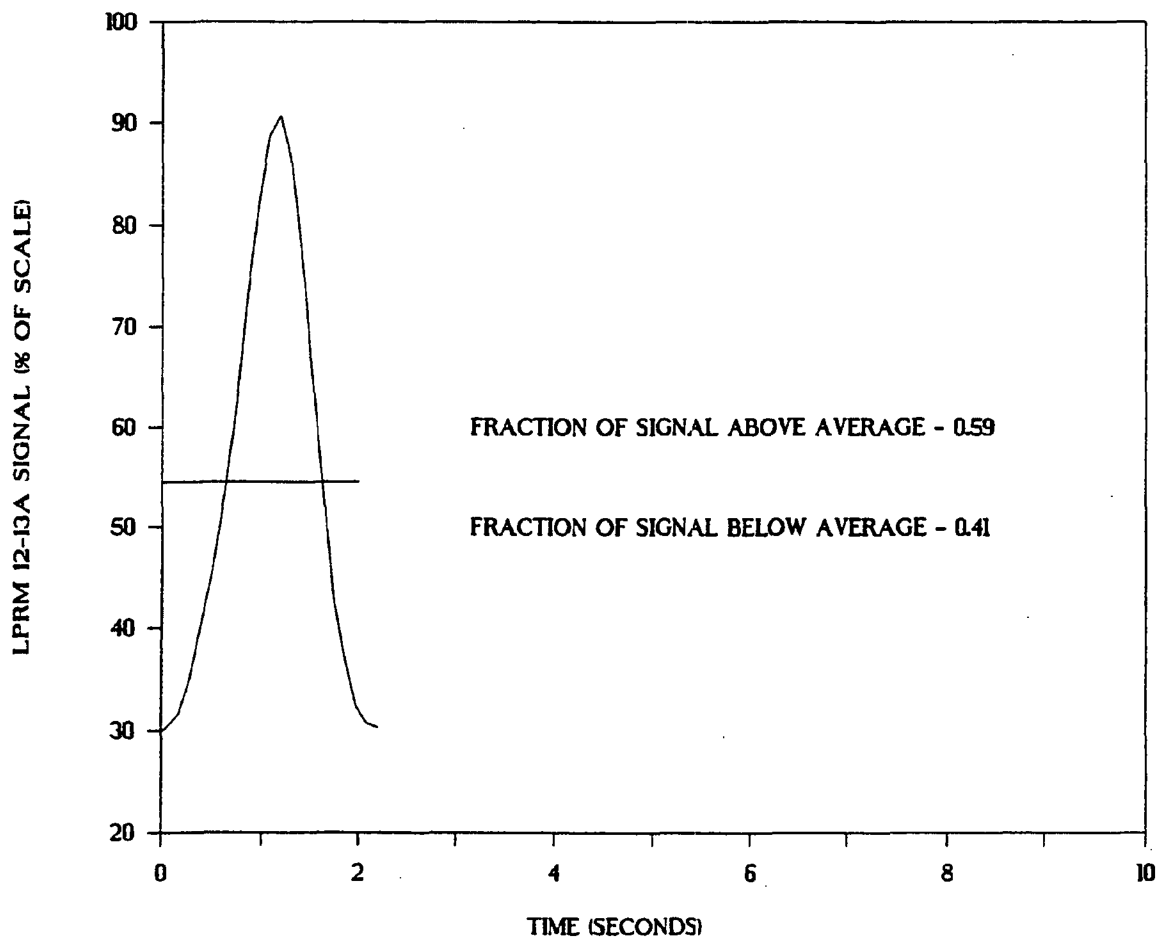


FIGURE 6-5. NON-LINEAR OSCILLATION AS MEASURED BY AN LPRM

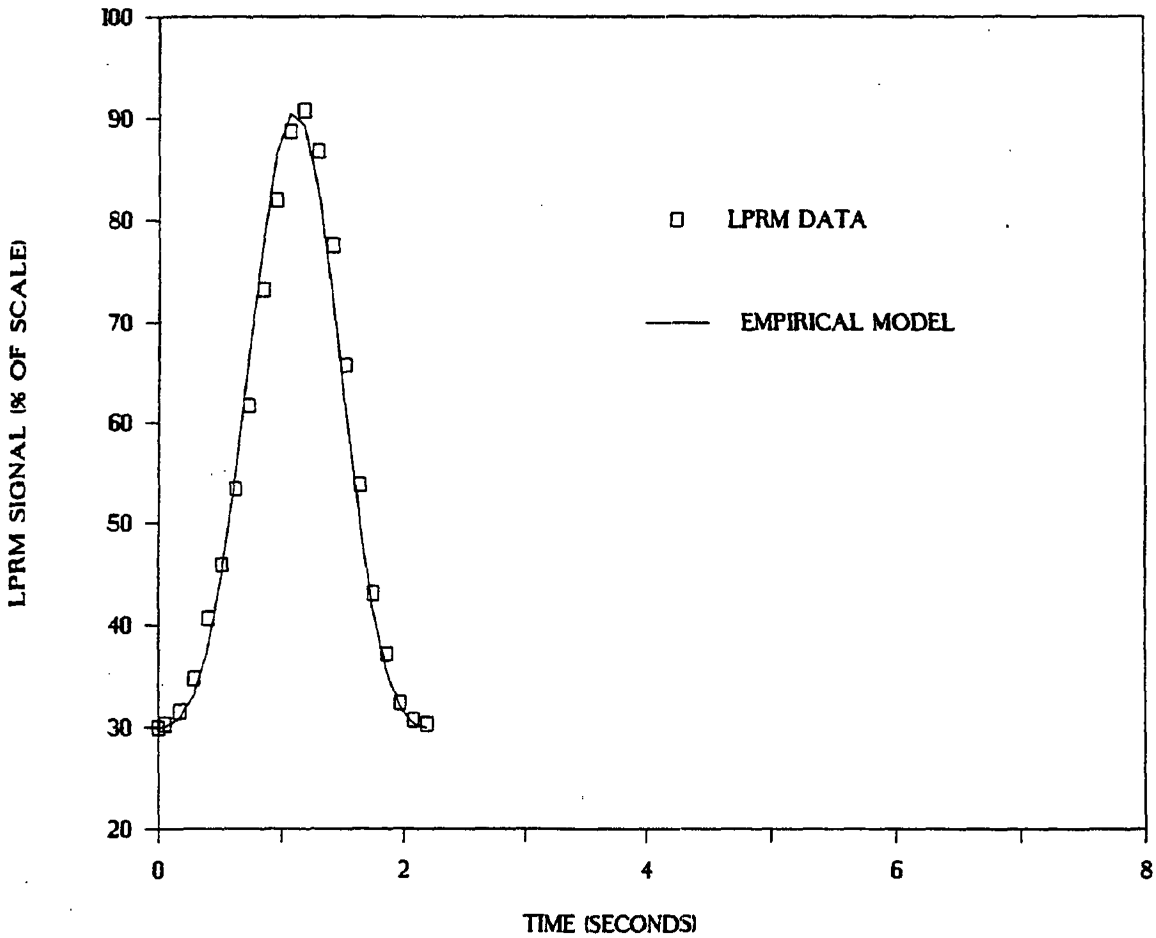


FIGURE 6-6. BASIC OSCILLATION MODEL - COMPARISON TO PLANT DATA

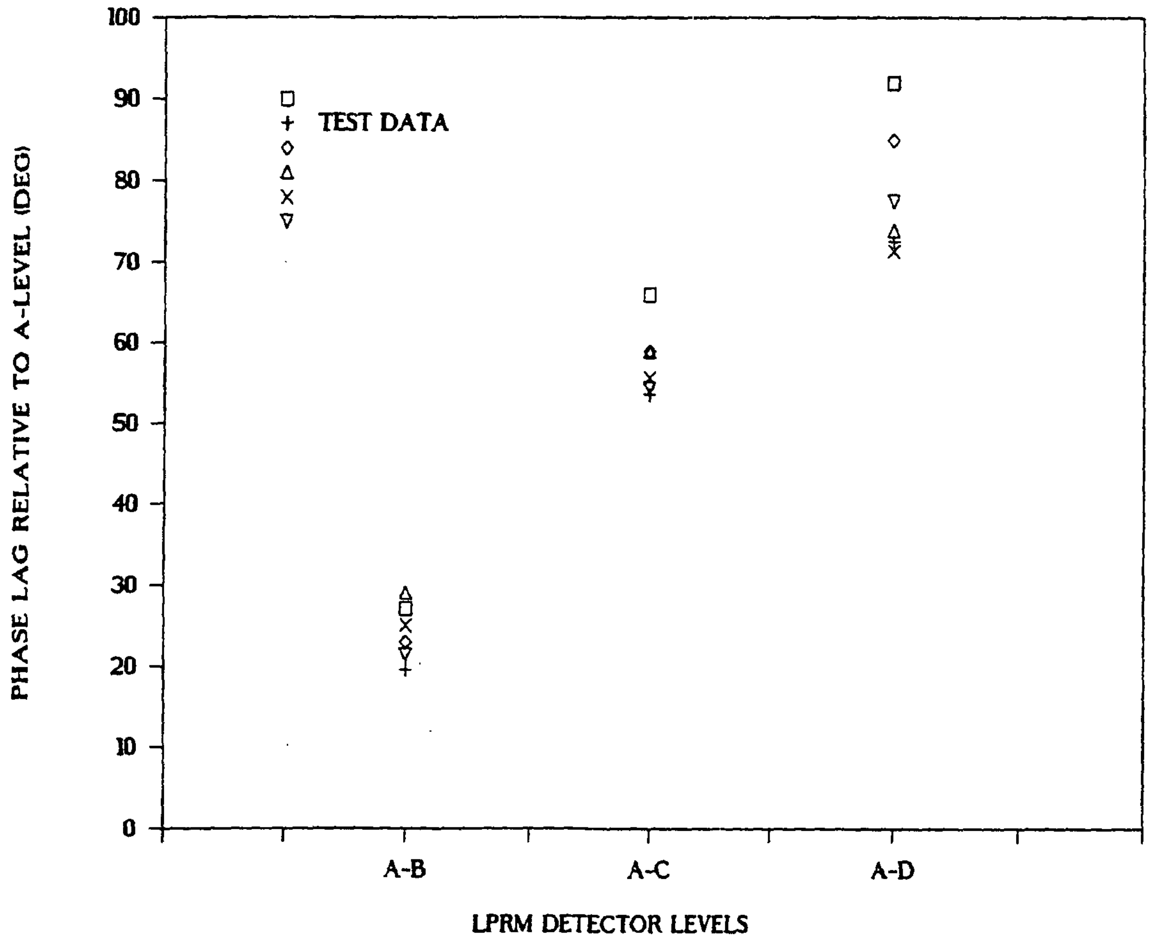


FIGURE 6-7. AXIAL PHASE LAGS

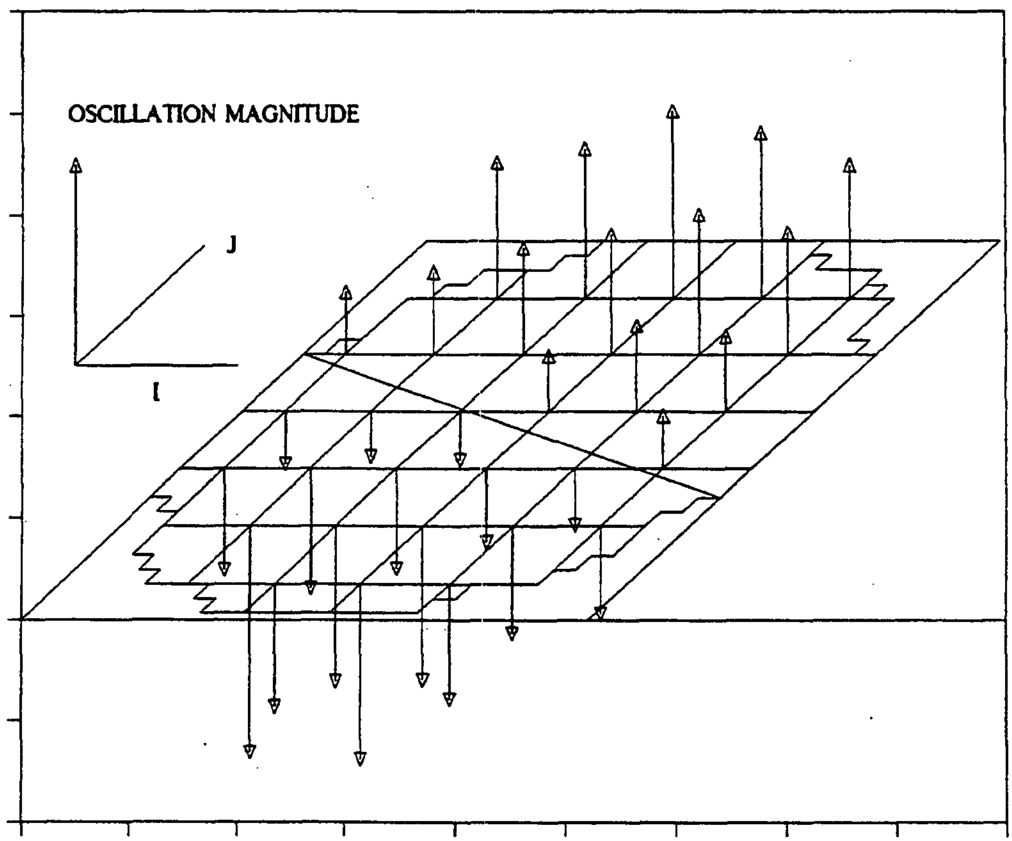


FIGURE 6-8. CAORSO CYCLE 2 TEST OSCILLATION CONTOUR

j \ i	08	16	24	32	40	48	56
57			0.27	0.44	0.48		
49	0.05	0.10	0.31	0.57	0.78	0.76	
41	0.22	0.09	0.14	0.38	0.60	0.79	0.48
33	0.46	0.36	0.11	0.15	0.39	0.58	0.44
25	0.66	0.55	0.35	0.11	0.14	0.31	0.27
17	0.85	0.77	0.55	0.36	0.09	0.10	
09		0.85	0.66	0.47	0.23	0.05	

0.XX = (P-M)/A OSCILLATION MAGNITUDE FOR LPRM(i,j) RELATIVE TO
(P-M)/A OSCILLATION MAGNITUDE FOR PEAK BUNDLE

FIGURE 6-9. PREDICTED LPRM OSCILLATION CONTOUR

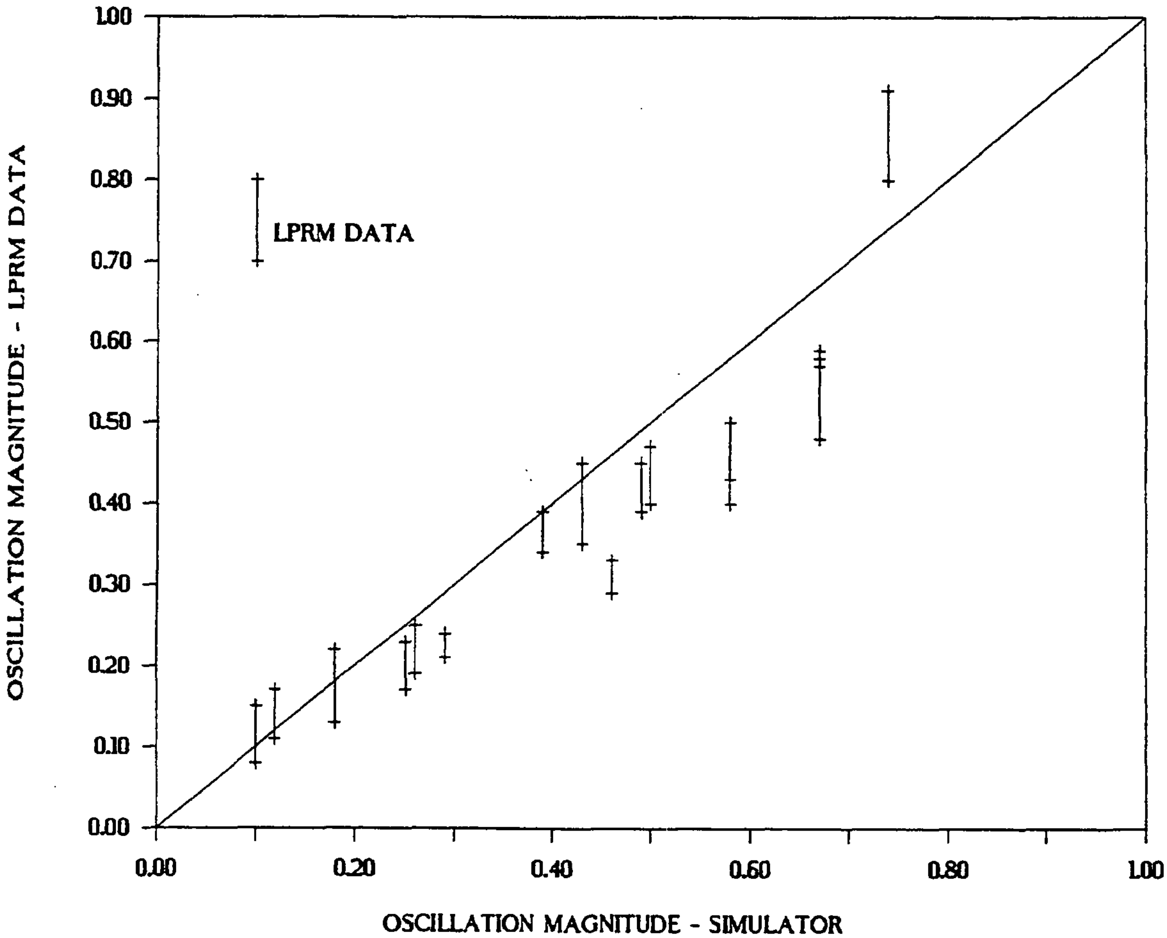


FIGURE 6-10. COMPARISON OF OSCILLATION CONTOUR - TEST DATA VERSUS GE 3D SIMULATOR PREDICTIONS

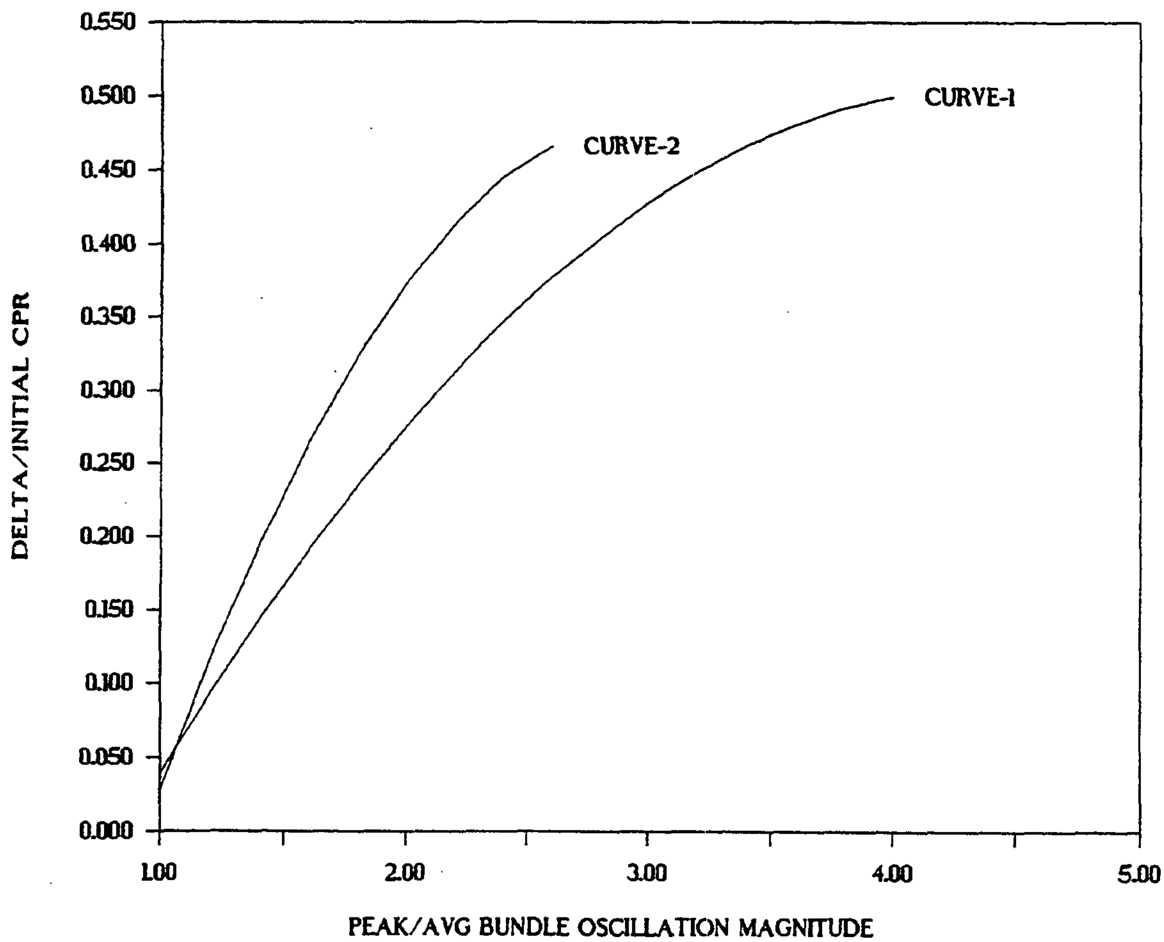


FIGURE 6-11. TYPICAL MCPR PERFORMANCE DURING OSCILLATIONS

i						
45		0.09	0.41	0.72	0.86	0.45
37	0.20	0.15	0.19	0.55	0.78	0.86
29	0.45	0.48	0.20	0.21	0.55	0.71
21	0.60	0.82	0.57	0.20	0.19	0.40
13		0.92	0.83	0.48	0.15	0.09
05			0.60	0.44	0.20	
i	04	12	20	28	36	44

0.xx = (P-M)/A oscillation magnitude for LPRM(i,j) relative to
(P-M)/A oscillation magnitude for peak bundle

FIGURE 6-12. BWR/4 EXAMPLE CONTOUR

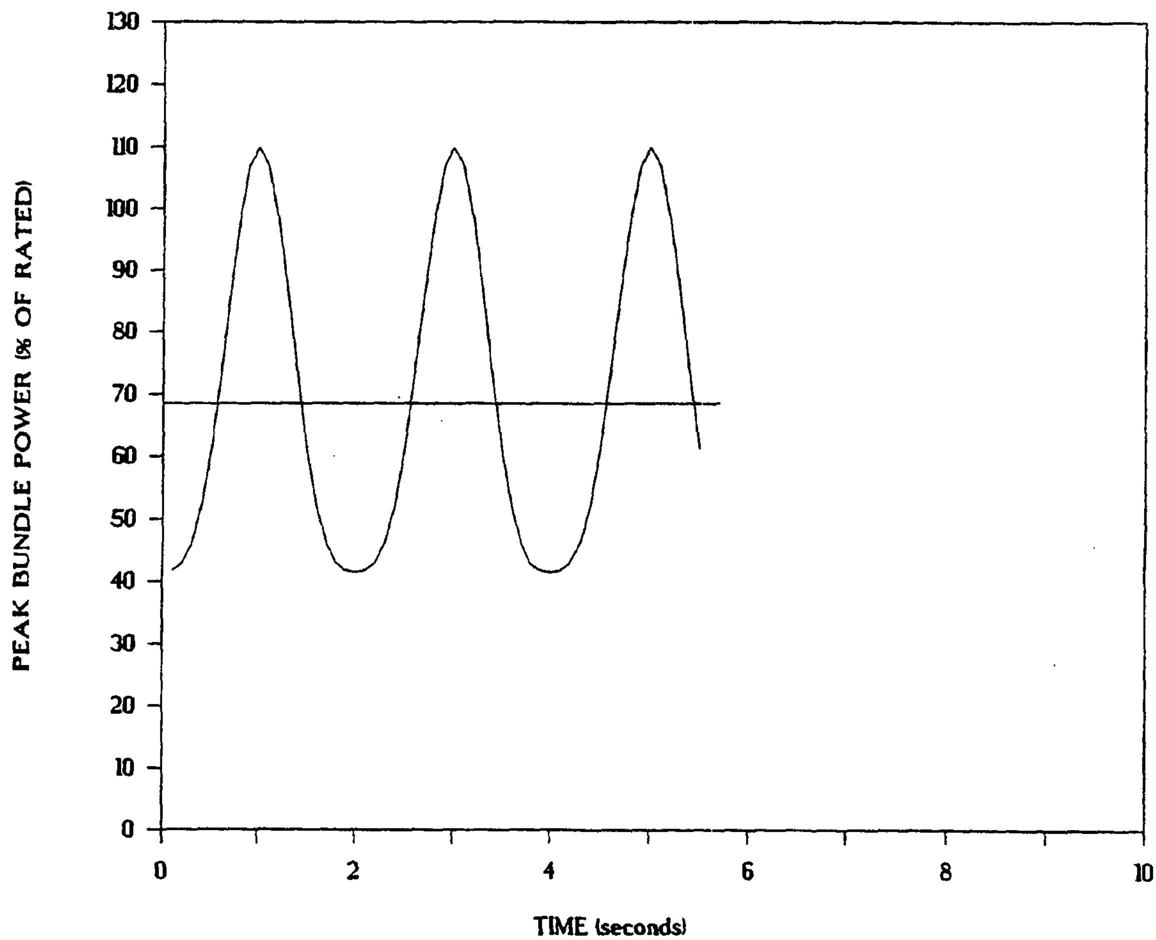


FIGURE 6-13. BWR/4 EXAMPLE - PEAK BUNDLE OSCILLATION MAGNITUDE

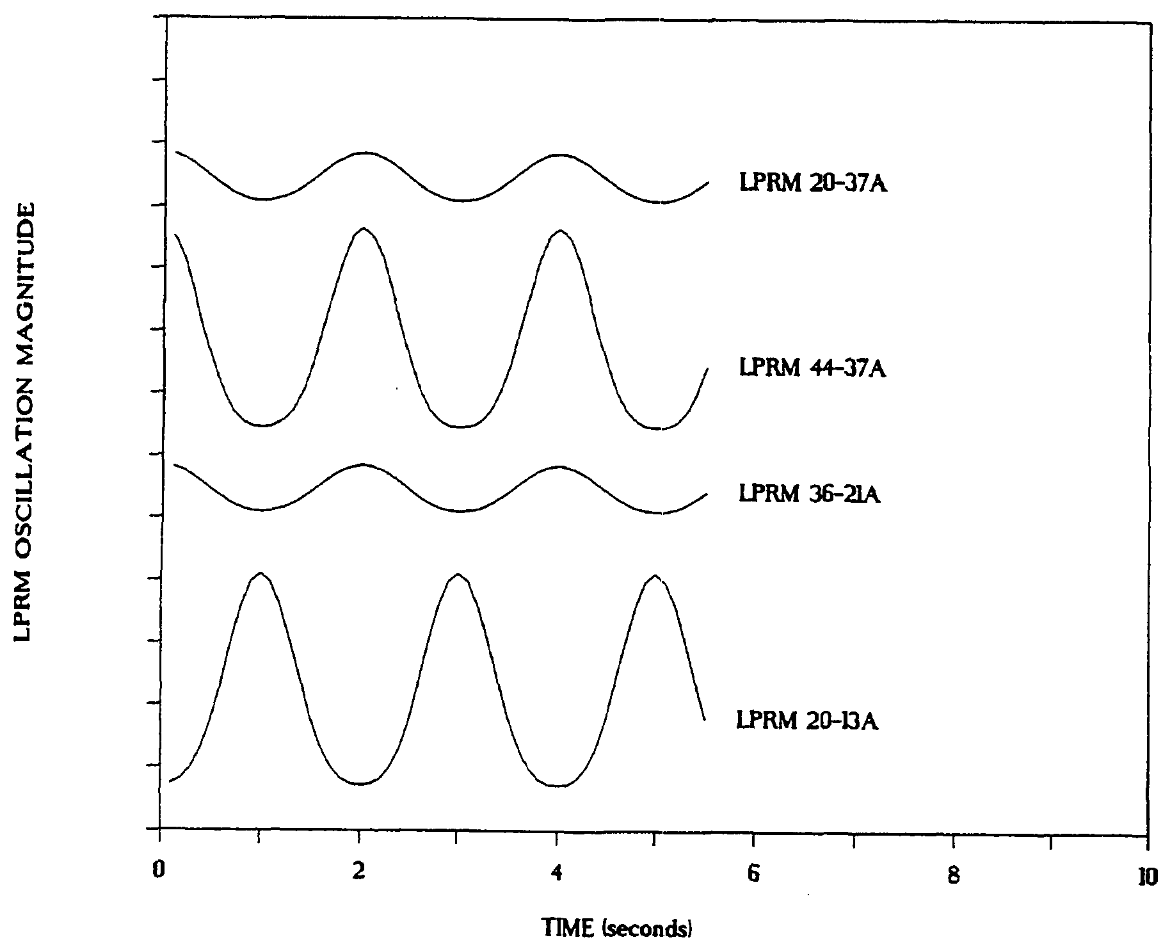


FIGURE 6-14. BWR/4 EXAMPLE - LPRM OSCILLATIONS (RADIAL DISTRIBUTION)

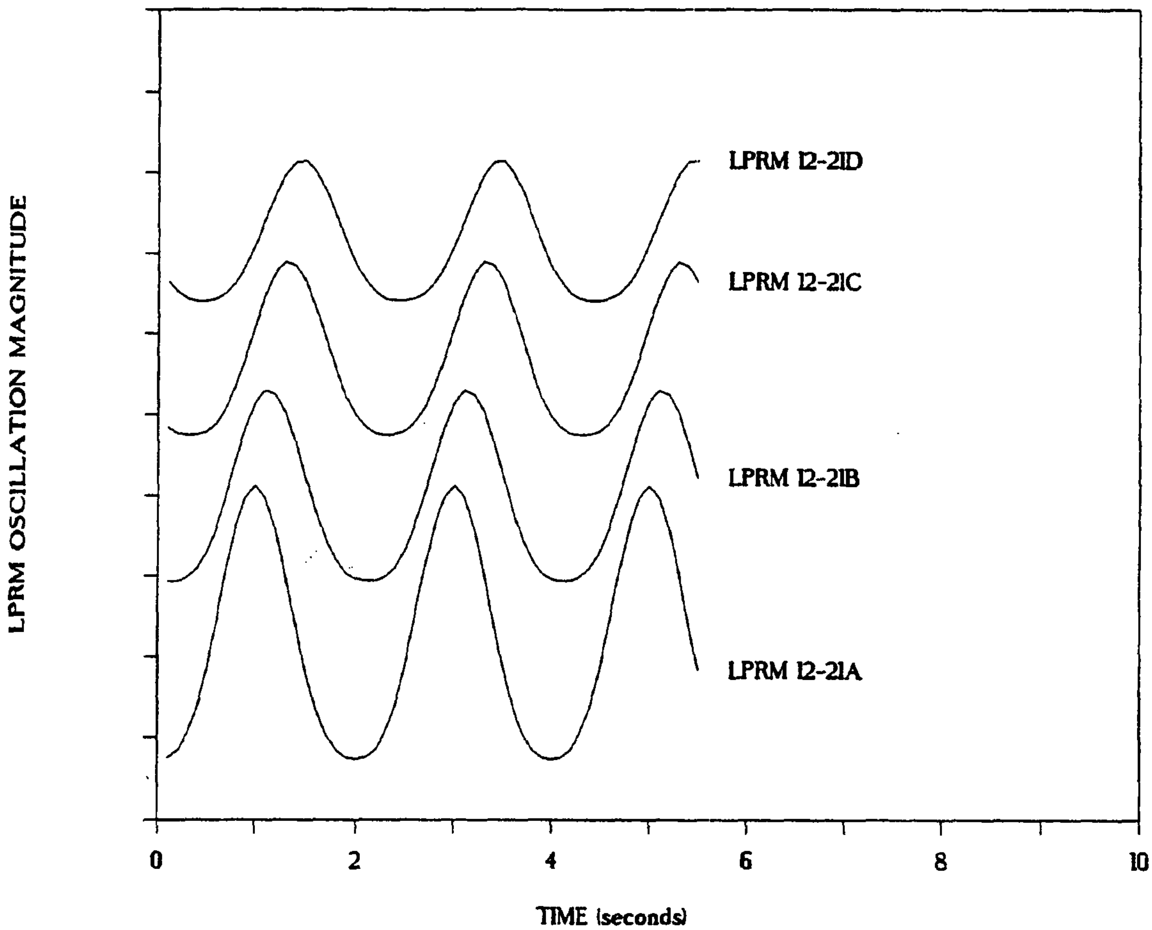


FIGURE 6-15. BWR/4 EXAMPLE - LPRM OSCILLATIONS (AXIAL DISTRIBUTION)

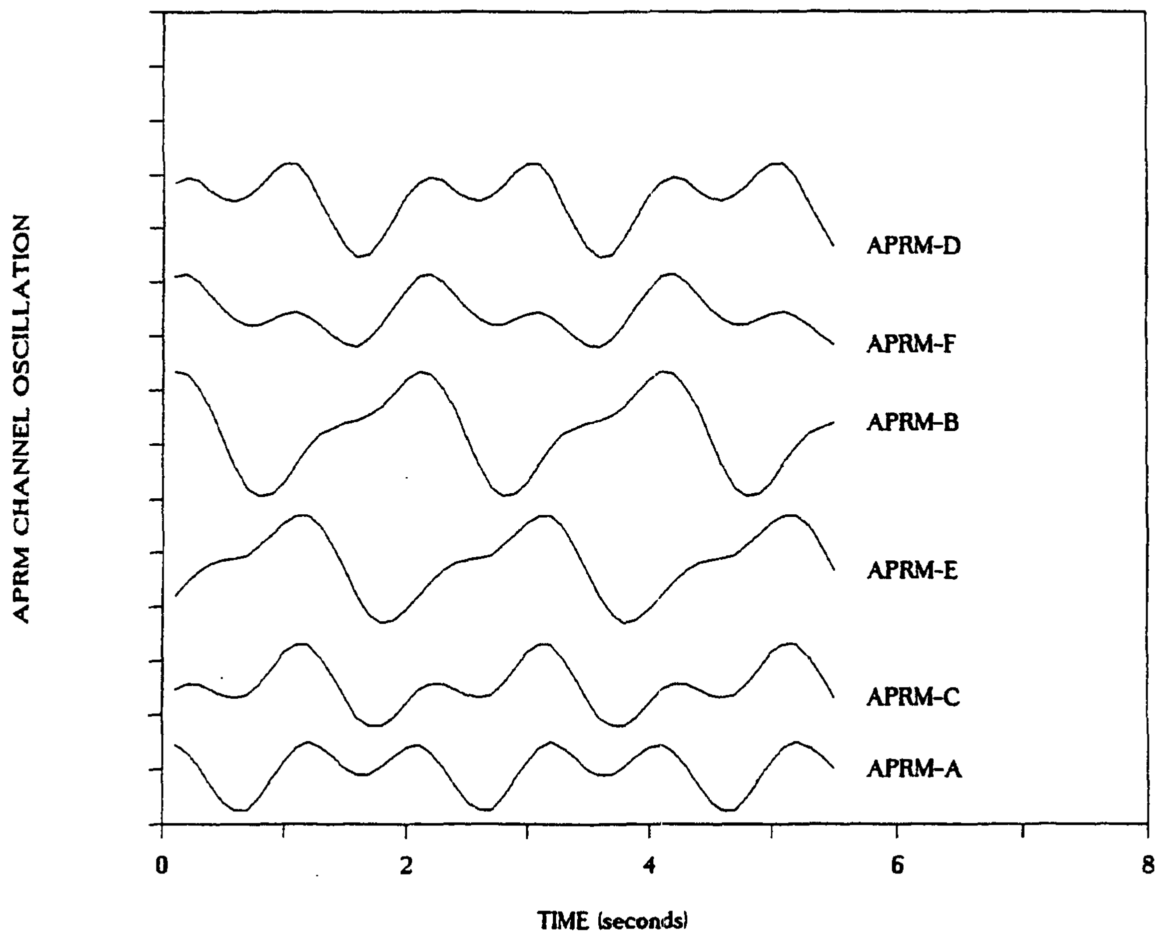


FIGURE 6-16. BWR/4 EXAMPLE - APRM OSCILLATIONS

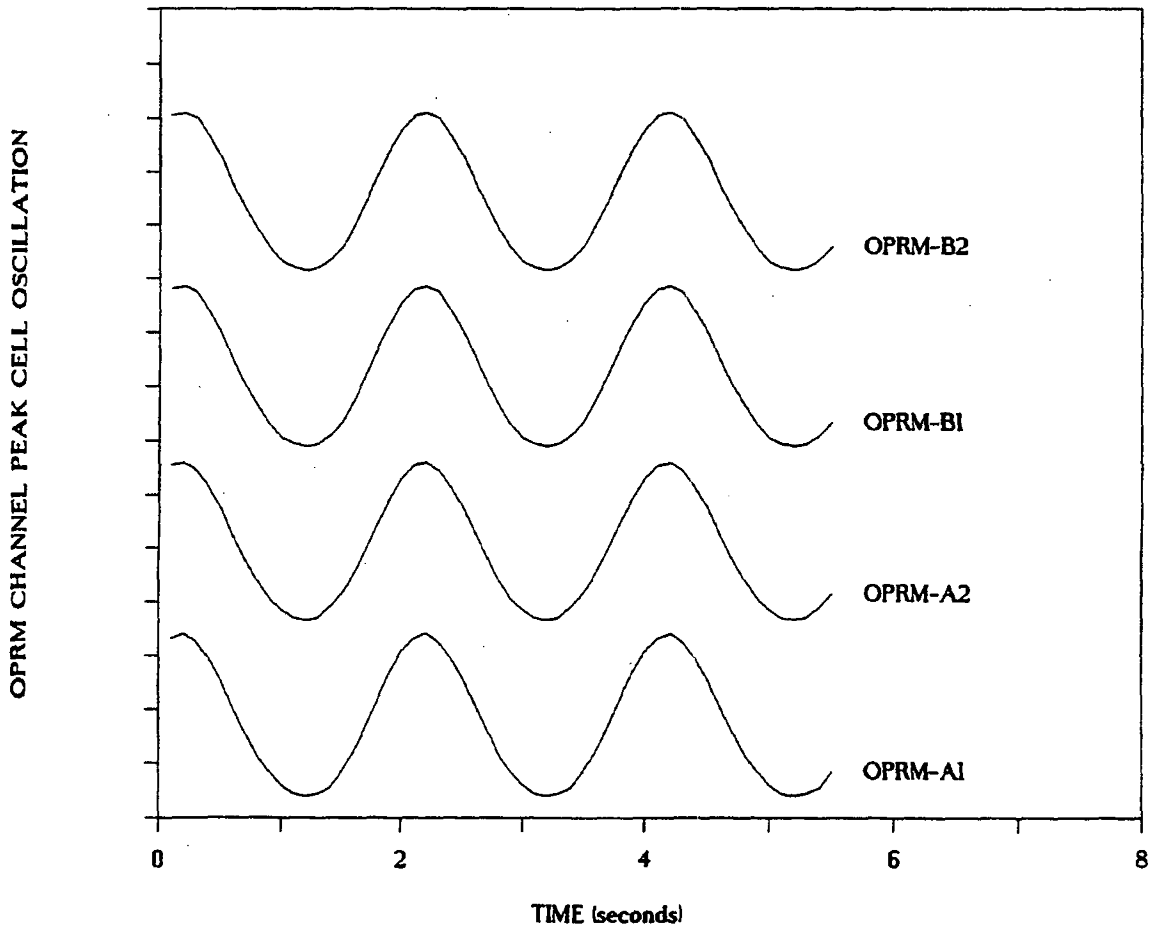


FIGURE 6-17. BWR/4 EXAMPLE - OPRM OSCILLATIONS

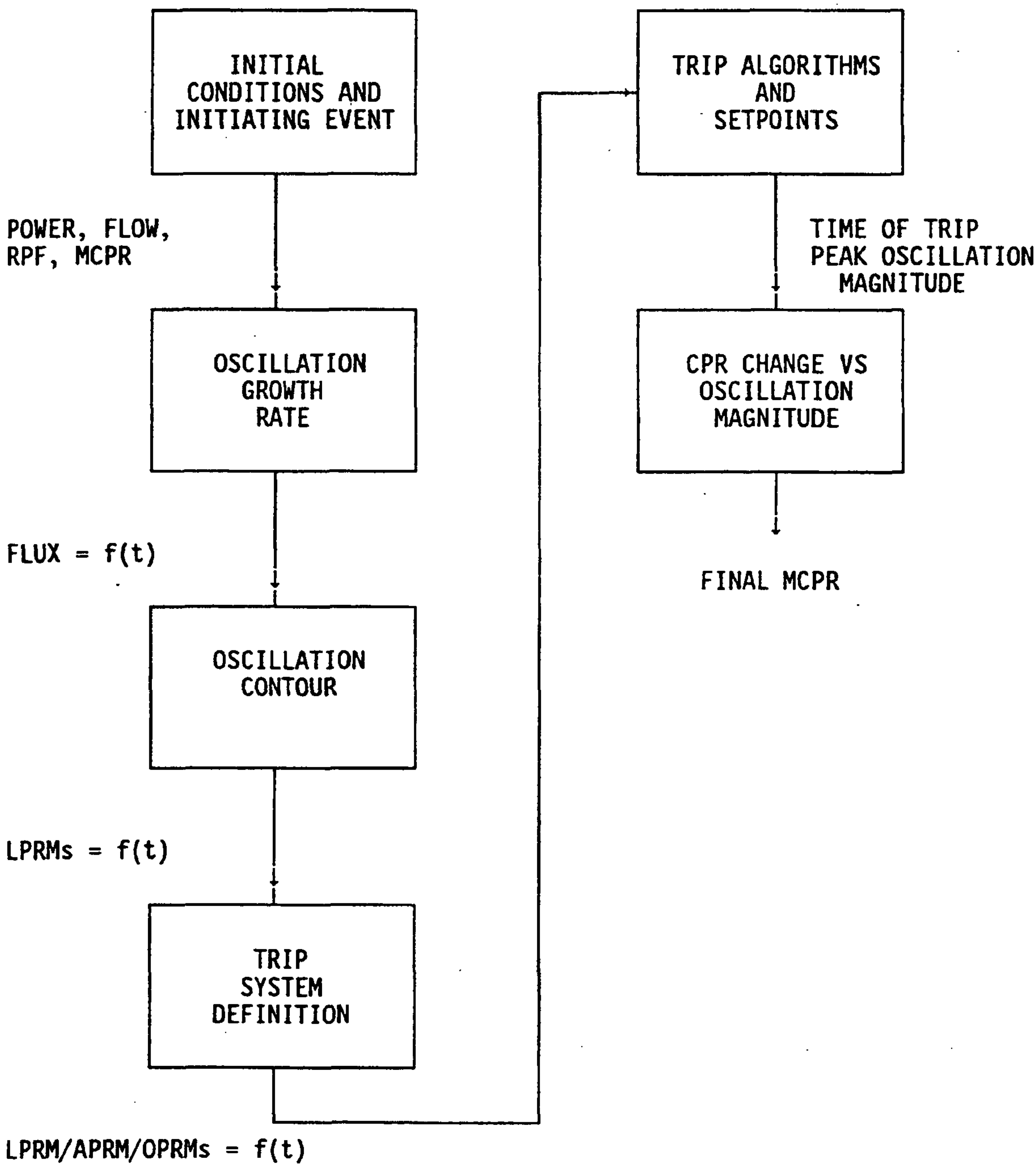


FIGURE 6-18. OSCILLATION METHODOLOGY ROADMAP

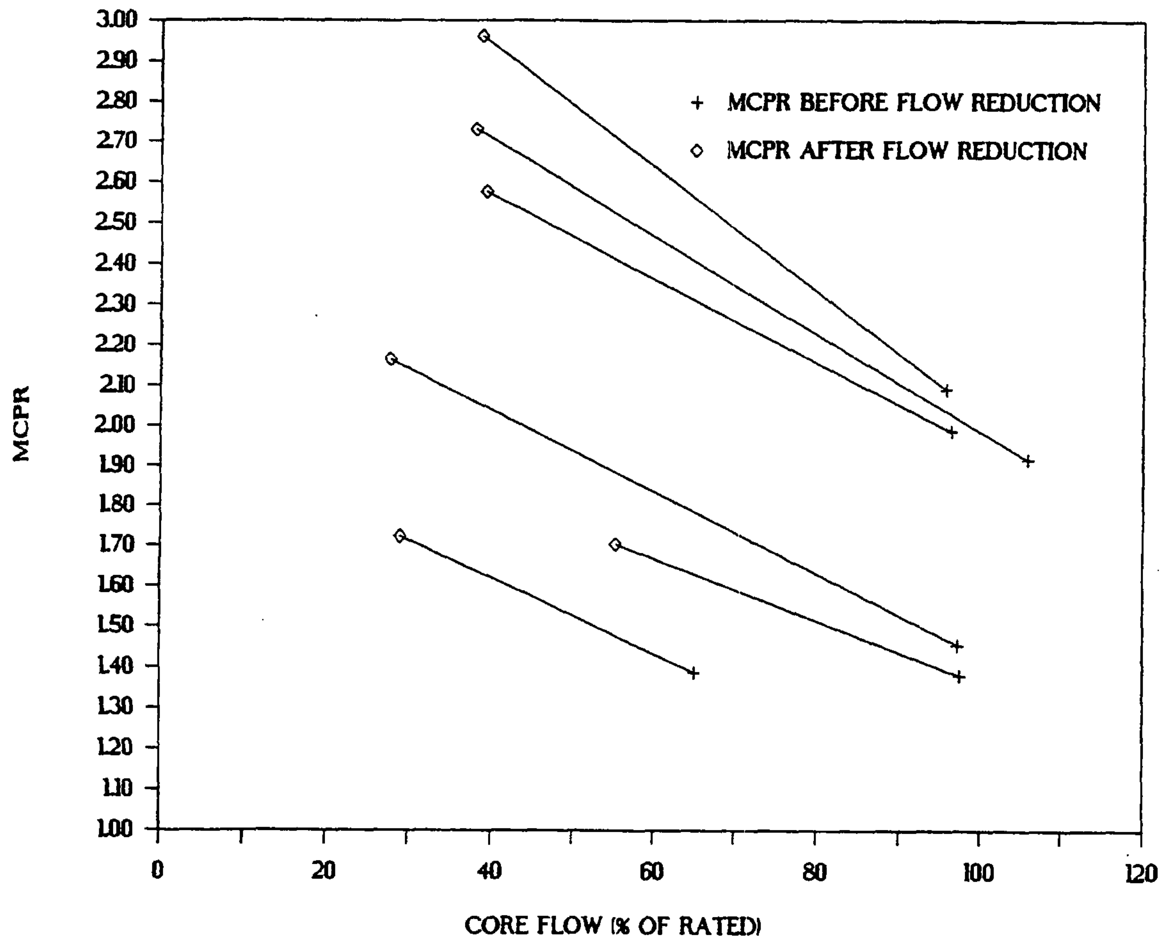


FIGURE 6-19. MCPR PERFORMANCE DURING FLOW REDUCTIONS - PROCESS COMPUTER DATA

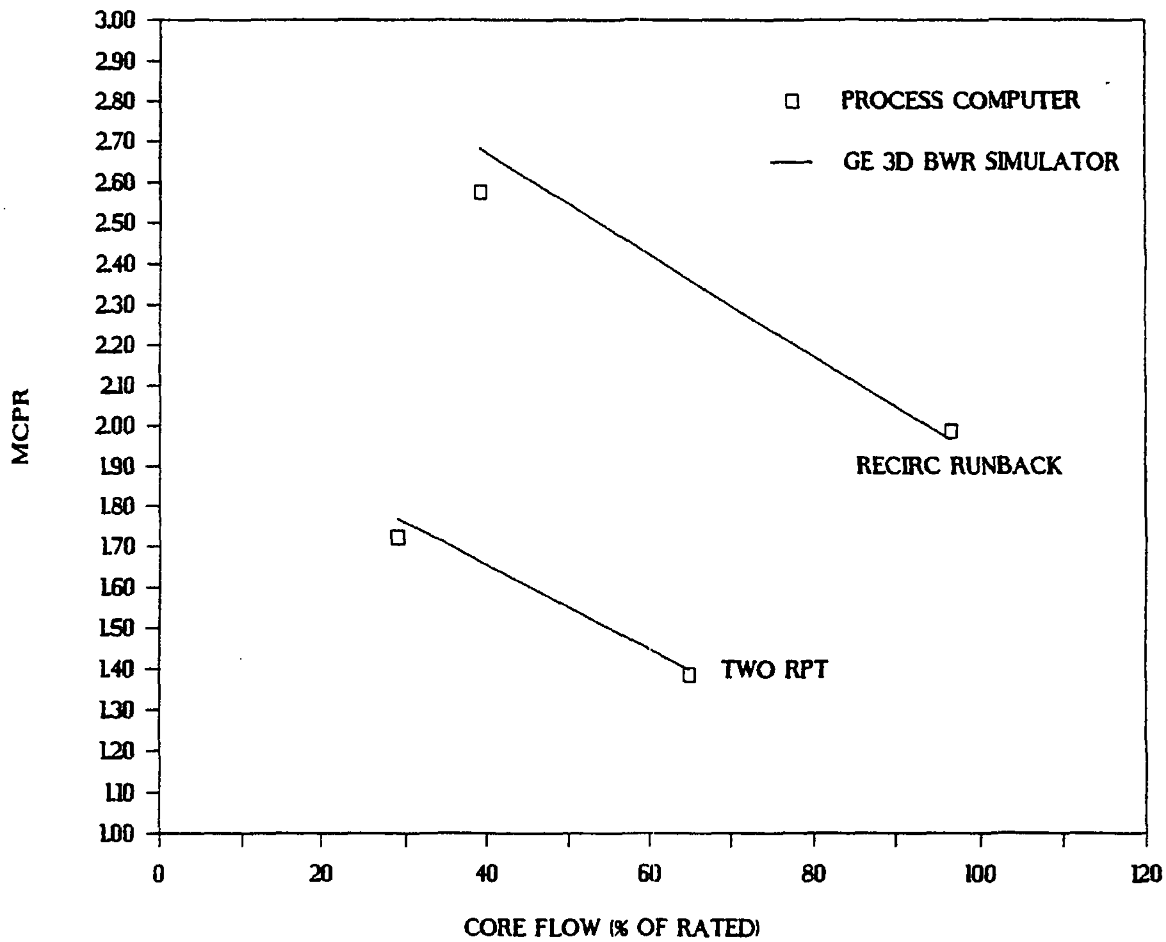


FIGURE 6-20. MCPR PERFORMANCE DURING FLOW REDUCTION - PROCESS COMPUTER VERSUS GE 3D BWR SIMULATOR RESULTS

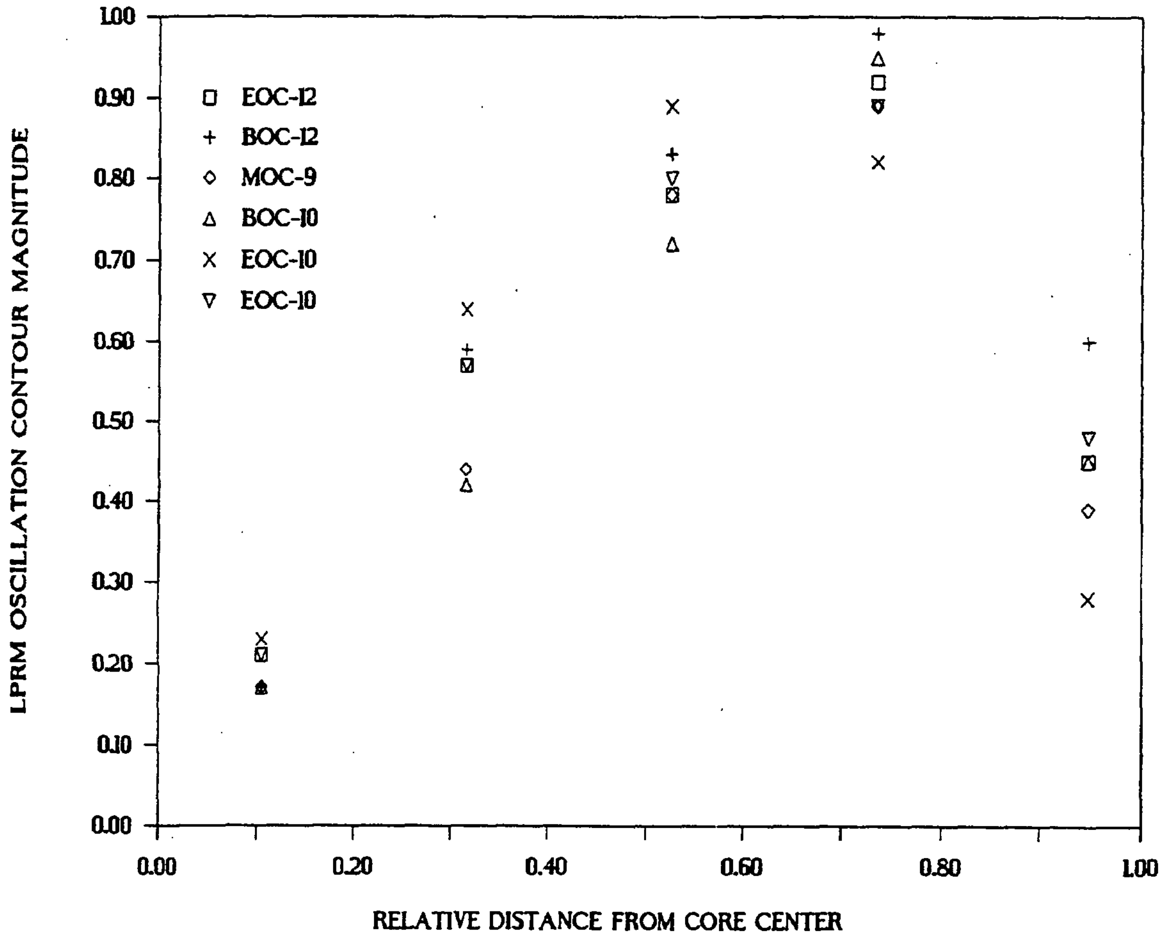


FIGURE 6-21. VARIATION IN 560 BUNDLE CORE OSCILLATION CONTOURS

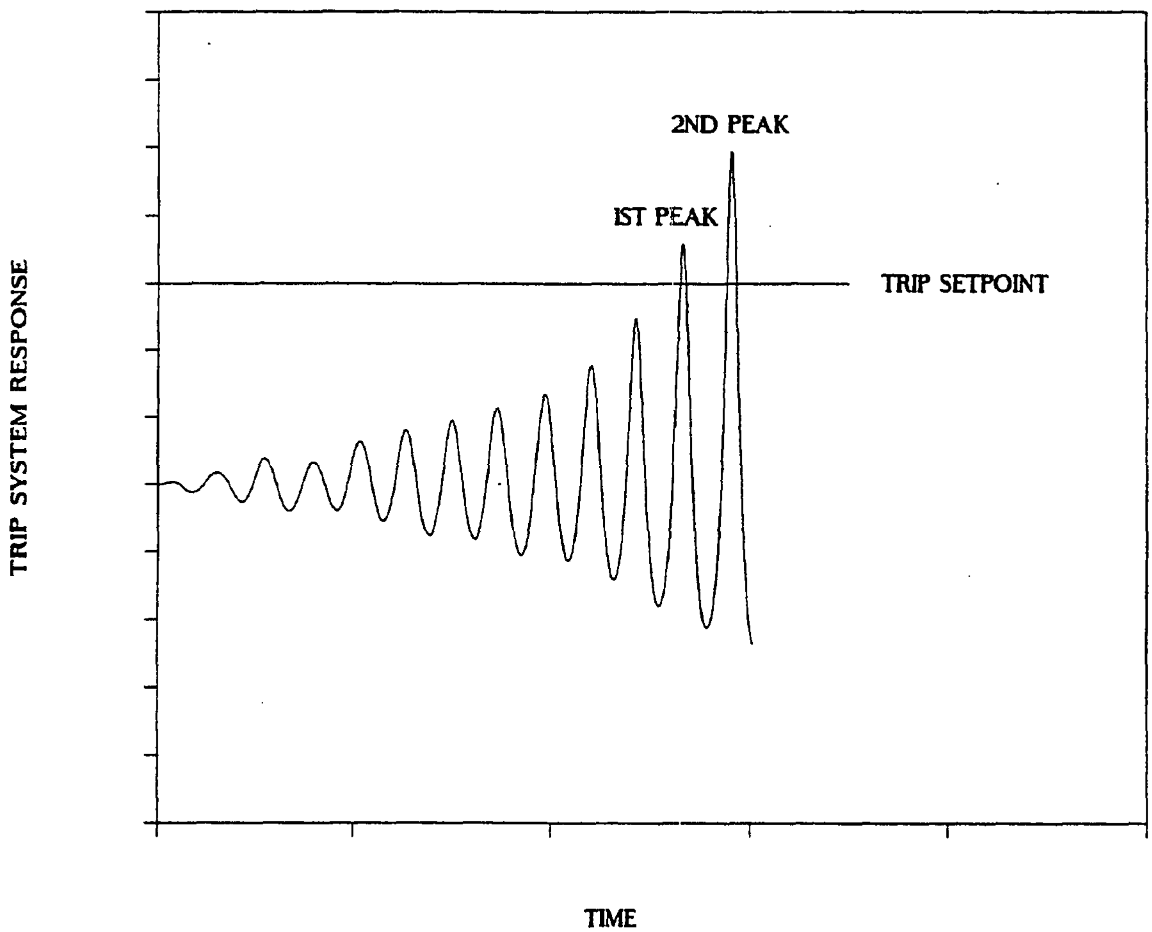


FIGURE 6-22. TRIP SYSTEM SETPOINT OVERSHOOT DURING OSCILLATIONS

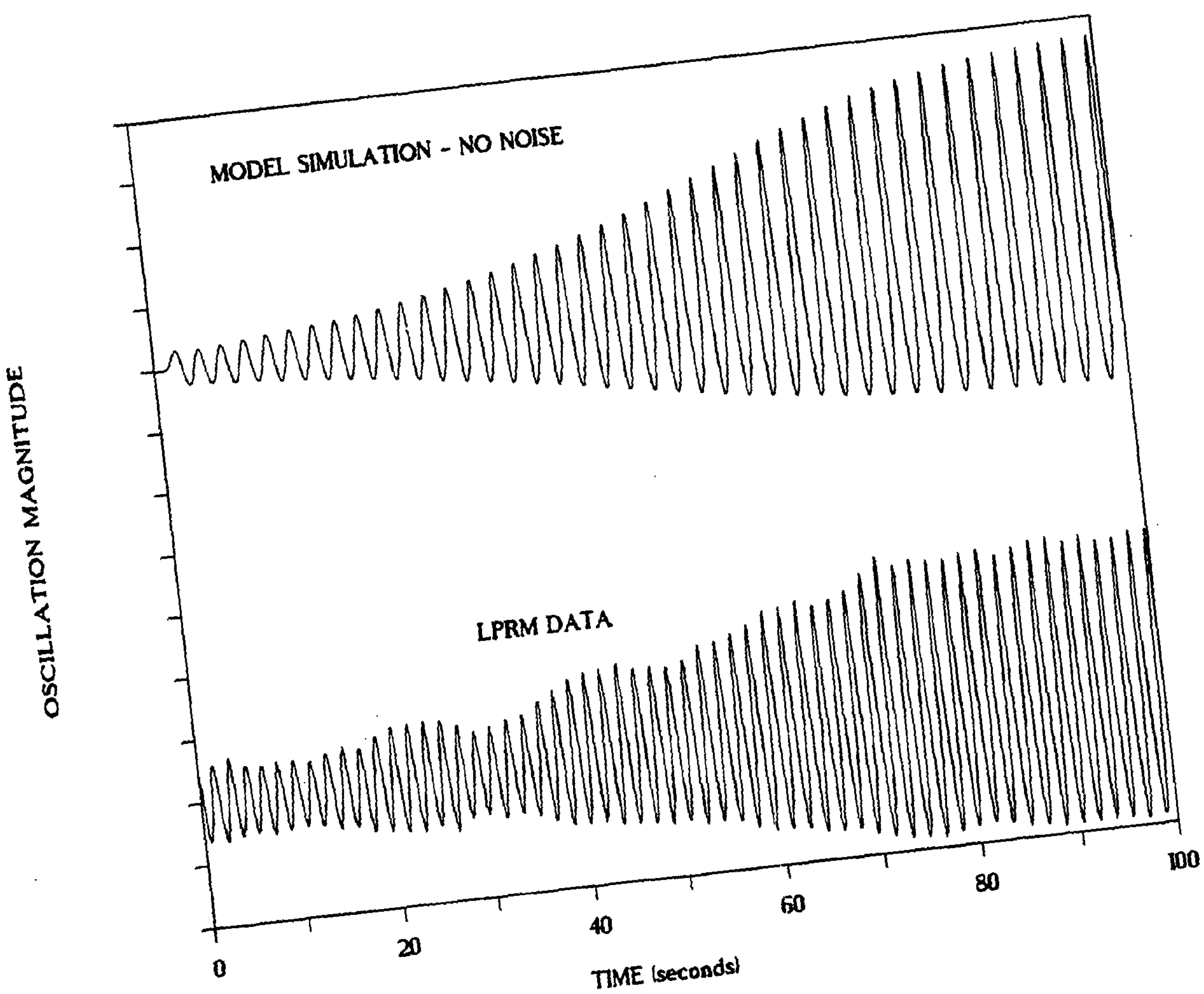


FIGURE 6-23. OSCILLATION GROWTH RATE - PLANT DATA VERSUS SIMULATED OSCILLATIONS WITHOUT NOISE

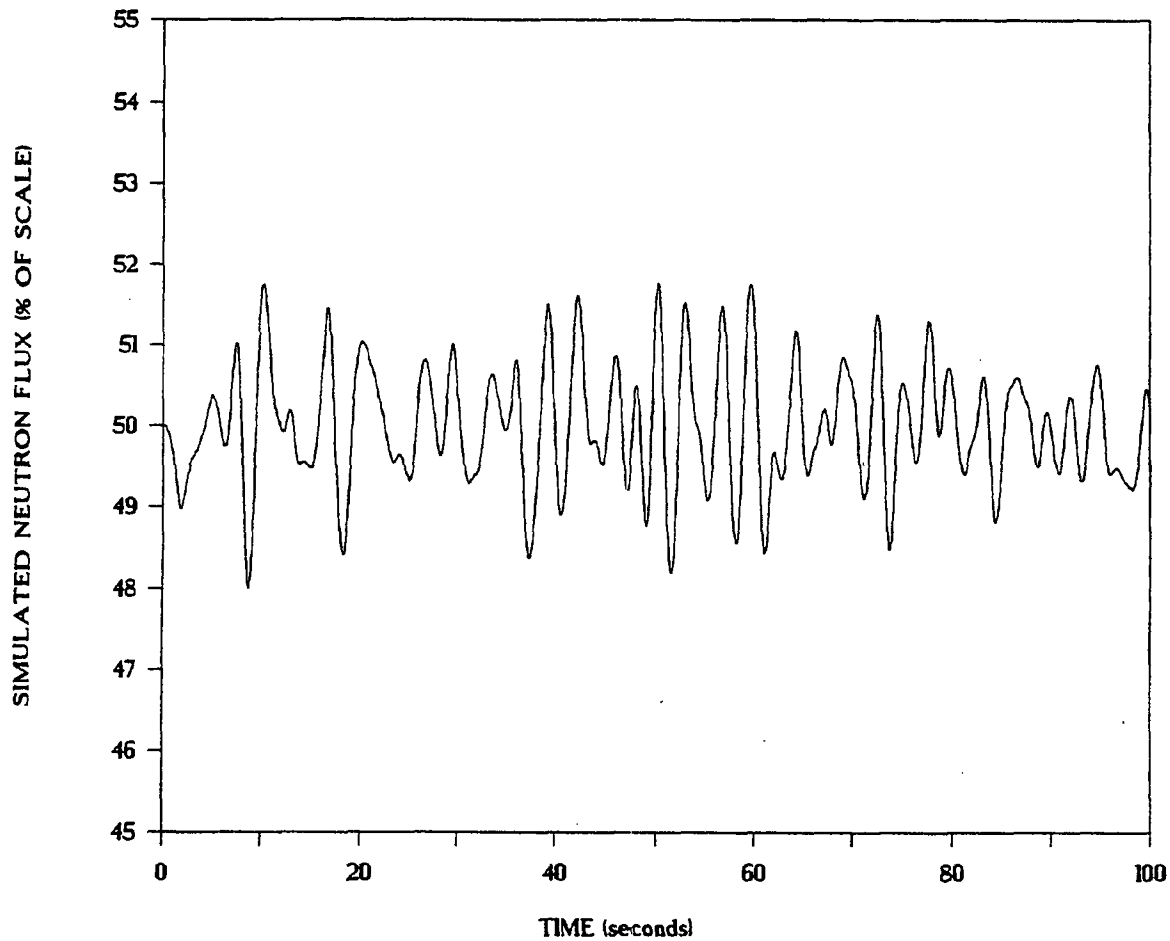


FIGURE 6-24. SIMULATED NOISE DATA - DECAY RATIO = 0.48

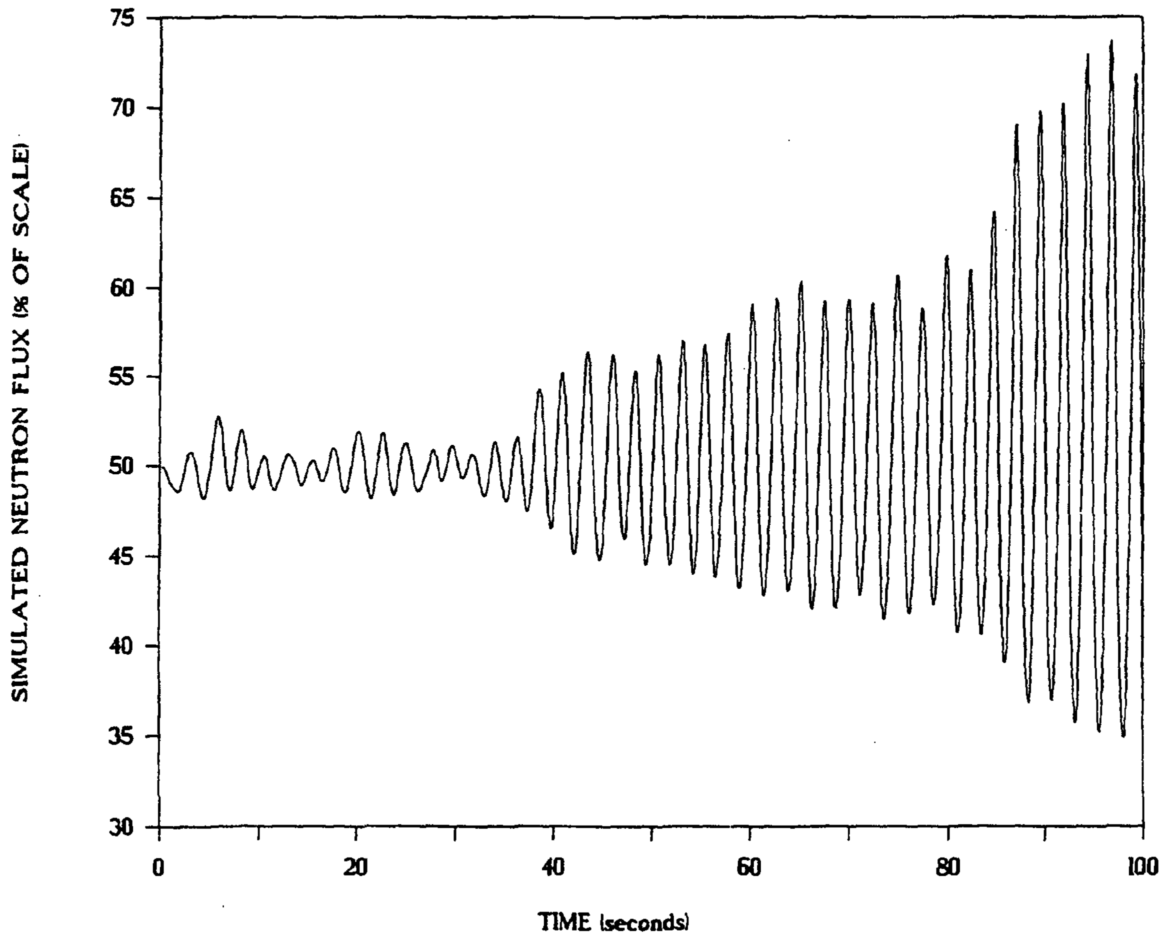


FIGURE 6-25. SIMULATED OSCILLATION SCENARIO - GROWTH RATE = 1.05

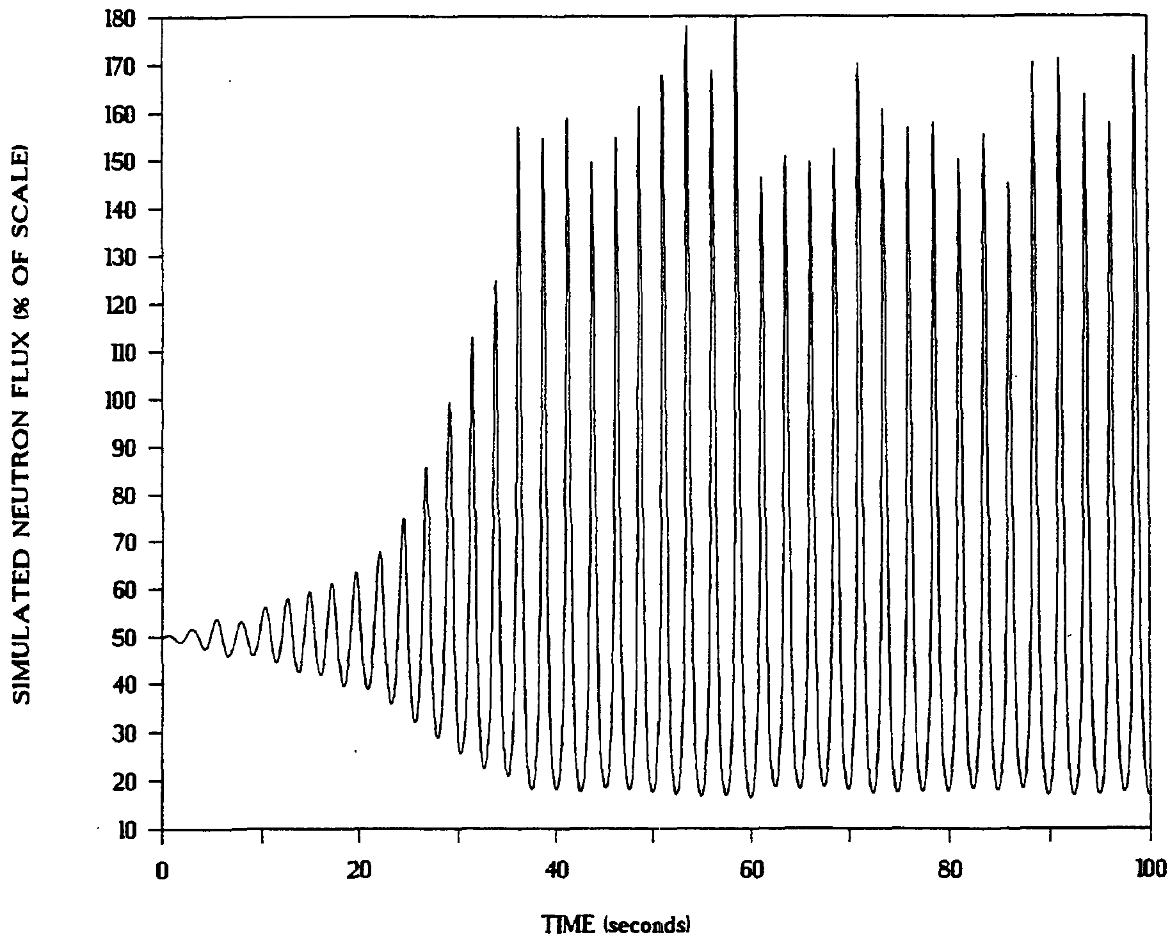


FIGURE 6-26. SIMULATED OSCILLATION SCENARIO - GROWTH RATE = 1.30

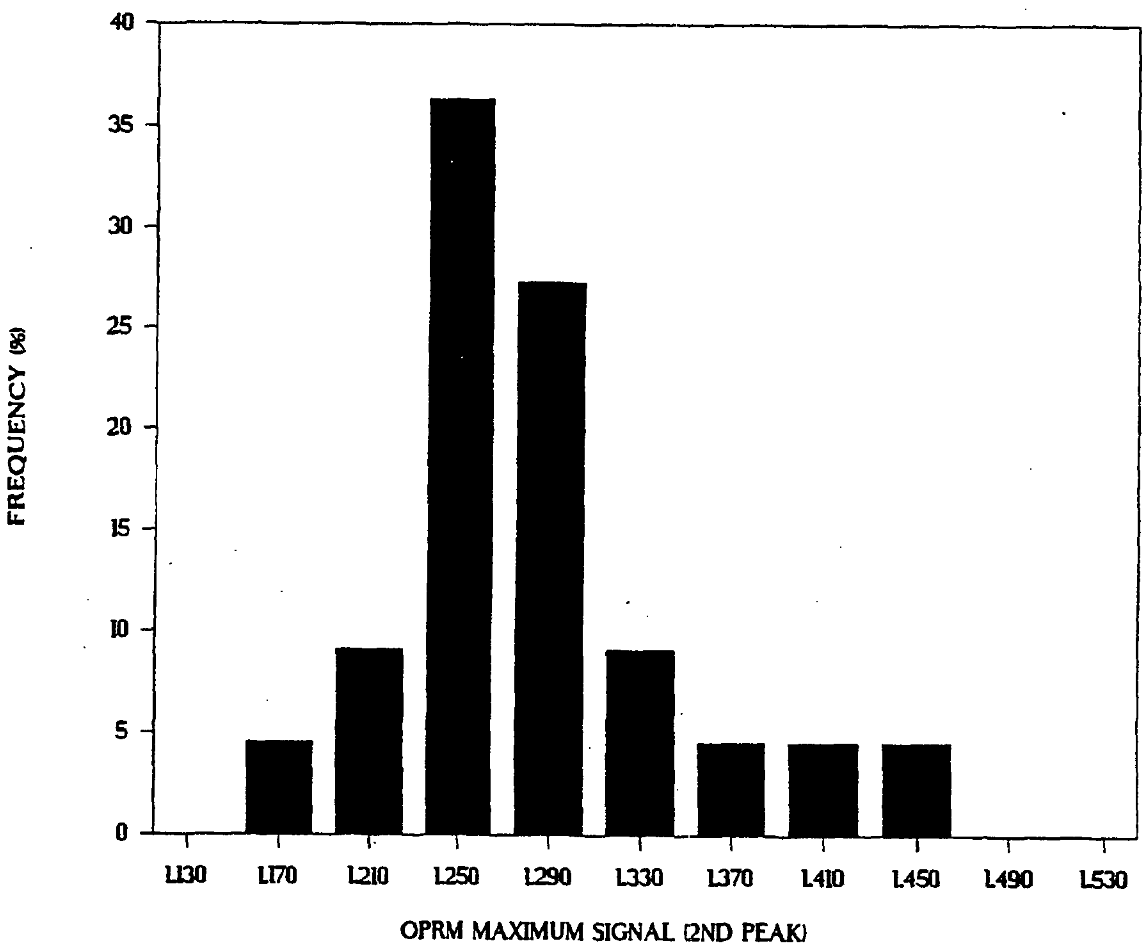


FIGURE 6-27. EXAMPLE SETPOINT OVERSHOOT DISTRIBUTION

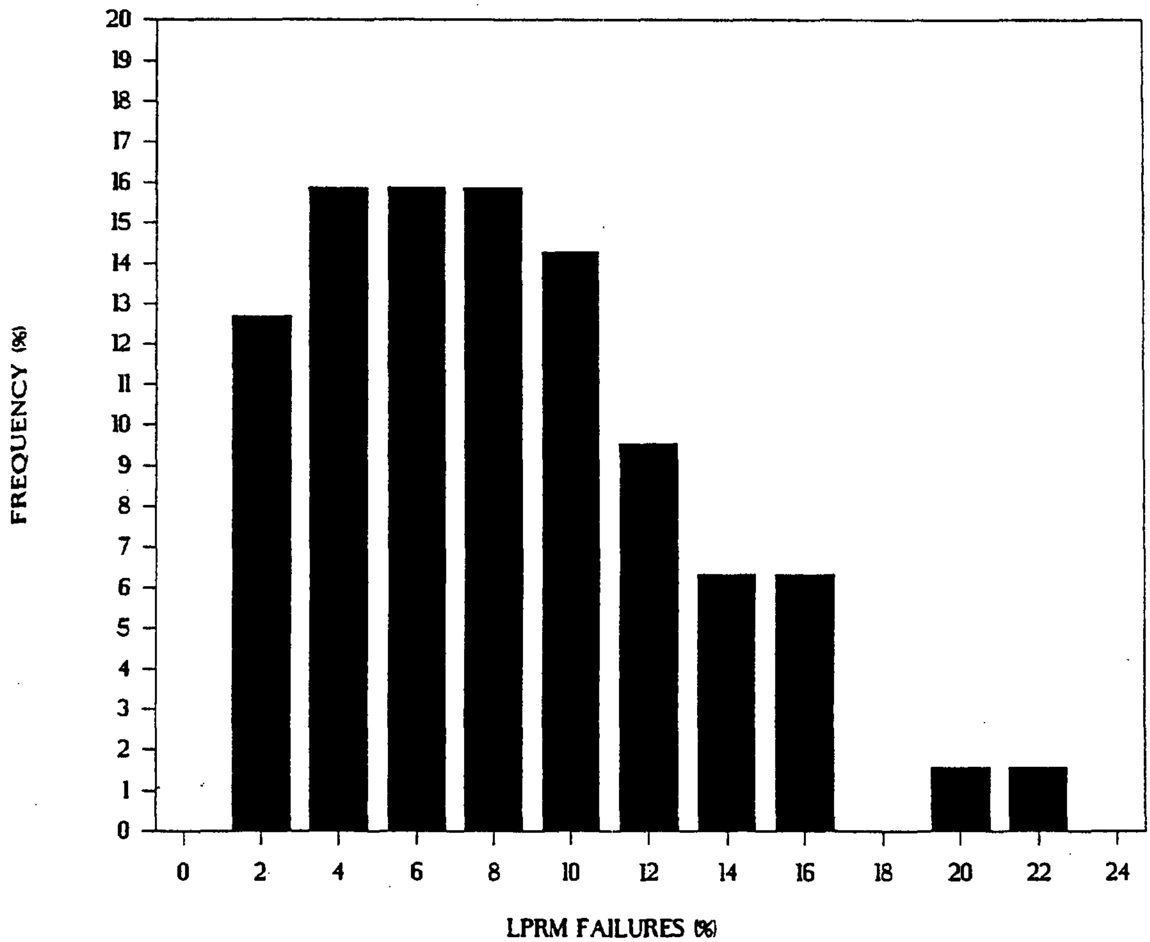
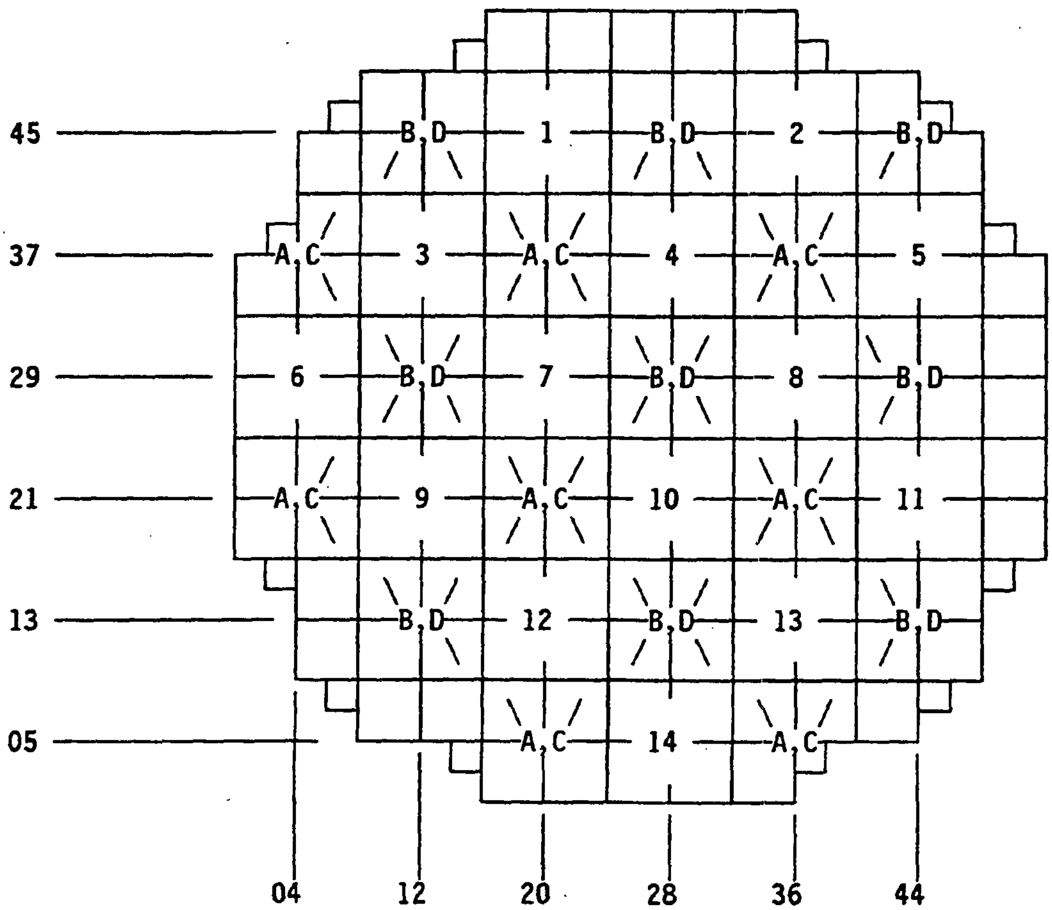


FIGURE 6-28. LPRM FAILURE RATE DATA

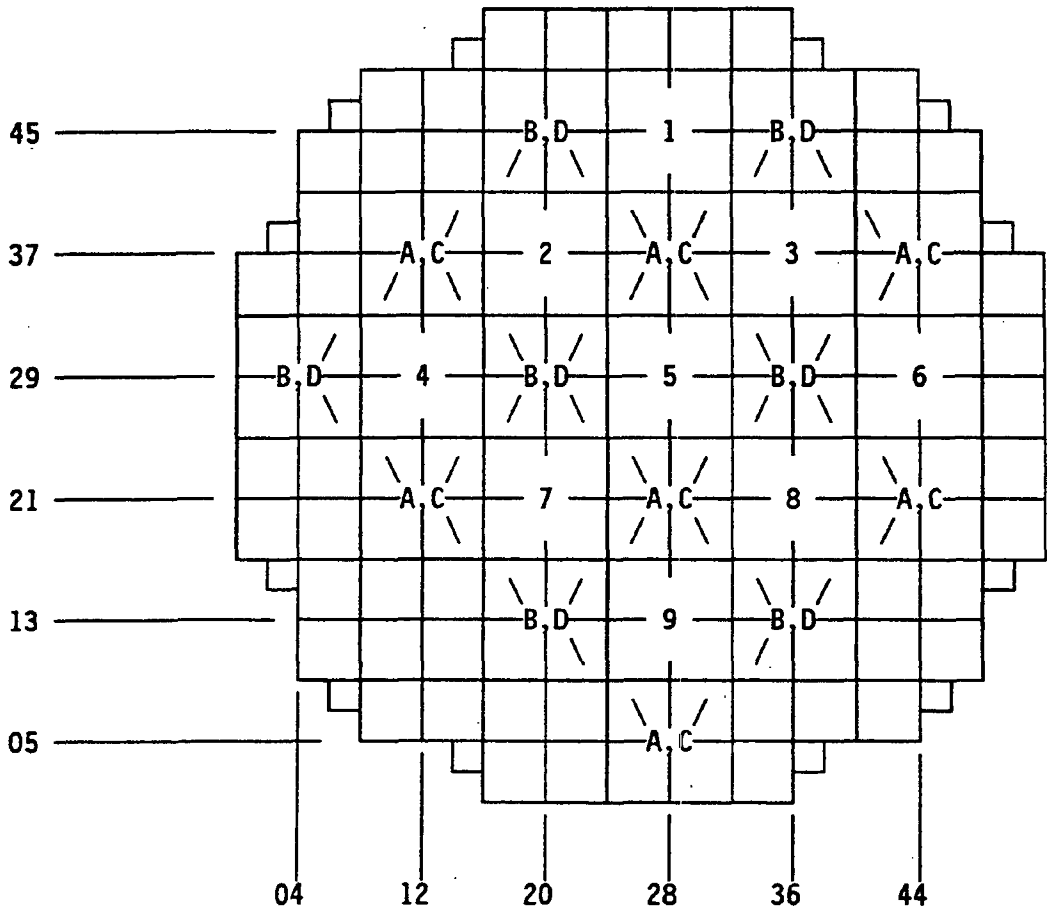



```
     x,y
   /     \
x,y   n   x,y
   \     /
     x,y
```

n = OPRM cell number

x,y = LPRMs assigned to OPRM-A1 channel (other LPRMs in string assigned to OPRM-A2 channel)

FIGURE 6-29. 560 BUNDLE LPRM ASSIGNMENTS TO OPRM A1(A2)




```
       x,y
    /       \
x,y    n    x,y
    \       /
       x,y
```

n = OPRM cell number

x,y = LPRMs assigned to OPRM-B1 channel (other LPRMs in string assigned to OPRM-B2 channel)

FIGURE 6-30. 560 BUNDLE LPRM ASSIGNMENTS TO OPRM B1(B2)

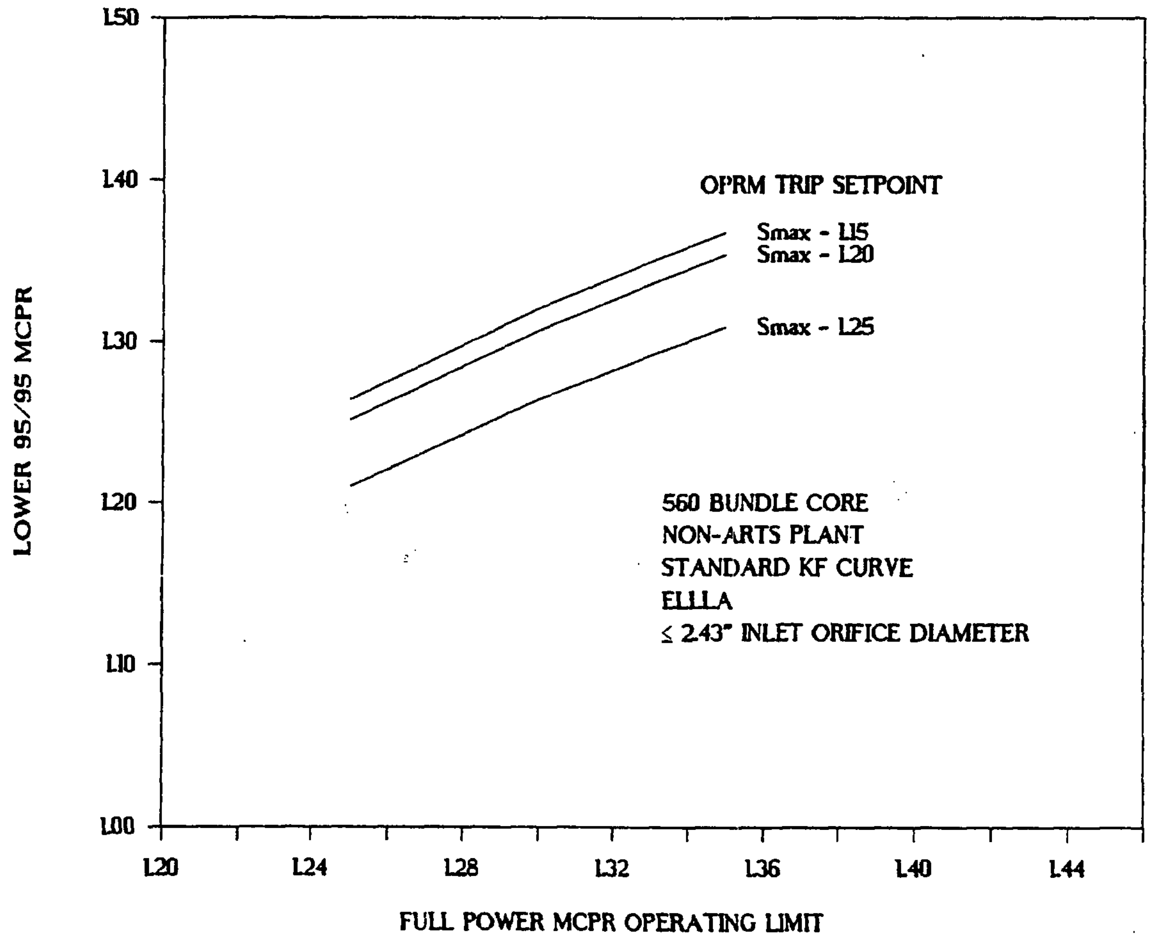


FIGURE 6-31. SAMPLE GENERIC ANALYSIS FORMAT

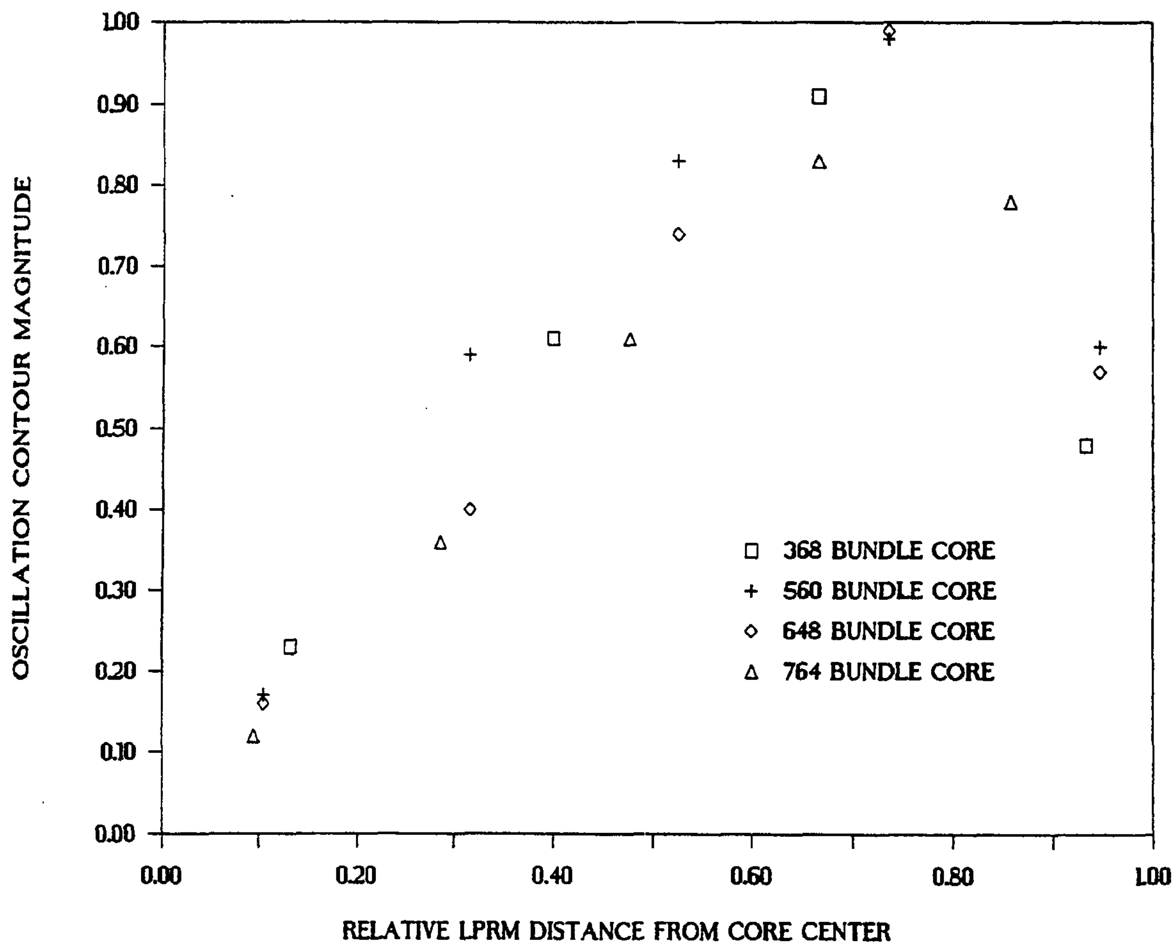


FIGURE 6-32. VARIATION OF OSCILLATION CONTOURS AS A FUNCTION OF CORE SIZE

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2. NEDO-31708, "Fuel Thermal Margin During Core Thermal Hydraulic Oscillations in a Boiling Water Reactor", June 1989.

3. BWROG Letter, S. D. Floyd to A. C. Thadani (NRC) providing report entitled "Licensing Basis for Long-Term Solutions to BWR Stability", February 2, 1990.

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14. NEDE-31096-A, "Response to NRC ATWS Rule - 10CFR50.62," February 1987.

15. NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model for BWRs," February 1986.

APPENDIX A

SOLUTION CONCEPTS

A.1 OPTION I-A - REGIONAL EXCLUSION

A.1.1 Solution Description

The objective of Option I-A is to assure compliance with GDC-12 by preventing the occurrence of instability. This is accomplished by preventing entry into power/flow regions where an instability may occur. The boundary of this excluded region is determined through application of the methodology described in Section 5.0 of the main body of this report. An example of an exclusion region is shown in Figure A-1. Upon entry into this exclusion region, the ASF trip function will cause the region to be exited.

For plants choosing to implement this solution, the existing flow-biased APRM trip circuit may be modified to provide the ASF. The new or modified ASF will be designed to the same requirements as the existing flow-biased scram function. The flow-biased APRM trip line may be modified as shown in the examples on Figure A-2. The modification may be implemented on existing circuit boards. There are also spare slots for an additional board on most APRM units. It is anticipated that for those plants with flow-biased neutron flux trips [i.e., no simulated thermal power monitor (STP)], the modified trip function will also be a neutron flux trip (no STP). For those plants with a STP, it will be retained in the modified trip function.

The ASF must assure that the system will insert a sufficient number of control rods to exit the region. One method is reactor trip. Other available systems such as Select Rod Insert (SRI) may also be used.

The flow-biased rod block signal is also modified in the same manner as the ASF signal, except that the intercept will be lower in power (estimated to be 5-10% lower). The offset between the trip setpoint and the rod block will depend upon whether the plant flow-biased circuitry includes a STP and the

amount of instrument noise typically experienced by the plant at the associated power level. There is no stability design basis requirement associated with selecting this difference since the ASF trip setpoint bounds the region that is potentially susceptible to instability. An example of an exclusion region with an associated rod block is presented in Figure A-2.

The implementation of this solution differs depending upon whether the ASF is a reactor trip or SRI. When a reactor trip is the ASF, the existing flow-biased APRM trip and rod block circuits are modified to provide multiple intercept and slope capability. The applicable intercepts and slopes are then maintained in a manner similar to current practices. When SRI is the ASF, a separate flow-biased circuit is implemented to initiate SRI.

The associated flow-biased APRM rod block trip function can be accomplished in one of two ways. The first way is to modify the existing flow-biased APRM rod block function to utilize the existing trip signal output above a specified flow, and to utilize the output from a new parallel flow-biased APRM rod block circuit that has multiple intercept and slope capability below the specified flow. A second way is to modify the existing flow-biased APRM rod block circuit with multiple intercept and slope capability as if the ASF were reactor trip instead of SRI.

A.1.2 Licensing Approach

The enforcement of an exclusion region with an ASF that prevents oscillations is the basis for meeting GDC-12. Modification of the existing Reactor Protection System flow-biased APRM trip and rod block circuits to detect entry into the exclusion region provides a highly reliable and redundant means for monitoring plant operation relative to the exclusion boundary.

The capability of the ASF to cause the exclusion region to be exited will depend upon the number and distribution of control rods which are inserted. When the ASF is a reactor trip, the region will be quickly exited. When the ASF is SRI, the region will be exited provided a sufficient number and distribution of control rods are inserted. To ensure an adequate control rod configuration is achieved subsequent to SRI initiation, special requirements

will be established. These requirements will address the variability of the impact control rod configuration has on power as a function of control rod pattern, dynamic xenon transients, reactor flow, reactor power and cycle average exposure. It is expected that such surveillance requirements may be met using 3-D nodal simulation techniques.

A.1.3 Methodology Application

The Option I-A concept utilizes the regional exclusion methodology of Section 5 of the main body of this report. This methodology establishes the boundary of a power/flow region where an instability may occur. The conservatism of the procedure used to define the exclusion boundary is confirmed by both steady state and event-based calculations.

A.1.4 Technical Specifications

Implementation of Option I-A with either reactor trip or SRI as the ASF will require modifications to existing Technical Specifications to identify setpoints, instrument operability and surveillance requirements. The following specifications are expected to be revised for implementation of Option I-A:

- Reactor Protection System Instrumentation Setpoints
- APRM Setpoints
- Reactor Protection System Instrumentation
- Control Rod Block Instrumentation
- Recirculation Loops

A.1.4.1 Reactor Protection System Instrumentation Setpoints

The RPS instrumentation Setpoints Specification will be modified if reactor trip is selected as the exclusion region ASF. The modification to existing Technical Specifications will identify setpoints for the multiple intercepts and slopes of the modified flow-biased APRM trip circuit while retaining current setpoints beyond the upper flow of the exclusion boundary. The modified specification will include a table similar to the following:

		Trip Setpoint	Allowable Value
(a)	Flow Biased		
	Slope 1 ($W > W1$)	$aW + y1$	$aW + y2$
	Slope 2 ($W2 \leq W < W1$)	$bW + y3$	$bW + y4$
	Slope 3 ($W3 \leq W < W2$)	$cW + y5$	$cW + y6$
	Slope 4 ($W < W3$)	$dW + y7$	$dW + y8$
(b)	High Flow Clamped	$x1$	$x2$

W = drive flow (% rated)
W1,W2,W3 = drive flow slope change points (% rated)
a,b,c,d = setpoint slopes (%/%)
yi = setpoint intercepts (% of rated)
xi = clamped setpoints (% of rated)

A.1.4.2 APRM Setpoints

For some plants, the APRM setpoints specification requires that the flow-biased scram trip and rod block setpoints be adjusted by a factor "T". T is defined as the ratio of the fraction of rated thermal power (FRTP) to the core maximum fraction of limiting power density (CMFLPD). A similar adjustment is made for units using ANF reload fuel. The purpose of this specification is to ensure that these setpoints are adjusted to account for differences in peaking from the peaking assumed in the calculation of core thermal limits. Since the new flow-biased setpoints protect against the onset of instability, T-Factors or their equivalent are not applicable to the setpoints for the lower trip function.

A.1.4.3 Reactor Protection System Instrumentation

The RPS instrumentation specification addresses requirements for each RPS trip function. These requirements are (1) operational condition for which function is required, (2) minimum number of operable channels per trip system, (3) LCO action statement, (4) response time and (5) surveillance requirements.

When reactor trip is the ASF, there are no additional requirements for the RPS instrumentation.

The RPS instrumentation is also used when the option for using SRI as the ASF is selected. There are no changes in the requirements of the RPS instrumentation specifications. In addition to the RPS instrumentation, there will be instrumentation utilized in achieving SRI actuation. This instrumentation will meet the same requirements for the operational condition for which the function is required and for surveillances. Requirements for the minimum number of operable channels per trip system, LCO action statements and response times will be established consistent with the configuration of the SRI instrumentation.

A.1.4.4 Control Rod Block Instrumentation Setpoints

Consistent with the philosophy of the Improved Technical Specifications developed by the BWROG (Reference 12), the rod block function associated with the flow-biased trip need not be included in the Technical Specifications. There is no stability design basis requirement associated with the rod block and, therefore, the rod block function does not satisfy the necessary criteria for inclusion in the Technical Specifications. For those plants with Rod Block instrumentation Technical Specifications, modifications similar to the RPS instrumentation Setpoints of Section A.1.4.1 are made.

A.1.4.5 Recirculation Loops

The current specification addresses two loop operation, single loop operation, and operation in natural circulation. In addition, some plants provide direction to monitor neutron flux noise when operating at high power and low core flow conditions. It is proposed that the specification be modified to include the following requirements, and that plants delete the neutron flux noise monitoring requirements where they occur in existing Technical Specifications.

A map of the exclusion region as defined by the modified flow-biased APRM setpoints should be included (the exclusion region power/flow map may be placed in the plant's Core Operating Limits Report (COLR)). The area outside the exclusion region is an area of unrestricted operation. The unrestricted area, as the name implies, is an area where operation is allowed without restriction with respect to stability. The analyses performed to define the exclusion region boundary demonstrate adequate margin for steady state operation and for transients which initiate and remain within the unrestricted area. For transients which initiate within the unrestricted area and result in entry into the exclusion region, an instability is precluded by the initiation of the ASF.

A.1.5 Operator Guidance

Operator guidance will be provided to avoid the exclusion region (i.e., the ASF) during startup and shutdown. This guidance would direct the operator to increase core flow prior to exceeding a predetermined power level, which would be lower than the new rod block associated with the ASF setpoint of the exclusion region. The flow increase would result in a state where, given an increase in power to the target rod line, operation would remain outside the exclusion region.

Similarly, the operator would be instructed to insert control rods during a controlled shutdown prior to decreasing flow below a predetermined value. The rod insertions would continue to a point below the rod line which bounds the lower end of the exclusion region. The operator will also be given appropriate direction to respond to flow reduction and loss of feedwater heating events to avoid unnecessarily entering the exclusion region and initiating an ASF. Additional operator guidance will be provided to recover from SRI actuation for plants selecting SRI as the ASF. This guidance will include directions for reviewing the conditions leading to SRI initiation, confirmation that the exclusion region was entered and recovery instructions for returning to the pre-SRI rod pattern.

A.2 OPTION I-C - REGIONAL EXCLUSION WITH STABILITY APRM FLUX TRIP

A.2.1 Solution Description

The objective of Option I-C is to assure compliance with GDC-12 using a combination of prevention and detection and suppression. This is accomplished during normal operation by administratively controlling entry into an exclusion region identified as being susceptible to thermal-hydraulic oscillations. Intentional entry is only permitted for two speed recirculation system (flow control valve) plants during pump upshifts. If the exclusion region is entered as a result of an unexpected operational event the Stability APRM Flux Trip is armed and the operator is required to immediately exit the region. Should an oscillation occur while exiting the region the Stability APRM Flux Trip system detects and automatically initiates suppression of the oscillation through actuation of an ASF prior to exceeding the MCPR Safety Limit.

The exclusion region is determined using the methodology described in Section 5.0 of the main body of this report. This methodology has been demonstrated to be conservative for operating conditions resulting from both steady state and transient conditions. A representative boundary for the exclusion region is presented in Figure A-1.

For plants choosing to implement this solution the RPS will be modified to include the Stability APRM Flux Trip system. When the exclusion region is entered an alarm is actuated to alert the operator that the region has been entered and that the Stability APRM Flux Trip system is armed. When armed, each Stability APRM Flux Trip channel compares the unfiltered APRM signal with the Stability APRM Flux Trip setpoint (SFSP).

The SFSP consists of the simulated thermal power (STP) signal plus a delta of about 2-10%. The STP is a conditioned APRM signal which simulates the actual fuel cladding surface heat flux. The heat flux and the instantaneous neutron flux are different due to the thermal conductivity characteristics of the fuel. These characteristics result in a more slowly varying fuel thermal response relative to the instantaneous or unfiltered APRM neutron flux signal. The SFSP will be smaller than the neutron flux magnitude associated with

oscillations large enough to violate the MCPR Safety Limit prior to suppression. When the instantaneous neutron flux signal exceeds the SFSP the ASF is actuated.

The ASF must assure that the system will insert a sufficient number of control rods to exit the region. One method is reactor trip. Other available systems such as SRI can be used if sufficient control rod worth is available to exit the exclusion region. The new or modified ASF will be designed to the same requirements as the existing flow-biased scram function. At a value of approximately 25-50% of the SFSP, an alarm function with a manual reset is provided. The exact value of the trip and alarm setpoints are determined as described in Section A.2.3. The qualitative relationships between the STP signal, the APRM flux noise, the SFSP, and the alarm setpoint are presented in Figure A-3.

The Stability APRM Flux Trip system is not capable of discriminating between true oscillations and other flux signals. If while operating in the exclusion region the instantaneous neutron flux exceeds the SFSP, an ASF is initiated by the RPS. It is often necessary during a pump upshift maneuver in a two speed recirculation system plant to operate in the exclusion region for short periods of time. During the actual pump upshift the instantaneous neutron flux increases quickly as power responds to the increased core flow. This increase could initiate a spurious Stability APRM Flux Trip system actuation in the absence of an oscillation.

To reduce the chance of spurious scrams during the pump upshift maneuver while in the exclusion region, the Stability APRM Flux Trip system has an option for bypass of the trip. When the bypass option is selected, administrative controls for manual detection and suppression of oscillations are used during the brief period of operation with the Stability APRM Flux Trip System bypassed.

A.2.2 Licensing Approach

The general licensing approach for Option I-C is to assure compliance with GDC-12 by preventing the occurrence of oscillations that could result in

violation of the MCPR Safety Limit. This is primarily accomplished through preventing operation in a region of potential instability defined by application of the methodology of Section 5.0. Secondarily, if that region is unintentionally entered as a result of an unplanned operational event, this solution provides automatic detection and suppression of unacceptable oscillation levels through the Stability APRM Flux Trip and ASF.

The SFSP will be selected to provide margin to expected magnitudes of unfiltered APRM signal noise levels. The Stability APRM Flux Trip system with the selected SFSP will then be demonstrated to provide adequate margin to the MCPR Safety Limit during a postulated neutron flux oscillation using the methodology of Section 6.0. Analytical MCPR Safety Limit compliance will be demonstrated for all expected modes of GE BWR neutronic/thermal-hydraulic neutron flux oscillations (defined in Section 6.1).

The bypass trip function will allow FCV plants to operate in or very near the exclusion region for the purpose of performing pump upshifts. The bypass will be controlled at the control room main panel and will conform to Reg. Guide 1.47. With the trip function bypassed, the operators will be required to monitor available instrumentation to ensure that instabilities are detected and suppressed in accordance with the guidelines provided in Bulletin 88-07, Supplement 1.

The capability of the ASF to reduce power to below the exclusion region will depend upon the number and distribution of control rods which are inserted. When the ASF is a reactor trip, the region will be quickly exited. When the ASF is Select Rod Insert (SRI), the region will be exited provided a sufficient number and distribution of control rods are inserted. To ensure an adequate control rod configuration is achieved subsequent to SRI initiation, special requirements will be established. These requirements will address the variability of control rod configurations, dynamic xenon transients, reactor flow, reactor power and cycle average exposure. It is expected that such surveillance requirements may be met using 3D nodal simulation techniques.

A.2.3 Methodology Application

The basic approach to establishing the SFSP is to first determine an acceptably low setpoint such that expected plant evolutions will not result in Stability APRM Flux Trip system actuation and then confirm that the setpoint provides adequate margin to the MCPR Safety Limit. Therefore, evaluation of plant operating data against potential Stability APRM Flux Trip system designs is a necessary step in the determination of the SFSP. Basic characteristics of oscillations are already known from test data and operating experience. Neutron flux signals generated by simplified point kinetics model (including the effects of random noise) and digitally recorded plant data during various maneuvers are analyzed to determine the margin to trip for expected plant maneuvers without oscillations.

The analytical methodology used to demonstrate that the SFSP provides margin to the MCPR Safety Limit is described in detail in Section 6.0. The methodology simulates LPRM and APRM responses to oscillations of various growth rates. These simulations explicitly account for delays associated with the trip system and ASF. Sections 6.2 and 6.3 detail how these delays and trip system setpoints are related to the MCPR performance of the limiting bundles, and how uncertainties are treated.

A.2.4 Technical Specifications

Implementation of Option I-C with either reactor trip or SRI as the ASF will require modifications to existing Technical Specifications to identify setpoints, instrument operability and surveillance requirements. The following specifications are reviewed to describe the expected revisions.

- Reactor Protection System Instrumentation Setpoints
- Reactor Protection System Instrumentation
- Recirculation Loops

A.2.4.1 Reactor Protection System Instrumentation Setpoints

The RPS Instrumentation Setpoints specification provides setpoints and allowable values for each RPS signal. A Stability APRM Flux Trip setpoint and the intercept and slope for the flow-biased description of the exclusion region must be added. For application to two speed recirculation system plants to accommodate the pump upshift maneuver, bypass provisions will also be specified. The modified specification will include a table similar to the following:

	Trip Setpoint	Allowable Value
Stability APRM Flux Trip		
(a) Trip System Bypass	aW + y1	aW + y2
(b) SFSP	STP + x1	STP + x2

W = drive flow (% rated)
a = bypass setpoint slope (%/%)
y1,y2 = bypass setpoint intercepts (% of rated)
x1,x2 = setpoint delta (% of rated)

A.2.4.2 Reactor Protection System Instrumentation

The RPS Instrumentation specification addresses requirements for each RPS trip function. These requirements are (1) operational conditions for which function is required, (2) minimum number of operable channels per trip system, (3) LCO action statement, (4) response time, and (5) surveillance requirements. This specification will have to be modified to include requirements for the Stability APRM Flux Trip system. These requirements will be the same as those for the existing flow-biased trip but with a note indicating the applicable operational conditions and that the trip function may be bypassed under administrative controls. When reactor trip is the ASF there are no additional requirements for the RPS instrumentation specification.

When SRI is selected as the ASF, the instrumentation used to achieve SRI actuation will be required to meet the same operational conditions and surveillance requirements as the instrumentation used to achieve reactor trip. Requirements for the minimum number of operable channels per trip system, LCO action statements, and response times will be established consistent with the configuration of the SRI instrumentation. These requirements will enforce the need to avoid the exclusion region and to follow the ICAs in the event of inadvertent entry with the SRI function inoperable.

A.2.4.3 Recirculation Loops

The current specification addresses two loop operation, single loop operation and operation in natural circulation. In addition, some plants provide direction to monitor neutron flux noise when operating are high power and low core flow conditions. It is proposed that the specification be modified to include the following requirements, and that plants delete the neutron flux noise monitoring requirements where they occur in existing Technical Specifications.

A map of the exclusion region as defined by the modified flow-biased APRM setpoints should be included (the exclusion region power/flow map may be placed in the plant's COLR). The area outside the exclusion region is an area of unrestricted operation. The unrestricted area, as the name implies, is an area where operation is allowed without restriction with respect to instability. The analyses performed to define the exclusion region boundary demonstrate adequate margin for steady state operation and for transients which initiate and remain within the unrestricted area.

An LCO for this specification will be provided only for two speed recirculation system plants to allow unplanned entry for the purpose of performing a pump upshift maneuver. Action statements will direct immediate rod insertion or flow increase upon indication of instability with the ASF bypassed.

A.2.5 Operator Guidance

Operator guidance will be provided to avoid the exclusion region during startup and shutdown. This guidance will direct the operator to increase core flow prior to exceeding a predetermined power level, which would be lower than the rod line associated with the exclusion region (and Stability APRM Flux Trip system actuation). The flow increase would result in a state where, given an increase in power to the target rod line, operation would remain outside the exclusion region.

Similarly, the operator would be instructed to insert control rods during a controlled shutdown prior to decreasing flow below a predetermined value. The rod insertions would continue to a point below the rod line which bounds the lower end of the exclusion region. The operator will also be given appropriate direction to respond to flow reduction and loss of feedwater heating events to avoid entering the exclusion region.

Additional direction for responding to upset conditions, including alarm response, will be provided. For example, following a trip of both reactor pumps, the operator will be directed to reduce reactor power by control rod insertion to a point below the exclusion region before attempting to restart the recirculation pumps. In all cases of inadvertent entry into the exclusion region, the operator will be directed to take immediate action to exit the region a manner designed to minimize the impact on instantaneous neutron flux levels (e.g., limiting the rate of flow increases, prohibiting rod withdrawal).

Operators of plants with two-speed reactor recirculation systems will implement, as necessary, procedural control of the selection of the trip bypass feature. These controls will permit selection of the trip bypass feature while in or near the exclusion region only for the purpose of performing the pump upshift. In addition to limiting the time with the trip function bypassed, these controls will specify a manual actuation of the ASF upon indication of an instability. Guidance similar to that contained in NRC Bulletin 88-07, Supplement 1 for operation in Region C will be provided for operation in the exclusion region.

A.3 OPTION I-D REGIONAL EXCLUSION WITH FLOW-BIASED APRM NEUTRON FLUX SCRAM

A.3.1 Solution Description

The objective of Option I-D is to assure compliance with GDC-12. This is accomplished by providing an administrative boundary for normal operations around the region where an instability could be expected to occur. This region has the same basis as the exclusion region defined in Section A.1. During normal operations, the boundary of the exclusion region is administratively controlled and operation within the region is avoided. An example of an exclusion region is presented in Figure A-1. If an unexpected operational event results in an entry into the exclusion region, steps will be taken to exit the region immediately. However, should oscillations occur, they will be automatically detected and suppressed by the Flow-Biased APRM Neutron Flux scram. For BWRs with tight fuel inlet orificing, the probability of regional (out-of-phase) oscillations is very low and the expected mode of oscillation is a core-wide (in-phase) oscillation. In addition, plants with small diameter cores are also less likely to experience regional oscillations because of the strong preference of the fundamental mode of the neutronics. Calculations will be performed to confirm that for core-wide oscillations the Flow Biased APRM Neutron Flux scram system will automatically detect and suppress oscillations prior to exceeding the MCPR Safety Limit. Protection commensurate with the low probability of regional oscillations will also be demonstrated for the Flow Biased APRM Flux scram system.

Oscillations will be detected by the existing APRM System. The oscillations will be suppressed by a Flow-Biased APRM Neutron Flux scram system, with an example shown in Figure A-4. The trip is based on comparing the unfiltered APRM signal to a setpoint that varies as a function of core flow. This option will not use a Simulated Thermal Power Monitor for detection. The Flow Biased APRM Neutron Flux scram system is part of the Reactor Protection System (RPS) trip logic and uses the one-out-of-two-taken-twice logic. Therefore, a spurious signal on one channel will not initiate a trip.

An RPS channel trip is generated when the unfiltered APRM neutron flux exceeds the flow-biased setpoint. The rod block function (also shown in Figure

A-4) serves as an alarm function and is typically located 8% lower in power than the scram setpoint. LPRM upscale and downscale alarms also provide the operators with indications of possible instabilities.

The Flow-Biased APRM Neutron Flux scram line relative to the estimated exclusion region for a BWR/4 with tight inlet orificing is shown in Figure A-5, and illustrates the close proximity of the scram line to the region of potential instability. At the condition of least margin to an instability, natural circulation and the maximum rod line, the scram setpoint is only several percent above the condition.

A.3.2 Licensing Approach

The general licensing approach for Option I-D is to assure compliance with GDC-12 by preventing the occurrence of instability-related oscillations that could result in a violation of the MCPR Safety Limit. This is primarily accomplished through preventing operation in the region of potential instability as defined in Section A.1 for Option I-A. However, if that region is unintentionally entered as a result of an unplanned operational event, this solution provides automatic detection and suppression of unacceptable oscillations through the Flow-Biased APRM Neutron Flux trip input to the RPS.

A.3.3 Methodology Application

A.3.3.1 Exclusion Region Boundary

The Option I-D concept uses the regional exclusion methodology of Section 5.0 of the main body of this report. This methodology establishes the boundary of a power/flow region in which an instability may occur. The conservatism of the procedure used to define the exclusion boundary has been confirmed by both steady state and event-based calculations (Sections 5.2.7 and 5.4). For Option I-D, intentional entry into the exclusion region during normal operations will be administratively prohibited.

To demonstrate the application of the methodology to a plant with tight fuel inlet orificing, Duane Arnold Cycle 10 was chosen. Duane Arnold currently is licensed for extended load line operation and has an uprated power density of 51 kW/l. Other key parameters which affect stability are summarized in Table A-1. Comparisons are also provided to other BWRs. Duane Arnold Cycle 10 contains predominantly GE 8x8 fuel. Core average Haling power shapes were generated at the EOC-10 using the GE 3D BWR Simulator. These power shapes are used in the radial and core average axial power distributions for the methodology described in Section 5.2. The full power Haling core average axial power shape which is used for the average channel power shapes when evaluating conditions at forced circulation flow rates is shown in Figure A-6. Also shown is the core average axial power shape that results from a flow reduction to natural circulation from the full power Haling condition. This axial power shape is used for the average channels when evaluating natural circulation conditions. The most negative void coefficient during the cycle is chosen for all analyses and occurs at EOC-1000 MWd/ST.

Calculations were performed at a high rod line, a lower rod line, and at natural circulation flow at the conditions defined in Figure A-7. The core and channel decay ratios for these conditions are summarized in Table A-2 and Figure A-8. The decay ratios are compared to the stability criteria (Section 5.1) in Figure A-8. Decay ratios less than the limit shown in Figure A-8 provide sufficient conservatism to potential instabilities. The intersection of the points which have calculated decay ratios equal to the stability criteria defines the exclusion region boundary. For the high rod line, the medium rod line, and natural circulation, interpolation is used between Points 1 and 2, 5 and 6, and 8 and 9, respectively. Based on these results, the exclusion region for Duane Arnold Cycle 10 is shown in Figure A-9.

The effect of the tight inlet orificing on channel decay ratios can be observed from the results in Figure A-8. This results in channel decay ratios which are well below 0.5 for all cases evaluated. This result will be discussed later in relation to expected modes of oscillation.

A.3.3.2 Expected Modes of Oscillation

The Flow-Biased APRM Neutron Flux scram will be demonstrated to be capable of responding to reasonably postulated modes of BWR stability-related oscillations. The stability design basis of the Flow-Biased APRM Neutron Flux scram system shall be to generate a trip signal during oscillations of sufficiently low amplitude to provide margin to the MCPR Safety Limit for expected modes of oscillations for cores with tight fuel inlet orificing. Tight fuel inlet orificing leads to a dominant core-wide mode, where the neutron flux in all fuel assemblies oscillates in phase. In addition, small diameter cores have very large eigenvalue separations for the azimuthal harmonics which reduces the probability of encountering regional oscillations. For core-wide oscillations, the Flow-Biased APRM Neutron Flux scram function provides direct detection and suppression prior to exceeding the MCPR Safety Limit (Section A.3.3.3). Plant data have also clearly demonstrated the ability of the APRMs to detect regional oscillations. Because the regional modes are unlikely to occur, more nominal assumptions are used in demonstrating compliance (Section A.3.3.3).

The expected modes of oscillation are dependent on the nuclear and thermal-hydraulic characteristics of the core. The existence of regional oscillations requires the excitation of the first-order azimuthal harmonic of the neutron flux. This may occur due to highly responsive channel hydraulic characteristics that are indicative of high channel decay ratios. The hydraulic response is necessary to provide sufficient reactivity to overcome the subcritical nature of the harmonic mode. Section 6.1 of the main body of this report discussed the basic characteristics of expected modes in BWRs. The following sections will discuss those characteristics of a specific class of plants which affect the potential modes of oscillation.

(1) <u>Channel Hydrodynamic Decay Ratio</u>

Section 5.0 of the main body of this report discusses the methodology and criteria used to predict whether oscillations will occur for a given set of inputs. Although the methodology concentrates on the calculation of core and channel decay ratios, the prediction of the potential for regional

oscillations is implicitly included. Most frequency domain codes do not explicitly evaluate the decay ratio for the harmonic modes of oscillation and, therefore, a correlation is developed that relates the regional decay ratio to the core and channel decay ratios. Alternatively, other methods can be developed to estimate the regional decay ratio.

To estimate the regional decay ratio, the characteristics of a regional oscillation must be understood relative to the characteristics of channel and core stability. Channel stability involves the response of a single channel under constant plenum-to-plenum pressure drop conditions with no nuclear feedback. Core-wide stability involves the coupled response of all channels in the core through the void reactivity feedback. This total core response, similar to a core-wide pressure perturbation, results in oscillations in the total core inlet and exit flows. The oscillation in the exit flow is then transmitted through the recirculation loop and is coupled back to the core inlet flow. Axially, a spatial harmonic is excited in the neutron flux (Reference 13) that is the result of the density wave oscillation. The void reactivity feedback must be large enough to sustain the axial harmonic. A regional oscillation is similar to both single channel and core-wide oscillations. In a regional oscillation, not only is the axial harmonic excited, but also an azimuthal harmonic is excited in the neutron flux. This is the result of channel flows in one-half of the core oscillating ouf-of-phase with channel flows in the other half of the core. This provides a spatial reactivity feedback that can excite the azimuthal harmonic.

Although the azimuthal harmonic is a subcritical mode, the channel hydraulics are more responsive in the regional mode (similar to a single channel oscillation) where the plenum-to-plenum pressure drop remains essentially constant.

In general, the choice of a dominant mode is influenced by the inertial resistance of the recirculation loop which inhibits oscillatory flow, the responsiveness of channel flows under various boundary conditions, and the subcritical reactivity associated with the harmonic modes. The recirculation loop inertia works to stabilize the core-wide mode and impose

constant pressure drop characteristics on the channels (leading to more responsive channel flows). More responsive channel flows under constant boundary conditions can provide additional reactivity feedback for the regional modes of oscillations. However, large eigenvalue separations of harmonic modes will tend to inhibit the excitation of the harmonic modes. For a regional mode of oscillation, the gain in channel flow responsiveness must be sufficient to overcome the subcritical nature of the neutronics. This section will concentrate on a discussion of the channel response as characterized by the channel decay ratio.

To separately study the potential for regional oscillations, a frequency domain analytical model was developed which includes a power feedback transfer function for individual channel hydrodynamic calculations. The channel hydraulics for this model are based on the GE one-dimensional transient model, ODYN (Reference 15). The one-dimensional liquid and conservation equations of mass and energy are linearized and the Laplace transform taken, such that small-perturbation techniques can be used. Perturbations in local flow variables (such as liquid and vapor velocities, void fraction) are then used in the momentum equation to calculate pressure drop perturbations. This form of the model is capable of calculating the standard channel hydrodynamic decay ratio.

As discussed above, regional oscillations are similar to single channel oscillations in that the plenum-to-plenum pressure drop remains essentially constant. However, a regional oscillation occurs with changes in neutron flux through the coupling with the void reactivity. This feedback term is not included with single channel models, since the heat flux is assumed to remain constant. The void reactivity feedback can be simulated by modeling the neutronics, specifically the harmonic modes. However, the feedback can also be estimated using results from the GE 3D BWR Simulator, which predicts the change in neutron flux for a given change in moderator density. The neutron flux change is a direct result of changes in the local moderator density, for the case of localized perturbations. For regional oscillations, this is a good assumption, since the fundamental mode is not significantly perturbed.

To account for the change in neutron flux (or heat generation rate within the fuel), a power feedback term is included in the channel model and is defined as

$$\delta P_k = (\delta P/\delta U)_k * \delta U_k \tag{A-1}$$

where

k = Axial node
P_k = Nodal power (heat generation rate),
U_k = Nodal water density
δ = Perturbation

For the channel hydrodynamic model, perturbations in channel flow will result in changes in coolant density, which, in turn, will cause changes in the two-phase pressure drop. This effect is fed back to the channel flow through the constraint that the total pressure drop must remain constant. Therefore, changes in moderator density are explicitly calculated in the channel model. To approximate the regional oscillation case, the effect of this change in moderator density must be fed back to the hydraulics model through the power feedback term of Equation A-1. Since the change in nodal heat generation rate occurs almost instantaneously following a change in local moderator density, the ($\delta P/\delta U$) term can be treated as a simple gain in the regional model with no phase lag. Since the feedback term will be applied separately at each axial node, the axial phase lag associated with the density wave oscillation will be explicitly included. Also, the nodal heat generation rate change is first applied to the fuel heat transfer model to correctly account for the gain and phase lag introduced by heat conduction through the fuel, gap, and cladding. Therefore, including the power feedback term of Equation A-1 approximately models the physical phenomena during a regional oscillation.

The more difficult task is to determine an appropriate representation of the ($\delta P/\delta U$) term in Equation A-1. Because of the nearly instantaneous change in neutron flux with moderator density changes, a quasi-steady-state assumption is used to estimate ($\delta P/\delta U$). The GE 3D BWR Simulator is used to

establish steady-state conditions at the specific point to be analyzed. The steady-state condition is then perturbed by changing the inlet flow for a group of channels that would be expected to oscillate in phase during a regional oscillation. The perturbation is small enough that the fundamental mode reactivity is not changed. The total core power is held constant and the perturbed power distribution and moderator density are calculated. For the channels with perturbed inlet flow, the change in nodal power and change in moderator density are explicitly calculated by the Simulator at each axial node such that $(\delta P/\delta U)_k$ can be determined.

Sensitivity studies were performed to determine an appropriate inlet flow perturbation and number of bundles to be perturbed. In general, the results are insensitive for inlet flow perturbations of 1% to 10%. Also, as the number of bundles perturbed increases, the resulting change in nodal power reaches a plateau such that perturbations in additional bundles will not increase the relative bundle power increases. This suggests that for a regional oscillation, a relatively small group of bundles in a local region provides the majority of the reactivity feedback necessary to sustain the oscillations.

The regional feedback model was compared to stability data for several known regional oscillations. For Caorso Cycle 2 stability tests, tests at KRB-B, and initial cycle testing at Leibstadt, the above procedure was followed to estimate $(\delta P/\delta U)_k$ for the hot and average channels. In all cases, the pure channel hydrodynamic decay ratios (no power feedback) for the hot channels were less than 1.0. When the power feedback term was included, the hot channels were predicted to have "regional" decay ratios greater than 1.0 for conditions of known regional oscillations. Also, the average channels were predicted to have "regional" decay ratios close to 1.0 ($\geq$ 0.9). In general, the channel decay ratios increased by 0.3 when the power feedback term was included. Additional sensitivity studies were performed to determine the sensitivity of the results to conditions of varying core decay ratios.

For conditions with increasing core decay ratio, the change in channel decay ratio increased. Also, as the channel decay ratio increased, the

relative changes in individual channel decay ratio increased. This is expected, since the channels with higher decay ratios are more responsive and have a larger change in moderator density during an oscillation. For a constant power feedback coefficient ($\delta P/\delta U$), this would result in more feedback to the hydraulics with a resulting increase in decay ratio.

Based on the results of the qualification cases and additional sensitivity studies, a map of the core versus channel decay ratios that would result in a "regional" decay ratio of greater than or equal to 1.0 was constructed. These results are summarized in Figure A-10, where a bounding curve has been included which encompasses all the results. For conditions outside this region, regional decay ratios would be predicted to be less than 1.0. The results in Figure A-10 were used along with other qualifications to generate the stability criteria shown in Figure A-11. Two core-wide oscillations that have been observed in GE BWRs are also plotted in Figure A-11 relative to the stability criteria. The two cases included are for the Vermont Yankee Cycle 8 limit cycle condition and the LaSalle-2 instability event. These results demonstrate the relatively low channel decay ratios calculated during core-wide oscillations.

Based on the results in Figures A-10 and A-11, for low channel decay ratios (< 0.5), even core decay ratios near 1.0 are predicted to result in a "regional" decay ratio less than 1.0. Physically, the channel decay ratio is a measure of the responsiveness of the channel flow. For a regional oscillation, this is very important in providing the extra feedback to sustain the oscillations. However, for very stable channels, even under constant plenum-to-plenum pressure drop conditions, sufficient flow oscillations are not available to sustain the oscillations. The core and channel decay ratios for a plant with tight fuel inlet orificing (Duane Arnold, Cycle 10) are summarized in Figure A-8. These results are based on the procedure of Section 5.2, which is specifically designed to provide a conservative estimate of the decay ratios. Even for these conservative conditions, considerable margin is maintained to the conditions predicted to result in regional oscillations. For conditions that result in exceeding the stability criteria of Figure A-11, the core decay ratios are from 0.8 to

1.0, with channel decay ratios of less than 0.4, resulting in a predicted core-wide mode.

An additional calculation was performed at the intersection of the 110% rod line and natural circulation (Table A-2, Point 10). This point is well within the defined exclusion region and, therefore, operator actions would result in minimizing the time at this condition. Even for this condition, the channel decay ratio calculated using the conservative procedure of Section 5.2 is less than 0.4. This condition would again be predicted to result in core-wide oscillations.

(2) Eigenvalue Separation for Small Diameter BWRs

The steady-state neutron flux in a BWR is analogous to the solution of the standard wave equation which involves eigenfunctions with unique properties which make them useful in transient and stability analysis. The eigenfunctions form a complete set and therefore, any function within the proper domain may be expanded in a linear combination of the eigenfunctions. In the modal synthesis method (Reference 12), the time-dependent neutron flux is expanded in terms of steady-state eigenfunctions with time-dependent coefficients. The fundamental mode is the eigenfunction with the largest eigenvalue and represents the steady-state solution to the neutron diffusion equation. The higher order modes, referred to as the harmonic modes, possess eigenvalues which are smaller than the eigenvalue of the fundamental. This difference in eigenvalues is commonly referred to as the eigenvalue separation of a harmonic and is a measure of the separation between the fundamental and harmonic mode.

At steady-state critical conditions, the eigenvalue of the fundamental is equal to 1.0 (critical) and, therefore, all harmonic modes must be subcritical (i.e., eigenvalues < 1.0). Subcritical modes will decay in time and are therefore not present at steady-state conditions. For a subcritical mode to be sustained, sufficient spatial reactivity feedback must be provided to overcome the eigenvalue separation. The best known type of spatial reactivity feedback associated with the harmonics is the result of changing xenon concentrations. For BWR stability, the existence of density

wave oscillations in the channels provides another type of spatial reactivity feedback that is capable of exciting the axial harmonics in the neutron flux. In addition, thermal-hydraulic oscillations in which channel inlet flows oscillate out-of-phase (e.g., inlet flow is increasing in one-half of the core while decreasing in the other half) are also capable of providing the spatial reactivity necessary to excite the radial and azimuthal harmonics (referred to as regional oscillations).

The existence of a sustained density wave oscillation in a BWR requires excitation of at least the first-order axial harmonic. The neutron flux at four axial locations in the core during a regional instability, as measured by the Local Power Range Monitors (LPRM), is shown in Figure A-12 along with the average neutron flux oscillation. The axial phase lag observed in the neutron flux oscillation is the result of the density wave oscillation and is related to the void transit time through the channel. When the normalized axial neutron flux distribution is examined as a function of time, relative to the steady-state distribution, the oscillating component of the axial harmonic can be observed, as shown in Figure A-13 for the data from Figure A-12. The change in normalized axial neutron flux distribution at two times during an oscillation period is shown in Figure A-13, where the two points differ by half a period. The same excitation of the axial harmonic is also predicted by the TRAC-G code for a core-wide oscillation as shown in Figure A-14.

In the radial and azimuthal directions, the higher harmonics can be excited. The most common harmonic oscillation that has been observed in worldwide BWRs is the first-order side-by-side oscillation. This is the first harmonic in the azimuthal direction. The mode is called a first-order mode, since there is only one line of symmetry across which the neutron flux associated with the harmonic mode changes sign (+ to -). Higher order modes have even larger eigenvalue separations (more subcritical) than most first order harmonics. As discussed in Section 6.2.2, the GE 3D BWR Simulator has been modified to calculate the harmonic mode power distributions and eigenvalues. The method has been validated by solving a benchmark problem consisting of a homogeneous rectangular parallelepiped reactor. The

analytical solution to this problem is known and is accurately predicted by the modified simulator.

The simulator has been used to predict the expected oscillation modes (harmonics) for a wide variety of core and fuel designs. The expected behavior of the eigenvalue separation as a function of core size has also been confirmed. As the core diameter is reduced, the buckling associated with the radial and azimuthal harmonics increases and therefore the eigenvalue separation also increases (i.e., more subcritical and therefore less likely to be sustained). The calculated variation of the eigenvalue separation for the axial and azimuthal harmonics as a function of the number of fuel assemblies (axial height constant) is shown in Figure A-15. These results are consistent with previous calculations of the harmonic modes performed during evaluations of Xenon instabilities in BWRs and PWRs.

From the results in Figure A-15, for core sizes less than 500 bundles, the axial harmonic is consistently predicted to be the lowest harmonic mode (i.e., highest eigenvalue). The azimuthal harmonic associated with regional oscillations is typically more than $2 in reactivity subcritical. For core sizes greater than 500 bundles, the eigenvalue separation of the axial and azimuthal harmonics are generally equal with no mode showing a specific preference. Therefore, for small cores (< 500 bundles) an additional factor that favors the core-wide oscillation mode is the large eigenvalue separation of the azimuthal mode. When coupled with the low channel decay ratios, the probability of regional oscillations is very low and, therefore, not an expected mode of oscillation for these plants.

A.3.3.3 Compliance to the MCPR Safety Limit

Section 6.0 of the main body of this report discusses the methodology used to evaluate the impact of oscillations on MCPR performance. The application of this methodology described in Section 6.3 is specific to those plants in which regional oscillations are the expected modes. The following sections discuss the application of the Section 6.0 methodology to those plants in which the core-wide mode is determined to be the expected mode of oscillation.

A.3.3.3.1 Core Wide Oscillations

Since the preferred mode of oscillation is assumed to be core-wide for cores with relatively "tight" inlet orifice diameters and small core size, this mode represents the design basis for Option I-D and the Flow Biased APRM Neutron Flux Trip System must be shown to provide an adequate level of protection against violating the MCPR Safety Limit. Therefore, analysis of the core-wide mode will follow the basic procedure outlined in Section 6.3 and use the same assumptions, except as stated below.

(1) Initial Conditions

For core-wide oscillations the assumptions and restrictions for initial conditions are the same as those of the generic methodology (Section 6.3.2.1). This includes initiation of the analysis with the plant operating at Technical Specification MCPR Operating Limits at the limiting rod line.

(2) Oscillation Contours

For a core-wide oscillation, the only mode of importance is the axial harmonic which is modeled directly as a higher order sinusoidal oscillation and the oscillation contour is constant for all LPRMs (i.e., same oscillation magnitude). The initial power distribution for a number of different points in the cycle is input to represent the variation in expected power distribution. This is the same as the approach used in the generic methodology (Section 6.3.2.2).

(3) Oscillation Growth Rate

The growth rate is the same as that assumed for the generic methodology (Section 6.3.2.2). The selected scenarios are evaluated against the APRM trip setpoint to determine the distribution of setpoint overshoots.

(4) Trip System Definition

For these analyses, the trip system is the Flow-Biased APRM Neutron Flux Trip. LPRMs will be assigned to their respective APRM channels and the trip setpoints will be determined for the appropriate final core flow during the oscillations.

(5) LPRM Failures

The LPRM failure assumptions are the same as those used in the generic methodology (Section 6.3.2.4).

(6) MCPR Performance

The treatment of MCPR variation with oscillation magnitude is the same as that used in the generic methodology (Section 6.3.2.5). However, since the change in CPR is a function of the oscillation mode, a correlation will be developed specifically for core-wide oscillations. An example of the difference in the CPR performance for a core-wide and regional oscillation with all other parameters held constant is shown in Figure A-16.

A.3.3.3.1.1 Example

The prototypical plant chosen for illustrating this analysis is the Duane Arnold Energy Center (DAEC), which is a 368 bundle, BWR/4, with 2.09 inch fuel inlet orifice diameter.

(1) Initial Conditions

In accordance with the methodology presented in the main report (Section 6.3.2.1), two initiating events are considered as precursors to the onset of thermal-hydraulically induced flux oscillations. The first is a trip of both recirculation pumps from full power conditions resulting in operation at natural circulation (approximately 30% rated flow). The second is a trip of both recirculation pumps from high power/low flow conditions indicative of startup operations. In accordance with the generic methodology, the

first initiating event will be assumed to occur 95% of the time and the second, the remaining 5% of the time.

For DAEC, initiating events were considered along two rod lines. The first is the 110% rod line, which is characteristic of operation in the extended operating domain. The second is the 100% rod line, which results in lower powers at natural circulation conditions and therefore requires larger oscillations to initiate a trip. The final power at natural circulation is based on a rod line which is conservative in the sense that it predicts powers at natural circulation conditions which are further away from the APRM trip setpoint than might be encountered during expected operation.

The MCPRs prior to the two initiating events were taken from the DAEC Technical Specifications. For rated power operation, the initial MCPR is 1.20, and for operation at high power/low flow during startup, the initial MCPR is 1.43. The MCPR increase due to flow coastdown along the given rod line was determined by use of the generic methodology.

(2) <u>Oscillation Contours</u>

The oscillation contour for core-wide oscillations is assumed constant throughout the core. The fundamental power distribution is based on a set of DAEC specific calculations at the conditions defined in Table A-3.

(3) <u>Oscillation Growth Rate</u>

In accordance with the generic methodology (Section 6.3.2.2), a range of growth rates was used to determine the distribution of first and second peak overshoots. For DAEC, the second peak distribution is chosen because of the Technical Specification scram insertion times (Figure A-17). For these oscillations, it is assumed that the period of the oscillation is two seconds. This value is consistent with observed oscillations in GE BWRs.

(4) Trip System Definition

A trip system definition consistent with the DAEC Flow-Biased APRM Neutron Flux Trip System is used. The standard single RPS channel failure is also assumed.

(5) LPRM Failures

The LPRM failure statistics used in the generic methodology are used for these analyses.

(6) MCPR Performance

A fit of CPR change as a function of oscillation magnitude which was based on a large core with "loose" inlet orifices (2.43 inch diameter) during core-wide oscillations (Figure A-16) was used in the example. It is postulated that the change in MCPR with oscillation magnitude is less severe for a plant like DAEC with "tight" inlet orifices and a more appropriate correlation will be used in final setpoint analyses.

(7) Results

The predicted MCPRs for both of the rod lines are shown in Table A-4.

A.3.3.3.2 Regional Oscillations

Since regional oscillations are unlikely in plants for which Option I-D is applicable, nominal assumptions will be made for the analysis of these oscillations. The behavior of the plant with respect to thermal limits will be characterized by expected values.

(1) Initial Conditions

For regional oscillations under Option I-D, the assumptions and restrictions are the same as those of the generic methodology (Section 6.3.2.1). However, instead of assuming the reactor begins at Technical

Specification MCPR limits, plant operating data will be used to define expected values of initial MCPR for full power and startup conditions. In addition, the final power at natural circulation will be based on a nominal rod line.

(2) Oscillation Contours

The approach is the same as for the generic methodology (Section 6.3.2.2).

(3) Oscillation Growth Rate

It is assumed that the growth rate is sufficiently slow that there is no significant overshoot. This is different than the generic methodology (Section 6.3.2.2). In general, the APRM response during a regional oscillation does not increase as fast as independent LPRM responses and, therefore, the overshoot is expected to be negligible. In addition, since regional oscillations are not expected to occur, a low growth rate is an appropriate assumption.

(4) Trip System Definition

For these analyses the trip system is the Flow-Biased APRM Neutron Flux Trip System. All APRM channels are assumed to be operational and no failures are assumed during the event. RPS failures are not common and in general result in more conservative conditions (i.e., channel trip).

(5) LPRM Failures

The number of failed LPRMs is the same as that used in the generic methodology (Section 6.3.2.4).

(6) MCPR Performance

The treatment of MCPR variation with oscillation magnitude is the same as that used in the generic methodology (Section 6.3.2.5). However, a

correlation will be specifically generated for the expected response of plants with tight fuel inlet orifices to properly account for the expected performance of the more stable channels.

A.3.3.3.2.1 Example

As was the case for core-wide oscillations, DAEC will be used for the analysis of regional oscillations.

(1) Initial Conditions

The initial conditions are the same as those used in the core-wide analysis with the following exceptions. Oscillations are most likely to occur at the intersection of the highest rod line and natural circulation flow and, therefore, analyses are only performed for the 110% rod line. The second is the use of anticipated MCPR values prior to the initiating event rather than Technical Specification limits. For DAEC, based on Cycle 10 data, the expected initial MCPR at rated power is 1.28. At startup conditions, an initial MCPR of 1.89 is assumed.

(2) Oscillation Contours

As discussed in Section B.3.3.2, negligible overshoot is assumed and the maximum signal value is equivalent to the trip setpoint.

(3) Trip System Definition

A trip system consistent with the DAEC Flow-Biased APRM Neutron Flux Trip System is used. It is assumed that all the channels are functional. The trip setpoint at natural circulation is 62% of rated.

(4) LPRM Failures

The LPRM failure statistics used in the generic methodology were used.

(5) MCPR Performance

The MCPR performance during oscillations is based on Curve 1 of Figure 6-11, which was generated for a regional oscillation in a large BWR with loose inlet orificing (2.43 inch diameter).

(6) Results

The predicted MCPRs for the maximum rod line are shown in Table A-5.

A.3.3.4 Sensitivity Studies

In Section A.3.3.3.1 examples were shown for a prototypical plant (DAEC). Since MCPR at full power may be expected to vary from plant to plant (and indeed from cycle to cycle for a given plant), a sensitivity study was performed to quantify the effect of variations in initial MCPR. The evaluations assumed regional oscillations occurred, similar to the results discussed in Section A.3.3.3.2.1.

Results were obtained for initial full power MCPRs from 1.28 to 1.48 to study the variation in final MCPR. For the two cases with MCPR of 1.38 and 1.48, an initial MCPR value of 2.209 was used for the low flow pre-event point. This value was used since it is more characteristic of a core with a full power MCPR in the 1.38 to 1.48 range. Operation along the 110% rod line was used for this analysis. The results from this study are shown in Figure A-18. As expected, more margin is provided for operation at higher initial MCPRs.

A.3.4 Technical Specifications

This section describes the philosophy of changes to Technical Specifications required for Option I-D. In general, changes to the APRM scram setpoints for protection against oscillations are not expected to be required. An LCO will be provided for exiting the region following an unplanned entry. An example of an exclusion region is shown in Figure A-1.

A map of the exclusion region as described in Figure A-1 will be included. An LCO for this specification will be provided to permit unrestricted operation outside the exclusion region. Action statements directing control rod insertion or flow increase in the event of unplanned entry into the region will also be included.

A.3.5 Operator Guidance

Operator guidance will be provided to avoid the exclusion region during normal operation. This guidance will direct the operator to increase core flow prior to exceeding a predetermined power level, which will be lower than the rod line associated with the setpoint of the exclusion region. The flow increase will result in a state where, given an increase in power to the target rod line, operation will remain outside the exclusion region. Similarly, the operator will be instructed to insert control rods during a controlled shutdown prior to decreasing flow below a predetermined value. The rod insertions will continue to a point below the rod line which bounds the lower end of the exclusion region. The operator will also be given appropriate direction to respond to flow reduction and LOFH events to avoid entering the exclusion region.

Additional direction for responding to upset conditions will be provided. For example, following a trip of both reactor recirculation pumps, the operator will be directed to reduce reactor power by control rod insertion to a point below the exclusion region before attempting to restart the recirculation pumps. In all cases of inadvertent entry into the exclusion region, the operator will be directed to take action to exit the region in a manner designed to minimize the impact on unfiltered neutron flux levels (e.g., limiting rate-of-flow increases, and prohibiting rod withdrawal within the exclusion region).

A.3.6 Operator Guidance

Operator guidance will be provided to avoid the exclusion region during startup and shutdown. This guidance will direct the operator to increase core flow prior to exceeding a predetermined power level, which would be lower than

the rod line associated with the exclusion region (and Stability APRM Flux Trip system actuation). The flow increase would result in a state where, given an increase in power to the target rod line, operation would remain outside the exclusion region. Similarly, the operator would be instructed to insert control rods during a controlled shutdown prior to decreasing flow below a predetermined value. The rod insertions would continue to a point below the rod line which bounds the lower end of the exclusion region. The operator will also be given appropriate direction to respond to flow reduction and loss of feedwater heating events to avoid entering the exclusion region.

Additional direction for responding to upset conditions, including alarm response, will be provided. For example, following a trip of both reactor pumps, the operator will be directed to reduce reactor power by control rod insertion to a point below the exclusion region before attempting to restart the recirculation pumps. In all cases of inadvertent entry into the exclusion region, the operator will be directed to take immediate action to exit the region a manner designed to minimize the impact on instantaneous neutron flux levels (e.g., limiting the rate of flow increases, prohibiting rod withdrawal).

A.4 OPTION III - LPRM BASED OSCILLATION POWER RANGE MONITOR

A.4.1 System Description

A.4.1.1 General Description

The Oscillation Power Range Monitor (OPRM) is a microprocessor-based monitoring and protection system which will detect a thermal-hydraulic instability, provide an alarm on small oscillation magnitudes, and initiate an Automatic Suppression Function (ASF) to suppress an oscillation prior to exceeding safety limits.

The OPRM is a Class 1E protection system which monitors the output of all installed Local Power Range Monitor (LPRM) detectors; the OPRM conforms to all applicable requirements of IEEE-279-1971. Four OPRM channels are provided. The OPRM channels provide inputs to trip logics which initiate an ASF.

The OPRM function is in parallel with, and independent of, the existing Class 1E and non-1E functions of the Power Range Neutron Monitoring (PRM) System. The OPRM does not affect the design bases for the existing PRM components, their calibration, or their separation schemes.

Each OPRM channel takes amplified LPRM signals from available locations in the PRM panels. These LPRM signals are grouped together such that the resulting OPRM response provides adequate coverage of expected oscillation modes. Each OPRM channel is comprised of a relatively large number of OPRM cells, where an OPRM cell represents a combination of several LPRMs (one to eight) in geometrically adjacent areas of the core. LPRM signals may be input to more than one OPRM cell within an OPRM channel. Individual LPRM signals may also be used directly to provide a more sensitive response to oscillations. The selection of the number of LPRMs (one to eight) combined and monitored as one input to the trip function involves a number of tradeoffs best evaluated during the detailed hardware design phase.

Each OPRM channel consists of a Class 1E microprocessor unit, either in a stand alone cabinet or as a subcomponent to the PRM cabinet. Several

microprocessor-based control units are commercially available for nuclear safety related service. Some modules, such as GE's NUMAC and Westinghouse's Eagle 21, have been reviewed by the NRC; SERs have been issued for their safety related use in other applications. As is typical for these types of devices, each OPRM channel performs safety-related and non safety-related functions, and interfaces with other 1E and non-1E systems, for which external and internal system isolation and fault tolerance per IEEE-384 are required. The software will be qualified in accordance with Regulatory Guide 1.152 requirements.

The above hardware descriptions, in conjunction with the sample detection algorithm defined in Appendix B, constitute the OPRM solution.

A.4.1.2 OPRM Suppression Functions

The OPRM function provides inputs to an ASF whose purpose is to suppress oscillations prior to exceeding the MCPR Safety Limit. The OPRM function would be installed and maintained as a Reactor Protection System (RPS) protection function or could provide inputs for a Select Rod Insert (SRI) function.

The SRI function, as presently used, is intended to reduce core power to less than the turbine bypass capacity, so that the unit avoids a scram during a load rejection event. The same function may be used to reduce power to suppress oscillations without a full scram. Verification of this function's capability to effectively terminate a thermal-hydraulic instability event is required for those plants which are considering this option.

For implementation as a RPS function, the four OPRM channels provide inputs, one for each RPS trip channel, for a one-out-of-two-taken-twice actuation trip logic. Reactor scram will therefore occur when at least one OPRM trip channel in each RPS trip system is in the tripped condition. For a Solid State RPS logic plant, the four OPRM channels provide inputs to the two-out-of-four channel to divisional logics (any two channels in trip will result in a reactor scram). For implementation as a SRI function, appropriate trip logic will be chosen to ensure high reliability of the ASF. For the purposes of this report, the OPRM channel assignments will be discussed relative to their assignments to a RPS function.

The instability trip function will not be used to initiate the Alternate Rod Insertion (ARI) system. The ARI trip function, installed per 10CFR50.62, is not required to provide an automatic scram function which is redundant and diverse to the instability trip function. ATWS/instability events have been shown to develop sufficiently slowly such that manual scram techniques, or manual ARI initiation, are sufficient to back up instability trip system failures. If the reactor becomes isolated from the main condenser, the ensuing transients provide the necessary ARI initiation from water level or reactor pressure signals.

A.4.1.3 LPRM to OPRM Assignments

The purpose of the OPRM design is to provide detection and suppression of expected oscillation modes in a BWR. This involves monitoring LPRMs throughout the core. Because the OPRM channels provide inputs to trip logics, appropriate channel redundancy and reliability are required. Additionally, the desire to avoid spurious actuations and allow for expected LPRM failures and bypasses must be incorporated into the system design. In general, all of these considerations were included in the original design of the Average Power Range Monitor (APRM) systems and therefore, where possible, the available LPRM assignment schemes between the APRM channels are used in the OPRM design. The following design decisions were made to satisfy the stated requirement:

Requirement	Design Feature
Detection of expected oscillation modes	All LPRMs (radially and axially) will be evaluated for providing input to the OPRMs. LPRM signals which are combined will be from geometrically adjacent areas.
Redundancy	Each LPRM string (four detectors) can provide input to at least two OPRM channels.

Requirement	Design Feature
Reliability Avoid spurious actuations Tolerant to LPRM bypass/failure	Up to eight LPRM signals will be combined to form the input for one OPRM "cell". This will minimize the possibility of a spurious actuation from a single LPRM. The response of any one OPRM cell can cause a trip of the associated OPRM channel.

Review of the LPRM to APRM assignments and separation schemes resulted in identification of three basic groups of plants; large cores (30 or more LPRM strings), small cores (20-24 LPRM strings) and the solid-state RPS design (Clinton BWR/6). The distribution of plants among the basic groups is shown in Table A-6. For the large cores, the LPRM to APRM assignments along with the use of unassigned LPRMs (Group A and B) provides a logical grouping of LPRMs into OPRM channels. LPRMs are assigned to OPRMs according to which APRM or unassigned LPRM group their input is provided. Each OPRM channel receives inputs from LPRMs which are assigned to either of two APRM channels or from LPRMs assigned to one APRM channel and LPRMs assigned to an LPRM group. These associations are shown in Table A-7. The four OPRM channels are designated as A, B, C and D, or when, referring to their association with RPS, it is more convenient to refer to the channels as A1, B1, A2, and B2, respectively.

The further combination of LPRM signals within an OPRM channel for the large core group also makes use of the natural distribution of LPRMs within the assigned channels. A basic diamond pattern was chosen to define the combinations of six to eight LPRMs within an OPRM channel, since the LPRM separation among the two RPS trip systems gives rise to this pattern. An example is shown in Figure A-19 of a diamond in which the LPRM strings at the four points of the diamond are in one RPS trip system and the LPRM string in the center of the diamond is in the other trip system. This LPRM association is known as an OPRM cell. The OPRM computes the response of the OPRM cell using two LPRM signals from each of the four LPRM strings at the corners of the diamond shown in Figure A-19. Where one of these corners falls on or outside the core periphery, the response is calculated using the remaining three LPRM

strings. Where more than one corner falls on or outside the periphery, a cell is not defined.

The specific LPRM assignments to each OPRM cell are also shown in Figure A-19, where a pattern is established that is repeated throughout the core. The four LPRM signals in each string are split into two groups (A/C and B/D, where A is the bottom LPRM in the string) where one group is assigned to OPRM channel A1 and the other group is assigned to A2 (B1 and B2 in the other trip system). For a given row or column of LPRM strings in the same trip system, this assignment is repeated (i.e., A/C to A1 and B/D to A2). The pattern is then reversed for alternating rows and columns (A/C to A2 and B/D to A1). This ensures that, axially, the LPRMs are uniformly distributed among the OPRM channels.

An example of the OPRM cells in OPRM channel A1/A2 for a core with 43 LPRM strings (764 fuel bundles/185 control rods) is shown in Figure A-20. Note that the two LPRM inputs from any one LPRM string that input to the same OPRM channel are used by two, three or four OPRM cells. This provides a significant amount of cell overlap, providing reliable detection of oscillations occurring in relatively small core regions. The overlap that exists between the OPRM channels assigned to the two RPS trip systems is shown in Figure A-21. The remaining assignment of LPRMs to OPRMs (OPRM B1/B2) for a core with 43 LPRM strings is shown in Figure A-22. For the 764 bundle core, there are 16 OPRM cells for each RPS A channel and 18 OPRM cells for each RPS B channel, with each cell consisting of six or eight LPRM inputs.

For small cores, the smaller number of LPRM strings results in each LPRM being shared among the RPS trip systems in the APRM system. No LPRM is shared between redundant trip channels. A similar concept can be expanded to the small core OPRM design where each LPRM string provides LPRM signals to two OPRM channels (one from trip system A and one from B). The assignments are staggered as in the large core design, except that for the A channels the staggering is along diagonals and for the B channels along rows. An example of this assignment scheme for a 20 LPRM string core is shown in Figure A-23. The

small core OPRM cell uses two LPRM signals from each of three selected LPRM strings, for six LPRM inputs each.

Alternatively for small cores, the LPRMs could be assigned to the OPRM channels with no sharing of LPRM signals between the two RPS trip systems or between redundant channels within a trip system. Each LPRM string would provide one LPRM input to each of the four OPRM channels. The axial distribution of these LPRMs between the OPRM channels would be uniform. Instead of a diamond pattern of assignments, the basic OPRM cell configuration would be that of a square, with each OPRM cell receiving four LPRM inputs. The square also provides a smaller geometrical spacing for combined LPRMs consistent with the smaller core size. For locations near the periphery where one corner of the square does not include an LPRM string, the OPRM cells would use the inputs from the remaining three LPRM strings. The basic assignments that would be used for this design are shown in Figure A-24. Final selection of the LPRM to OPRM assignments will be determined during the hardware design phase.

For the solid-state RPS design, each LPRM string provides one input to each of the four separate APRM channels. These APRM channels are then combined in a two-out-of-four, channel to divisional logic. This same separation could be maintained similar to the alternative small core OPRM assignment scheme, where each LPRM string would provide one input to each OPRM channel with four LPRMs combined in a cell. Because of the larger size of the solid state RPS plant (33 LPRM strings), this same separation scheme could be used with the diamond pattern of the large core group except that the center LPRM string would also provide an input to the OPRM cells for the diamond. This would result in a basic OPRM cell containing five LPRM signal inputs. Alternatively, the LPRM to OPRM assignments could be performed exactly as in the large core group. Final selection of the LPRM to OPRM assignments will be determined during the hardware design phase.

OPRM channel redundancy is evident from the LPRM assignments shown in Figures A-20, A-22, A-23 and A-24. Each LPRM string of four detectors feeds signals to at least two OPRM trip channels for complete regional redundancy within an RPS trip system. OPRMs associated with RPS trip systems A and B are

essentially equal in sensitivity due to the OPRM cell overlap in the large cores, and the pattern of shared LPRMs in the small cores. Monitoring multiple axial locations in each LPRM string provides an assurance of reliability and, together with the overlap of multiple OPRM cells, allows tolerance of multiple bypassed LPRMs in any individual OPRM cell.

For any size unit, a single LPRM string may also be used to provide a more localized OPRM cell of one or two detectors. Such use of fewer detectors per cell (the limit becomes one detector per cell) results in a system more sensitive to regional oscillations, but also more susceptible to false trips due to LPRM noise or malfunction. Depending on owner-specific weighting of pros and cons, an OPRM system using, for example, two LPRMs per cell may be desirable. An example of such an application to a 764 bundle unit is shown in Figure A-25. Half of the LPRM strings are assigned to RPS A (shown in the figure) and the remainder to RPS B (not shown). The A and C level LPRMs in this example are combined to make one cell that is input either to RPS channel A1 or channel A2. The B and D level LPRMs are then similarly combined for input to RPS channel B1 or channel B2.

A.4.1.4 OPRM Trip and Alarm Functions

Each OPRM channel will perform a real time analysis of LPRM signal responses. For each cell, the OPRM will compute a response based on the assigned LPRM inputs. Determination of peak-to-average values of LPRMs or OPRM cells is used to evaluate the magnitude of oscillations. Time averaging of responses is used to provide a time dependent baseline for normalizing oscillation magnitudes. The signal sampling and computation frequency will be well above the expected thermal-hydraulic oscillation frequency, essentially producing a continuous and simultaneous measurement of all defined OPRM cells. Any individual OPRM cell which satisfies the conditions and criteria of the trip algorithm will be sufficient to produce the OPRM channel trip function.

A description of a sample OPRM algorithm is provided in Appendix B. Many forms of algorithms may be used to distinguish oscillations from noise or other plant occurrences. Algorithms use the known frequency of oscillations (i.e., 0.3 to 0.7 Hz) to aid in screening out other signal variations such as

electrical spiking, 60 Hz noise, and reactor system and control system transients. Time delays for the channel trip introduced by the algorithm will be appropriately factored into the setpoint analyses as discussed in Section 6.3.

The OPRM alarm is designed to provide the operator with warning prior to a channel trip. This is accomplished using the same algorithm as discussed above for the channel trip with a lower setpoint.

A.4.1.5 OPRM Operating Bypasses

The OPRM will be operable in the power range of operation (i.e., Mode 1). However, historical data and operating experience have shown that the protective function is not required at low core power or at high core flow conditions. Because spurious actuation is always a concern for trip channels, it is appropriate to provide an operating bypass under conditions when the protective function is not required. LPRM signal noise at low power conditions and bistable core flow at high core flow conditions are examples of system inputs which may result in unnecessary trips if the OPRM is not bypassed.

The OPRM interfaces with the existing APRM Reactor Recirculation System Drive Flow Units, which are used for a high core flow system bypass. This bypass is removed under single loop operating conditions above the low power bypass setpoint.

The OPRM also interfaces with the APRM module outputs for a low power system bypass. As an alternative, the OPRM may directly compute core power from LPRM averages, with operator interface as necessary to calibrate the LPRM average to a plant heat balance.

Because this bypass is enabled by process conditions, the OPRM trip function becomes automatically functional given any plant operating transient of interest. These bypass functions are considered to be part of the Class 1E portion of the OPRM.

A.4.1.6 LPRM Bypasses

The OPRM communicates with the LPRM amplifier cards such that the OPRM can detect and account for bypassed LPRMs. The OPRM cell is considered to be active when the number of valid LPRM signals is greater than or equal to a number to be determined during the design phase, which may range from one (for the individual LPRM OPRM example) to three or four (for the eight LPRMs per OPRM cell design).

LPRM signals which do not exceed a minimum value may contain process or signal noise components which may affect the OPRM cell response. The OPRM algorithm may bypass LPRM signals which fall below system requirements, or may normalize the LPRM oscillation peak magnitude to a core wide average. The adjustments are automatically enabled or removed based on process conditions.

A.4.1.7 Operator Interface

No direct control interface is required for the safety-related OPRM trip function. However, the OPRM has a wide range of alarm, display, and other capabilities. The specific non-1E functions will be defined in the engineering development of the OPRM function, device, and software. No credit is taken for operator interface in achieving the system functional performance requirements.

A.4.2 <u>Licensing Approach</u>

A.4.2.1 Overview

A.4.2.1.1 GDC-12 Compliance

The OPRM is designed to automatically detect and suppress stability related neutron flux oscillations which could result in conditions exceeding the MCPR Safety Limit. Reliability is enhanced by using highly redundant OPRM cells providing input to a safety grade trip system described in Section A.4.1. These design features ensure compliance with GDC-12 .

Analytical MCPR Safety Limit compliance will be demonstrated for all expected modes of GE BWR neutronic/thermal-hydraulic neutron flux oscillations (defined in Section 6.1). Although the OPRM system is expected to be capable of responding to other postulated modes of flux oscillations, demonstrating MCPR compliance for such modes is not necessary, for the reasons described in Section 6.1.

A.4.2.1.2 Design/Licensing Philosophy

The OPRM uses a large fraction of, if not all, operable LPRMs. Because they are evenly distributed throughout the reactor, the fission chamber LPRMs are capable of immediately responding to any neutron flux oscillations capable of creating an MCPR concern.

The overall design philosophy of the OPRM system is to generate a trip signal at a sufficiently low oscillation amplitude such that margin to the MCPR Safety Limit is provided. Operating experience with core-wide and regional oscillations shows that LPRMs readily respond to oscillations, and the OPRM system, consisting of many cells, will readily respond as well.

LPRMs also readily respond to a wide variety of normal operating maneuvers and expected events, including direct electrical or mechanical malfunctions of the LPRM detector, seals, cable, or amplifier. Individual LPRMs are also subject to electrical interference, and can respond very strongly to nearby control rod motion. For these reasons, the OPRM system may use multiple LPRMs as a means of maintaining a strong response to a neutron flux oscillation while minimizing the susceptibility to false signals associated with a single LPRM, or will utilize a detection algorithm designed to minimize the susceptibility to false signals associated with a single LPRM.

A.4.2.1.3 Setpoint Basis

The OPRM oscillation recognition algorithm is intended to discriminate between true stability-related neutron flux oscillations and other flux variations that may be expected during plant operation. The algorithm design has two primary objectives. The first is to provide a sufficiently low

amplitude trip setpoint such that minimum reliance on analysis is required to demonstrate MCPR margin during a postulated neutron flux oscillation. Second, the algorithm must be capable of identifying stability-related neutron flux oscillations and discriminating against false signals from other expected plant evolutions. This design objective is essential for maintaining reliable power operation while simultaneously minimizing unnecessary challenges to the suppression function.

Extensive evaluation of operating plant data is required to determine the combination of algorithm and OPRM setpoints which meet the design objectives. Once the algorithm is defined and the minimum amplitude for the OPRM setpoint is determined, confirmatory analysis to demonstrate that the OPRM design provides margin to the MCPR Safety Limit will be performed for expected oscillation modes using the Section 6.0 methodology. The algorithm and setpoints discussed in Appendix B have been evaluated using these methods. The results of the example calculations are presented in Section 6.3.3.

The final algorithm/setpoint design may be subjected to in-plant testing, with the plant trip function disabled, to ensure the design adequately discriminates against expected plant transient responses. This testing will be done on a lead plant or lead plant-type basis or could be a part of the normal testing program for each unit installing an OPRM system. During this testing period, current plant procedures based on the Interim Corrective Actions of Reference 1 will be retained.

A.4.2.2 Oscillation Types and Modes

The OPRM is capable of responding to the expected modes of BWR stability-related oscillations. The design basis of the OPRM system shall be to generate a trip signal during oscillations of sufficiently low amplitude to provide margin to the MCPR Safety Limit for all expected modes of BWR oscillations.

These expected modes are core-wide, first-order side-by-side, and first-order precessing. Section 6.1 describes the expected modes and the basis for limiting consideration to these modes.

A.4.2.3 OPRM Design Features

A.4.2.3.1 Trip Function Bypass Setpoints

The OPRM trip function is automatically bypassed below approximately 30% power (low power bypass) and above approximately 60% core flow (high flow bypass). The reasons for this are (1) to allow selected maintenance and calibration activities to be performed during normal unit operation without creating channel trips, and (2) to prevent unnecessary false OPRM trips from potential operating events, LPRM instrument malfunctions, electrical interference, etc. Additionally, the low power bypass ensures that LPRM signals of sufficient magnitude are available so that most OPRM cells will be operable (Section A.4.2.3.2 below).

The automatic bypass features do not affect OPRM system operability requirements. In the run mode with core flow above the high flow bypass or power less than the low power bypass, the OPRM is operable, but the trip function is bypassed. The trip function is automatically enabled upon operation above the low power bypass and below the high flow bypass. Automatic bypass features are common in RPS design. Examples include the turbine control valve fast closure scram bypass when steam flow is less than a specified value and the MSIV closure scram in startup or hot standby modes which is bypassed at low pressure.

These setpoints have been selected to conservatively bound the regions of the power/flow map where an instability may occur. They are based on GE-BWR world-wide operating experience, which has not resulted in any instabilities below 40% power or above 45% core flow. The setpoints are experience based, and are very conservative. Additionally, the setpoints can be confirmed using the Section 5.1 methodology.

A.4.2.3.2 LPRM Bypass on Low Signal Level

To ensure that only valid LPRM signals are utilized in the OPRM, low reading LPRMs are automatically excluded from each OPRM cell's calculated response. The setpoint will be at approximately 5% to 10% of scale, which is consistent

with a similar bypass built into existing Rod Block Monitor (RBM) circuitry. A very low-reading LPRM is normally a bypassed or malfunctioning LPRM which would not respond to changes in neutron flux and, therefore, must not be included as an OPRM input signal.

A.4.2.3.4 Bypassed LPRM Detector Basis

In the current APRM system, it is not uncommon to have some LPRMs bypassed due to an electrical or mechanical failure of one of the detector's components. When a failure is diagnosed, plant personnel manually bypass the LPRM, resulting in a downscale signal which automatically excludes it from being used in the OPRM system.

OPRM operability is not significantly impacted by such bypassed LPRMs, and no changes to existing LPRM/APRM Technical Specifications are required. This is because:

(1) The highly redundant and low minimum number of required LPRMs in the OPRM cell design ensures that large numbers of cells will remain operable, even with very large numbers of LPRMs bypassed.

(2) Approximately 75% (100% for BWR/6) of the LPRMs are subject to existing Technical Specification operability requirements, which ensure that at least half of these LPRMs are operable. Technical Specifications also require at least two LPRMs per axial level for each APRM to be operable, which essentially mandates even more LPRMs in service because failure distributions are not always evenly distributed per each axial level inside ach APRM channel.

(3) Actual reliability of LPRMs is very high relative to what would create an OPRM reliability concern. Because only a fraction of the available LPRMs is needed for an OPRM cell to be operable, as few as 50% or less of the core's LPRMs could enable all OPRM cells to be operable. Given the significant redundancy of the cell configuration of each OPRM channel, even half of the cells being inoperable per trip channel would not significantly alter its capability.

(4) Actual LPRM reliability at the worst points in a cycle is estimated to be better than 75%, which is far beyond what is needed. Because LPRMs are required for efficient monitoring of the reactor's power distribution, each utility has a strong incentive to maintain high LPRM reliability.

To verify that bypassed LPRMs will not create an OPRM operability concern, a confirmatory statistical evaluation using the Monte Carlo approach described in Section 6.3.2.4 will be performed. The evaluation will consider the actual number of LPRMs used per OPRM cell and plant or plant-type specific OPRM configurations. An example calculation is provided in Section 6.3.3.

A.4.2.3.5 Oscillation Recognition

As described in Section A.3.1.4, the OPRM features an algorithm that is able to discriminate between stability related neutron flux oscillations and other neutron flux variations that are expected to occur in the plant. The algorithm monitors the OPRM cell responses and provides a trip signal if an oscillation with sufficient magnitude is detected. Two examples of such an algorithm are provided in Appendix B with the algorithm described in Section B.1 as the specific algorithm for this option. The algorithm described in Section B.2 is an example of a viable alternative. The detection and suppression licensing methodology described in Section 6.0 explicitly accounts for the oscillation magnitudes possible with chosen algorithm setpoints, and the delays and overshoots that may occur prior to oscillation termination by the ASF.

A.4.2.3.6 OPRM Alarm Basis

The OPRM alarm circuit will:

(1) Provide control room warning of an oscillation prior to an OPRM trip.

(2) Not alarm during expected plant normal operation and expected maneuvers.

A.4.2.4 Operational Considerations

Because the OPRM design and licensing basis is to automatically detect and suppress expected modes of neutronic/thermal-hydraulic oscillations, no operating restrictions are required. Consistent with systems which provide alarms or safety functions, operating procedures will be generated for responding to OPRM alarms and half or full trip conditions. The generically expected operator actions for these conditions are discussed in Section A.4.5.

It is expected that plant procedures will be developed which alert operators to conditions that could result in oscillations, and describe the appropriate actions to manually suppress them should they occur. The emphasis of such procedures will be scram avoidance, since safe operation of the reactor will be assured by the ASF.

A.4.3 Methodology Application

A.4.3.1 Oscillation Algorithm

The approach to establishing the OPRM setpoints is to first determine an acceptably low setpoint such that expected plant evolutions will not result in an OPRM system trip, and then confirm that the setpoint provides margin to the MCPR Safety Limit. Therefore, evaluation of plant operating data against potential trip algorithms is a necessary step in the determination of the setpoints. Basic characteristics of oscillations are already known from test data and operating experience. Sample trip algorithms have been conceptually designed based on this information, and are described in Appendix B. Neutron flux signals generated by a simplified point kinetics model (including the effects of random noise) and digitally recorded plant data during various maneuvers are analyzed to determine the margin to trip for expected plant maneuvers without oscillations. Desired trip margins will be established based on trip avoidance and previous experience with other trip systems.

The analytical methodology used to demonstrate that the trip algorithm provides margin to the MCPR Safety Limit is described in detail in Section 6.0. The methodology simulates LPRM and OPRM cell responses to oscillations of

various growth rates. These simulations explicitly account for delays associated with oscillation recognition by the algorithm and delay time associated with the ASF. Sections 6.2 and 6.3 detail how these delays and the algorithm setpoints themselves are related to the MCPR performance of the limiting bundles, and how uncertainties associated with oscillation recognition are accounted for. Digitally recorded data from actual BWR instability events will also be evaluated using the trip algorithms. This evaluation will further confirm that the trip algorithm can readily identify the occurrence of oscillations.

A.4.3.2 Detection and Suppression Oscillation Methodology

Any detection and suppression system requires a method for relating the LPRM responses during expected modes of oscillations to the MCPR of the limiting fuel assemblies. This basic oscillation methodology for the OPRM is described in detail in Section 6.2. Application of this methodology, described in Section 6.3, considers the various uncertainties and initial conditions associated with defining the oscillation response of limiting bundles. Section 6.4 describes how this generic methodology will be applied to determine plant specific setpoints. Appropriate parameters will be reviewed to ensure that the setpoint remains adequate for future reload cycles.

A.4.3.3 Trip Bypass on Low Power/High Flow

As described in Section A.4.2.3.1, the OPRM trip function is bypassed at high core flows and low power levels based on conservative application of oscillation operating experience. The regional exclusion boundary definition methodology, described in Section 5.1, will be used to confirm that the high flow and low power trip bypass setpoints are appropriate.

A.4.4 Technical Specification Implementation Philosophy

The OPRM instrumentation provides inputs to RPS or SRI functions and, therefore, the Technical Specifications involving the OPRM system will be similar to the existing RPS instrumentation specifications. The specifications will provide detail involving the minimum number of channels required per trip

system. The LPRM operability requirements for the APRMs will be sufficient and, therefore, no additional LPRM requirements for the OPRM will be necessary. The specification will also provide information regarding applicability and actions required if the requirements of the OPRM specification are not met. Setpoints and surveillance requirements similar to other RPS instrumentation will be provided such that proper testing and operability can be performed/determined. Surveillance frequencies and allowable out-of-service times will be developed consistent with the reliability of the microprocessor design. Notations regarding bypassing of the system and other features will also be included. With this system installation, the requirement for actions denoted in NRC Bulletin 88-07 and SIL 380 will be eliminated.

A.4.5 Operator Guidance

The OPRM system will generate alarms, half scram or SRI actuations, or full actuations as required. Operator interface with the OPRM system will be through front panel annunciators, CRT displays or other information presentation systems. Operator response to OPRM system alarms and channel trips (half or full) will be similar to operator response to other alarms and trips and will be supported by appropriate training. The following recommendations are provided by the BWROG regarding operator response to OPRM system alarms and trips.

> The operator will be required to investigate the cause of the alarm or channel trip. Upon determination that the OPRM system has contributed to the alarm or channel trip, the operator will proceed to locate the area of oscillation through the use of back panel CRT displays, flashing downscale/upscale LPRM lights and alarms, review of LPRM hardwire displays via rod selection, or other optional displays. Having determined the cause and area of concern relating to the OPRM system alarm or channel trip, the operator will take proper actions to mitigate thermal-hydraulic instabilities if evidence of such instabilities exists. Operator action to suppress thermal-

hydraulic instabilities may include insertion of CRAM rods and/or an increase in core flow as deemed appropriate.

In addition, for a trip of both reactor recirculation pumps, the operator will assure the reactor is stable before attempting to restart the recirculation pumps. All plants are expected to provide operator training/guidance in oscillation prevention and mitigation for scram or SRI avoidance reasons.

A.5 OPTION III-A - LPRM BASED SYSTEM

A.5.1 System Description

A.5.1.1 General Description

The Local Power Range Monitor (LPRM) based system (LBS) is a microprocessor-based monitoring and protection system capable of detecting a thermal-hydraulic instability, providing an alarm on low oscillation magnitude, and initiating an Automatic Suppression Function (ASF) to suppress an oscillation prior to exceeding safety limits.

The LBS is a Class 1E protection system which monitors the output of selected LPRM detectors. The LBS conforms to all applicable requirements of IEEE-279-1971. LBS channels are provided in a one-to-one correspondence to the Average Power Range Monitor (APRM) channels. The LBS provides inputs to trip logics which initiate an ASF.

The LBS function is in parallel with, and independent of, the existing Class 1E and non-1E functions of the Power Range Neutron Monitoring (PRM) System. The LBS does not affect the design bases for the existing PRM components, their calibration, or their separation schemes.

The LBS utilizes amplified LPRM signals for selected LPRM detectors from available locations in the PRM panels. These LPRM signals are chosen from the associated APRM channels such that the resulting LBS channel response provides adequate coverage of expected oscillation modes. Each LBS channel comprises up to eight LPRM inputs from one APRM channel, representing geometrically diverse (one per octant) regions of the reactor core. In this way, each LBS channel has the capability to detect expected oscillation modes.

Each LBS channel consists of a Class 1E microprocessor unit fabricated on a circuit board sized to fit in an LPRM flux amplifier card slot in the APRM chassis. The LBS card supports up to eight LPRM inputs, an APRM flux input, and an APRM (drive) flow input.

The LBS does not represent the first application of microprocessors in nuclear service. Several microprocessor based control units are commercially available for nuclear safety-related service. Some modules, such as GE's NUMAC and Westinghouse's Eagle 21, have been reviewed by the NRC; SERs have been issued for their safety related use in other applications. As is typical for these types of devices, these systems perform safety-related and non safety-related functions, and interface with other 1E and non-1E systems, for which external and internal system isolation and fault tolerance per IEEE-384 are required. The software will be qualified in accordance with Regulatory Guide 1.152 requirements.

A.5.1.2 Suppression Function

The LPRM-based system provides inputs to an ASF whose purpose is to suppress oscillations prior to exceeding the MCPR Safety Limit. It may be installed and maintained as a Reactor Protection System (RPS) trip function or may provide inputs for a Select Rod Insert (SRI) function.

The SRI function, as presently used, is intended to reduce core power to less than the turbine bypass capacity, so that the unit avoids a scram during a load rejection event. The same function may be used to reduce power to suppress oscillations without a full scram. Verification of this function's capability to effectively terminate a thermal-hydraulic instability event will be required for those plants considering this option.

For implementation as a RPS function, the LPRM-based system provides input to the appropriate RPS trip channel(s) consistent with the function and number of channels being added. The LBS channels tie into the existing APRM trip logic, providing the fourth APRM trip function (the others being inoperable, upscale flux, and upscale flow-biased neutron or thermal flux). Reactor scram will therefore occur when at least one channel in each RPS trip system is in the tripped condition. For a solid state RPS logic plant, any two RPS channels in trip will result in a reactor scram. For implementation as a SRI function, appropriate trip logic will be chosen to ensure high reliability of the ASF. For the purposes of this report, channel assignments will be discussed relative to a RPS function.

The instability trip function will not be used to initiate the Alternate Rod Insertion (ARI) system. The ARI trip function, installed per 10CFR50.62, is not required to provide an automatic scram function which is redundant and diverse to the instability trip function. ATWS/instability events have been shown to develop sufficiently slowly such that manual scram techniques, or manual ARI initiation, are sufficient to back up instability trip system failures. If the reactor becomes isolated from the main condenser, the ensuing transients will provide the necessary ARI initiation from water level or reactor pressure signals.

A.5.1.3 LPRM Assignments

The purpose of the LPRM-based system is to provide detection and suppression of expected oscillation modes in a BWR. This is achieved by monitoring LPRMs that are representative of flux levels in core regions that are most responsive in the expected oscillation modes. The LBS does not utilize a full complement of LPRM inputs. The choice of LPRM inputs for the LBS ensures adequate response to expected oscillation modes.

Because the LBS provides inputs to trip logics, channel redundancy and reliability are required. Additionally, protection from spurious actuations and allowances for expected LPRM failures and bypasses must be incorporated into the system design. In general, all of these considerations were included in the original design of the Average Power Range Monitor (APRM) systems and, therefore, where possible, the available LPRM assignment schemes between the APRM channels are used. The following system design features are incorporated to satisfy the stated design requirements:

Requirement	Design Feature
Detection of expected oscillation modes	LPRMs at the most responsive core locations during expected oscillation modes are selected.
Redundancy	A sufficient number of LPRM inputs (up to eight per LBS channel) are selected. LBS redundancy remains the same as APRMs.
Reliability Avoid spurious actuations Tolerant to LPRM bypass/failure	A total of 32, 48, or 64 LPRMs (one LPRM from each core octant per LBS channel) will be monitored. A minimum of eight operable LPRMs can assure adequate core stability protection. The eight LPRMs must be distributed such that each RPS channel monitors two LPRMs. The two LPRMs monitored by an RPS channel cannot be in the same core octant or in mirror-symmetric core octant pairs. The use of a detection algorithm (e.g. Appendix B) to process several individual LPRM signals assures a reliable trip function and high sensitivity to oscillations, while minimizing spurious actuations due to a single LPRM malfunctions.

Sample octant boundaries for a small-core BWR are displayed in Figure A-26. This choice of octant boundaries minimizes the impact of bypassed or failed LPRMs on LBS sensitivity, since the strongest response during expected oscillation modes occurs on the octant boundaries (i.e., the immediately-adjacent octant will show a comparable response during expected oscillation modes). Alternate octant boundaries are displayed in Figure A-27 (rotated 22.5° from Figure A-26) which are chosen to provide the strongest response of particular LBS channels during expected oscillation modes (i.e., boundaries are chosen such that the strongest response during expected oscillation modes occurs midway between boundaries).

Sample LPRM assignments to LBS channels for small cores using the boundary strategy of Figure A-26 are shown in Figure A-28. For purposes of operability determination, LPRM string 16-17 (arbitrarily) monitors the west-southwest octant and string 32-33 monitors the south-southwest octant. Sample LPRM assignments for large cores using a similar boundary strategy are shown in Figure A-29.

The LPRM assignment strategy consists of selecting one LPRM in each core octant for each of the LBS channels. LPRMs are selected based on ensuring sensitivity to oscillations while minimizing the impact of LPRM failures. The ability to cause an ASF is assured for oscillations confined to regions as small as a core octant. In this assignment strategy, every LBS channel receives an LPRM input from each core octant; consequently, the reliability and redundancy of the LBS channels remain consistent with the APRM operability requirements.

The design objective, therefore, is to monitor 32, 48, or 64 LPRM inputs individually, one from each core octant for each of the four, six, or eight LBS channels. The licensing basis of the LBS solution, however, allows for a minimum of eight operable LPRM inputs, two from each RPS channel. The two LPRMs in an RPS channel cannot be in the same core octant or in mirror-symmetric core octant pairs. By applying mirror symmetry across the core center, in effect, all eight core octants are monitored by only four octants in each RPS trip system for the expected modes of oscillation.

A.5.1.4 Trip and Alarm Functions

Each LBS channel performs a real-time analysis of LPRM signal responses. Determination of peak-to-average values of LPRM signals is used to evaluate the magnitude of oscillations. Time-averaging of responses is used to provide a time-dependent baseline for normalizing oscillation magnitudes. The signal sampling and computational frequency are well above the expected oscillation frequency, essentially producing a continuous and simultaneous measurement of all LPRM inputs to the LBS channel. Any individual LPRM signal input which satisfies the conditions and criteria of the trip algorithm will be sufficient to produce a LBS channel trip.

A description of a sample algorithm is provided in Appendix B. Many forms of algorithms may be used to distinguish oscillations from noise or other plant occurrences. Algorithms use the known frequency of oscillations (i.e., 0.3 to 0.7 Hz) to aid in screening out other signal variations such as electrical spiking, 60 Hz noise, and reactor system and control system transients. Time delays for the channel trip introduced by the algorithm will be appropriately factored into the setpoint analyses as discussed in Section 6.3.

The LBS alarm is designed to provide the operator with warning prior to a channel trip. This may be accomplished using the same algorithm as discussed above for the channel trip with a lower setpoint.

A.5.1.5 Operating Bypasses

The LPRM-based system will be operable in the power range of operation (i.e., Mode 1). However, historical data and operating experience have shown that the protective function is not required at low core power or at high core flow conditions. Because spurious actuation is always a concern for trip channels, it is appropriate to provide an operating bypass under conditions when the protective function is not required. LPRM signal noise at low power conditions and bistable core flow at high core flow conditions are examples of system inputs which may result in unnecessary challenges if these systems are not bypassed.

The system interfaces with the existing APRM Reactor Recirculation System Drive Flow Units, which are used for a high core flow system bypass. This bypass is removed under single loop operating conditions above the low power bypass setpoint. The system also interfaces with the APRM module outputs for a low power system bypass.

Because this bypass is enabled by process conditions, the trip function becomes automatically functional given any plant operating transient of interest. The bypass functions are considered to be part of the Class 1E portion of the solution.

A.5.1.6 LPRM Bypasses

The LPRM-based system communicates with the LPRM amplifier cards such that it can detect and account for (in a limited sense) bypassed LPRMs. Minimum LPRM operability requirements for ensuring LBS operability (consistent with Section A.5.1.3) will be administratively controlled by Technical Specifications.

LPRM signals which do not exceed a minimum value may contain process or signal noise components which may affect system response. The detection algorithm may bypass LPRM signals which fall below system requirements, or may normalize the LPRM oscillation peak magnitude to a core-wide average. The adjustments will be automatically enabled or removed based on process conditions.

A.5.1.7 Operator Interface

No direct control interface is required for the LPRM-based system safety-related trip functions. However, the trip functions have a range of alarm, display, and other capabilities. The specific non-1E functions will be defined in the engineering development of the function, device, and software. No credit is taken for operator interface in achieving the system functional performance requirements.

A.5.2 Licensing Approach

A.5.2.1 Overview

A.5.2.1.1 GDC-12 Compliance

The LPRM-based system will be designed to automatically detect and suppress stability-related neutron flux oscillations which could result in conditions exceeding the MCPR Safety Limit. Reliability will be enhanced by using a variety of LPRM inputs to redundant safety grade LBS trip channels described in Section A.5.1. These design features ensure compliance with GDC-12.

Analytical MCPR Safety Limit compliance will be demonstrated for expected modes of GE BWR neutronic/thermal-hydraulic neutron flux oscillations (defined in Section 6.1). Although the LBS is expected to be capable of responding to other postulated modes of flux oscillations, demonstrating MCPR compliance for such modes is not necessary, for the reasons described in Section 6.1.

A.5.2.1.2 Design/Licensing Philosophy

By using a geometrically-diverse selection of operable LPRMs, the LBS uses the best available instrumentation for detecting an oscillation. Because they are distributed throughout the reactor, the fission chamber LPRMs are capable of immediately responding to any neutron flux oscillations capable of creating a MCPR concern.

The overall design philosophy of the LBS is to generate a trip signal at a sufficiently low oscillation amplitude such that margin to the MCPR Safety Limit is provided. Operating experience with core-wide and regional oscillations shows that LPRMs readily respond to oscillations, and the LBS will readily respond as well.

LPRMs also readily respond to a wide variety of normal operating maneuvers and expected events, including direct electrical or mechanical malfunctions of the LPRM detector, seals, cable or amplifier. Individual LPRMs are also subject to electrical interference, and can respond very strongly to nearby control rod motion. For these reasons, the LBS detection algorithm will be designed to minimize the susceptibility to false signals associated with a single LPRM.

A.5.2.1.3 Setpoint Basis

The LBS detection algorithm is intended to discriminate between true stability-related neutron flux oscillations and other flux variations that may be expected during plant operation. The algorithm design has two primary objectives. The first is to provide a sufficiently low amplitude trip setpoint, such that minimum reliance on analysis is required to demonstrate MCPR margin during a postulated neutron flux oscillation. Second, the

algorithm must be capable of identifying stability-related neutron flux oscillations and discriminating against false signals from other expected plant evolutions. This design objective is essential for maintaining reliable power operation, and simultaneously minimizes unnecessary challenges to the suppression function.

Extensive evaluation of operating plant data is required to determine what combination of algorithm and LBS setpoint will meet the design objectives. Once the algorithm is defined and the minimum amplitude for the LBS setpoint is determined, confirmatory analysis to demonstrate that the LBS design provides margin to the MCPR Safety Limit will be performed for expected oscillation modes using the Section 6.0 methodology.

The final algorithm/setpoint design may be subjected to in-plant testing, with the plant trip function disabled, to ensure that the design adequately discriminates against expected plant responses. This testing could be done either on a lead plant or lead plant-type basis, or could be a part of the normal testing program for each unit installing an LBS.

During this testing period, current plant procedures based on the ICAs of Reference 1 will be retained.

A.5.2.2 Oscillation Types and Modes

The LBS is expected to be capable of responding to any reasonably postulated mode of BWR stability-related oscillations. The design basis of the LBS shall be to generate a trip signal during oscillations of sufficiently low amplitude to provide margin to the MCPR Safety Limit for expected modes of BWR oscillations. These expected modes are core-wide, first order side-by-side, and first order precessing. Section 6.1 describes the expected modes and the basis for limiting considerations to these modes.

A.5.2.3 LBS Design Features

A.5.2.3.1 Trip Function Bypass Setpoints

The LBS trip function will be automatically bypassed below approximately 30% power (low power bypass) and above 60% core flow (high flow bypass). The reasons for this are (1) to allow selected maintenance and calibration activities to be performed during normal unit operation without creating channel trips, and (2) to prevent unnecessary false LBS trips from potential operating events, LPRM instrument malfunctions, electrical interferences, etc. Additionally, the low power bypass ensures that LPRM signals of sufficient magitude are available so that meaningful LBS processing can be performed (see Section A.5.2.3.2).

The automatic bypass features do not affect LBS operability requirements. In the run mode with core flow above the high flow bypass or power less than the low power bypass, the LBS is operable, but the trip function is bypassed. The trip function will automatically be enabled upon operation above the low power bypass and below the high flow bypass. Automatic bypass features are common in RPS design. Examples include the turbine control valve fast closure scram bypass when steam flow is less than a specified value and the MSIV closure scram in startup or hot standby modes which is bypassed at low pressure.

These setpoints have been selected to conservatively bound the regions of the power flow map where an instability may occur. They are based on GE-BWR world-wide operating experience, which has not resulted in any instabilities below 40% power or above 45% core flow. The setpoints are experienced based, and are very conservative. Additionally, the setpoints can be confirmed using the Section 5.1 methodology.

A.5.2.3.2 LPRM Bypass on Low Signal Level

To ensure that only valid LPRM signals are utilized in the LBS, low reading LPRMs are automatically excluded from the LBS calculated response. The setpoint will be at approximately 5% to 10% of scale, which is consistent with

a similar bypass built into existing Rod Block Monitor (RBM) circuitry. A very low reading LPRM is normally a bypassed or malfunctioning LPRM which would not respond to changes in neutron flux and, therefore, must not be included as an LBS input signal.

A.5.2.3.3 Bypassed LPRM Detector Basis

In the current APRM system, it is not uncommon to have some LPRMs bypassed due to an electrical or mechanical failure of one of the detector's components. When a failure is diagnosed, plant personnel manually bypass the LPRM, resulting in a downscale signal which automatically excludes it from being used in the LBS.

LBS operability is not significantly impacted by such bypassed LPRMs since the failures occur randomly and only a small fraction of the monitored LPRMs are needed to satisfy the system requirements. However, minor changes to existing LPRM/APRM Technical Specifications are required to ensure operability of the LBS when LPRMs are bypassed.

The design basis of the LBS assumes four operable LPRMs for each RPS trip system (A and B). Through the use of mirror symmetry (based on the expected oscillation modes discussion of Section 6.1 and core loading and control rod patterns in use at all BWRs), the minimum LPRM operability requirement can be met with two operable LPRMs in each RPS channel (i.e., A1, A2, B1 and B2). LPRMs in the same RPS channel cannot be located in the same core octant or mirror-symmetric core octant pair. The Technical Specifications will provide administrative controls to ensure that this minimum operability requirement is met.

To assess the impact of bypassed LPRMs on LBS operability, a confirmatory statistical evaluation using the Monte Carlo approach described in Section 6.3.2.4 will be performed. The evaluation will consider the actual number of LPRMs in plant or plant-type specific LBS configurations.

A.5.2.3.4 Oscillation Recognition

As described in Section A.5.1.4, the LBS features an algorithm that will be able to discriminate between stability-related neutron flux oscillations and other neutron flux variations that are expected to occur in the plant. The algorithm will monitor the LPRM responses and provide a trip signal if an oscillation with sufficient magnitude is detected. Two examples of such an algorithm are provided in Appendix B with the algorithm described in Section B.1 as the specific algorithm for this option. The algorithm described in Section B.2 is an example of a viable alternative. The detection and suppression licensing methodology described in Section 6.0 explicitly accounts for the oscillation magnitudes possible with chosen algorithm setpoints, and the delays and overshoots that may occur prior to oscillation termination by the ASF.

A.5.2.3.5 LBS Alarm Basis

The LBS alarm circuit will:

(1) Provide control room warning of an oscillation prior to an LBS trip.

(2) Not alarm during expected plant normal operations and expected maneuvers.

A.5.3 <u>Operational Considerations</u>

Because the LBS design and licensing basis is to automatically detect and suppress expected modes of neutronic/thermal-hydraulic oscillations, no operating restrictions are required. Consistent with systems which provide alarms or safety functions, operating procedures will be generated for responding to LBS alarms and half or full trip conditions. The generically expected operator actions for these conditions are discussed in Section A.5.5.

It is expected that plant procedures will be developed which alert operators to conditions that could result in oscillations, and describe the appropriate actions to manually suppress them should they occur. The emphasis

of such procedures will be scram avoidance, since safe operation of the reactor will be ensured by the ASF.

A.5.4 Methodology Application

A.5.4.1 Oscillation Algorithm

The approach to establishing the LBS setpoints is first to determine an acceptably low setpoint such that expected plant evolutions will not result in an LBS system trip, and then confirm that the setpoint provides margin to the MCPR Safety Limit. Therefore, evaluation of plant operating data against potential trip algorithms is a necessary step in the determination of the setpoints. Basic characteristics of oscillations are already known from test data and operating experience. Sample trip algorithms have been conceptually designed based on this information, and are described in Appendix B. Neutron flux signals generated by a simplified point kinetics model (including the effects of random noise), and digitally recorded plant data during various maneuvers are analyzed to determine the margin to trip for expected plant maneuvers without oscillations. Desired trip margins will be established based on trip avoidance and previous experience with other trip systems.

The analytical methodology used to demonstrate that the trip algorithm provides margin to the MCPR Safety Limit is described in Section 6.0. The methodology simulates LPRM and LBS responses to oscillations of various growth rates. These simulations explicitly account for delays associated with oscillation recognition by the algorithm and delay time associated with the ASF. Sections 6.2 and 6.3 detail how these delays and the algorithm setpoints themselves are related to the MCPR performance of the limiting bundles, and how uncertainties associated with oscillation recognition are accounted for.

Digitally recorded data from actual BWR instability events will also be evaluated using the trip algorithms. This evaluation will further confirm that the trip algorithm can readily identify the occurrence of oscillations.

A.5.4.2 Detection and Suppression Oscillation Methodology

Any detection and suppression system requires a method for relating the LPRM responses during expected modes of oscillations to the MCPR of the limiting fuel assemblies. This basic oscillation methodology for the LBS is described in Section 6.2. Application of this methodology (Section 6.3) considers the various uncertainties and initial conditions associated with defining the oscillation response of limiting bundles. Section 6.4 describes how this generic methodology will be applied to determine plant specific setpoints. Appropriate parameters will be reviewed to ensure that the setpoint remains adequate for future reload cycles.

A.5.4.3 Trip Bypass on Low Flow/High Power

As described in Section A.5.2.3.1, the LBS trip function is bypassed at high core flows and low power levels based on conservative application of oscillation operating experience. The regional exclusion boundary definition methodology (Section 5.1) will be used to confirm that the high flow and low power trip bypass setpoints are appropriate.

A.5.5 Technical Specification Implementation Philosophy

The LBS channels provide inputs to RPS or SRI functions, and therefore, the Technical Specifications associated with the LBS system will be similar to the existing RPS instrumentation specifications. The specifications will provide detail involving the minimum number of channels required per trip system. LPRM operability requirements for assuring the operability of each LBS channel will also be specified. The specification will also provide information regarding applicability and actions required if the requirements of the LBS specification are not met. Setpoints and surveillance requirements similar to other RPS instrumentation will be provided such that proper testing and operability can be performed/determined. Surveillance frequencies and allowable out of service times will be developed consistent with the reliability of the microprocessor design. Notations regarding bypassing of the system and other features will also be included. With this system

installation, the requirement for actions denoted in NRC Bulletin 88-07 and SIL 380 will be eliminated.

A.5.6 Operator Guidance

The LBS system will generate alarms, half scram or SRI actuations, or full actuations as required. Operator interface with the LBS system will be primarily through front panel annunciators and through existing Neutron Monitoring System user interfaces. Operator response to LBS alarms and channel trips (half or full) will be similar to operator response to other alarms and trips and will be supported by appropriate training. The following recommendations are provided by the BWROG regarding operator response to LBS alarms and trips.

> The operator will be required to investigate the cause of the alarm or channel trip. Upon determination that the LBS has contributed to the alarm or channel trip, the operator will proceed to locate the area of oscillation through the use of flashing downscale/upscale LPRM lights and alarms, review of LPRM hardwire displays via rod selection, or other optional displays. Having determined the cause and area of concern relating to the LBS alarm or channel trip, the operator will take proper actions to mitigate thermal-hydraulic instabilities if evidence of such instabilities exists. Operator action to suppress thermal-hydraulic instabilities may include insertion of CRAM rods and/or an increase in core flow as deemed appropriate.

In addition, for a trip of both reactor recirculation pumps, the operator will assure the reactor is stable before attempting to restart the recirculation pumps. All plants are expected to provide operator training/guidance in oscillation prevention and mitigation for scram or SRI avoidance reasons.

Table A-1

BWR PLANT PARAMETERS

	DA	VY	MONT	FITZ	B1	B2	LS2
Fuel Assemblies	368	368	484	560	560	560	764
Power Density (kW/l)	51	51	40	51	51	51	50
APRM Trip Setpoint at Natural Circulation	62	54	62	62	62	54	62
Inlet Orifice Diameter (in)	2.090	2.222	2.148	2.090	2.430	2.090	2.430
Cycle	11	15	14	10	8	9	4
Fuel Type	GE 8x8	GE 8x8	GE 8x8	GE 8x8	GE 8x8	GE 8x8	GE 8x8
Expected Eigenvalue Separation (-$)							
Axial Mode	1.7-2.1	1.7-2.1	1.3-1.9	1.5-1.8	1.5-1.8	1.5-1.8	1.3-1.5
Azimuthal Mode	2.0-2.7	2.0-2.7	2.1-2.5	1.3-1.7	1.3-1.7	1.3-1.7	1.0-1.3

DA = Duane Arnold
VY = Vermont Yankee
MONT = Monticello
FITZ = FitzPatrick
B1 = Brunswick 1
B2 = Brunswick 2
LS2 = LaSalle-2

Table A-2

DUANE ARNOLD CYCLE 10 REGION BOUNDARY CALCULATIONS

Point *	Power/Flow (%/%)	Hot Channel Decay Ratio	Core Decay Ratio
High Rod Line			
1	66.8/40.0	0.28	0.77
2	65.0/38.0	0.31	0.86
3	64.1/36.9	0.33	0.92
Medium Rod Line			
4	54.7/40.0	0.21	0.63
5	51.0/35.0	0.25	0.78
6	48.7/32.0	0.30	0.91
Natural Circulation			
7	37.0/30.0	0.23	0.62
8	42.0/30.0	0.27	0.78
9	45.0/30.0	0.31	0.90
10	52.0/30.0	0.39	1.12

* See Figure A-7 for definition of points.

Table A-3

FUNDAMENTAL AND AZIMUTHAL HARMONIC POWER DISTRIBUTIONS (CONTOURS) FOR DAEC ANALYSIS

Cycle	Cycle Exposure (MWd/ST)	Harmonic Orientation
10	200	NWSE
10	200	NESW
10	3000	EW
10	3000	NS
10	5400	NWSE
10	5400	NESW
10	End-of-Cycle	NWSE
10	End-of-Cycle	NESW

Table A-4

DAEC MCPR RESULTS FOR CORE-WIDE OSCILLATIONS

	MCPR 95/95
110% Rod Line	1.417
100% Rod Line	1.391

Table A-5

DAEC MCPR RESULTS FOR REGIONAL OSCILLATIONS

	MCPR 50/50
110% Rod Line	1.266

Table A-6

LPRM CONFIGURATIONS FOR U.S. BWRs
(BWR/3-6)

Plant Name		Number of LPRM Strings	Plant Category
Vermont Yankee Duane Arnold		20	Small Core
Monticello		24	Small Core
Millstone Pilgrim		30	Large Core
FitzPatrick Brunswick 1,2	Hatch 1,2 Cooper	31	Large Core
River Bend		33	Large Core
Clinton		33	Solid State RPS
Dresden 2,3 Quad Cities 1,2	Perry	41	Large Core
Browns Ferry 1,2,3 Peach Bottom 2,3 Hope Creek Susquehanna 1,2 Hanford 2	Fermi-2 Limerick 1,2 LaSalle 1,2 Nine Mile Point 2	43	Large Core
Grand Gulf		44	Large Core

Table A-7

LARGE CORE LPRM ASSIGNMENTS
764 BUNDLE PLANT

BWR/3-5

OPRM	RPS Channel	APRM*	LPRM Group
A	A1	A,E	
B	B1	B,F	
C	A2	C	A
D	B2	D	B

BWR/6 (except Clinton)

OPRM	RPS Channel	APRM	LPRM Group
A	A1	A,E	(not applicable to BWR/6)
B	B1	B,F	
C	A2	C,G	
D	B2	D,H	

* Conventions for APRM/RPS assignments vary from plant to plant.

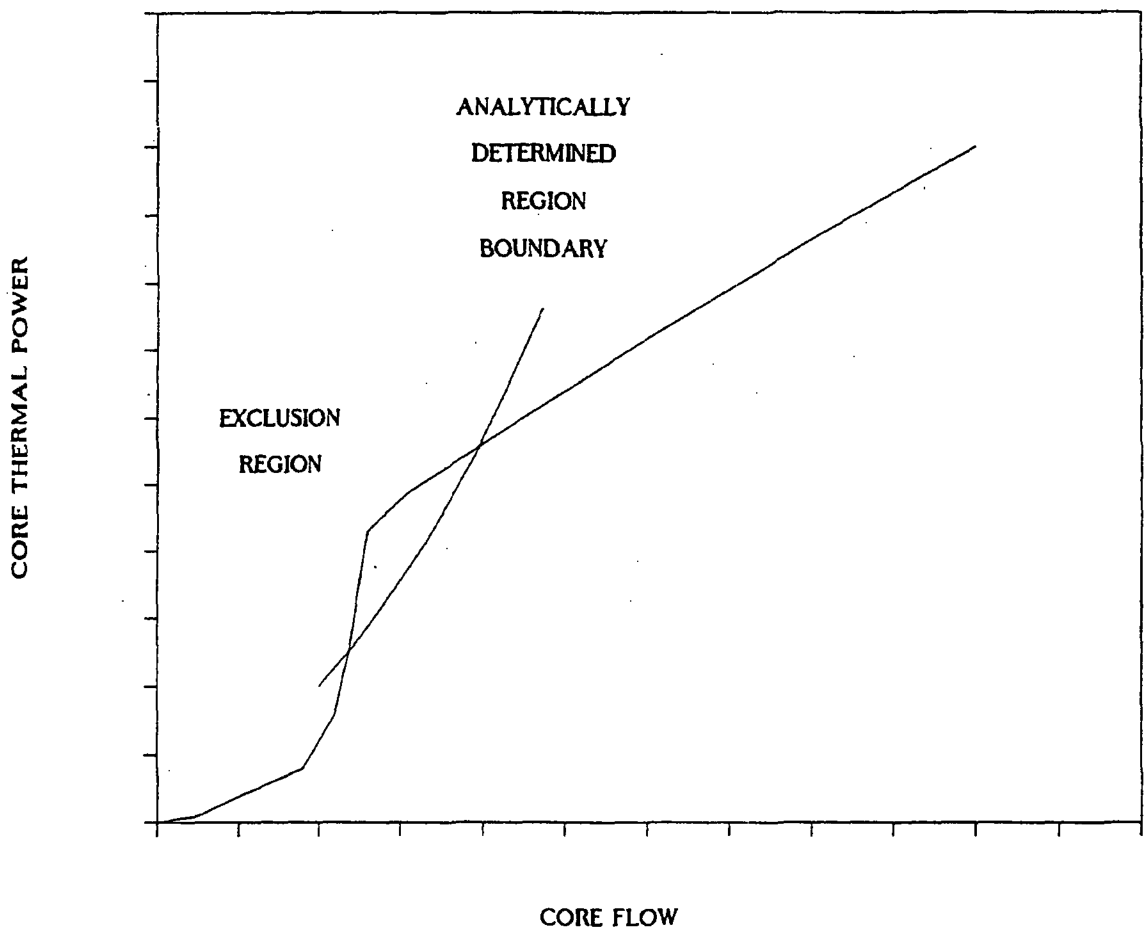


FIGURE A-1. EXAMPLE EXCLUSION REGION BOUNDARY

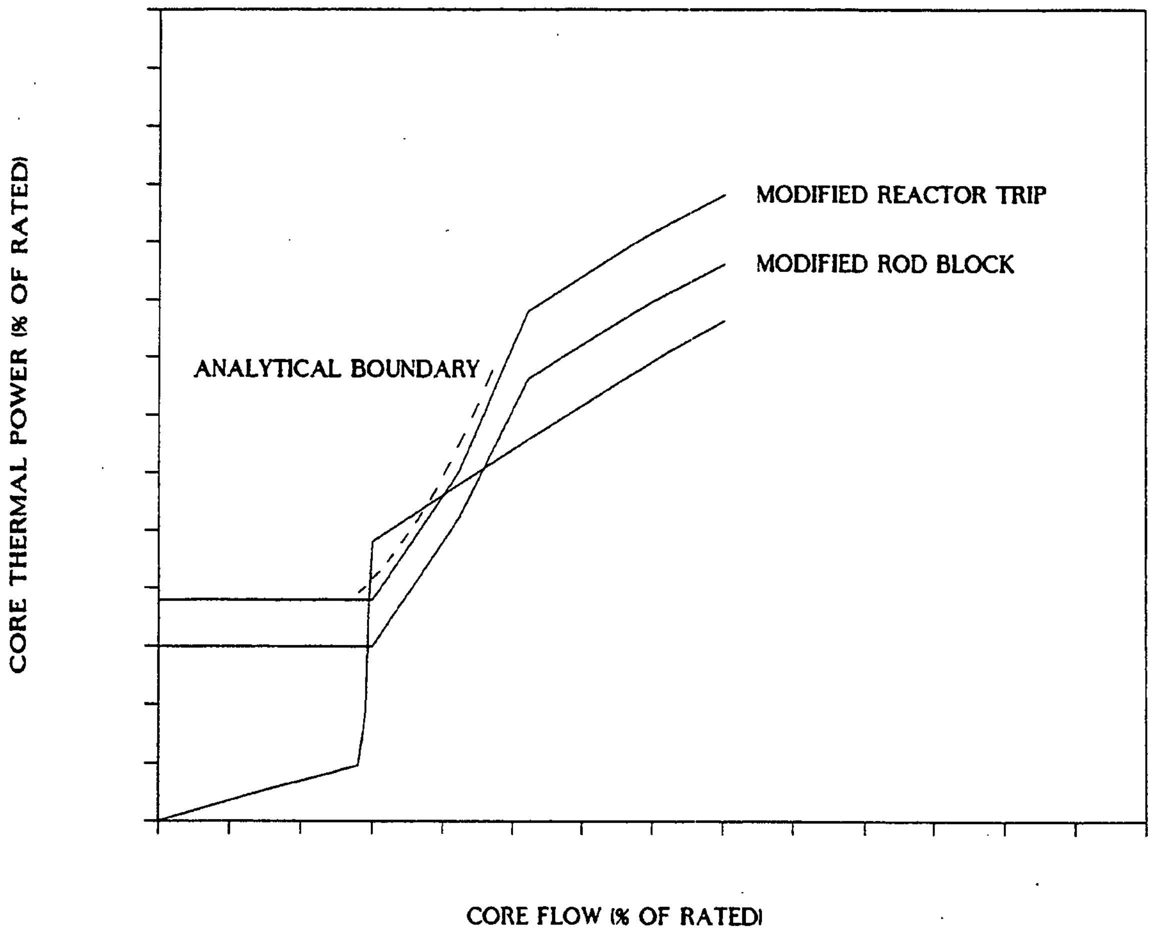


FIGURE A-2. EXAMPLE EXCLUSION REGION WITH MODIFIED FLOW BIASED REACTOR TRIP AND ROD BLOCK FUNCTIONS

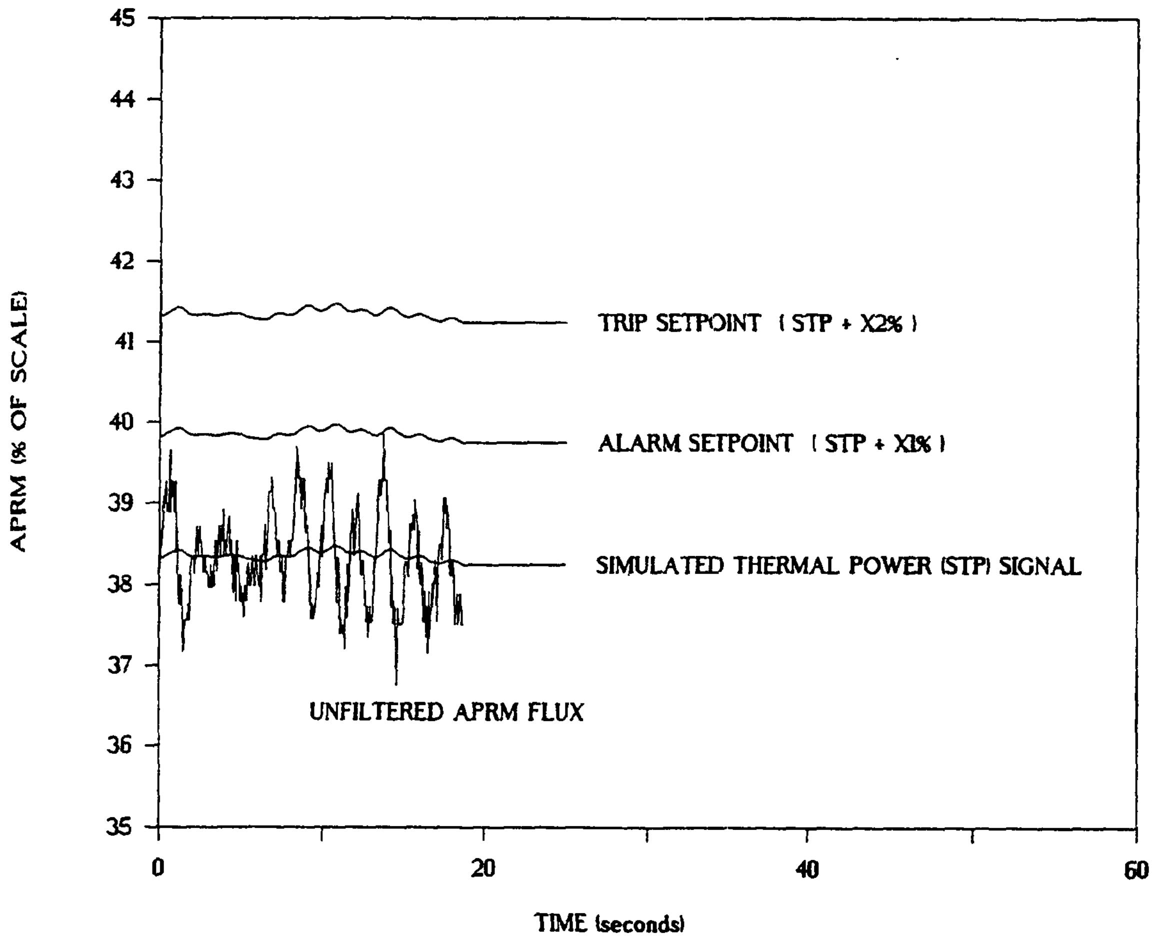


FIGURE A-3. OPTION I-C APRM STABILITY FLUX TRIP

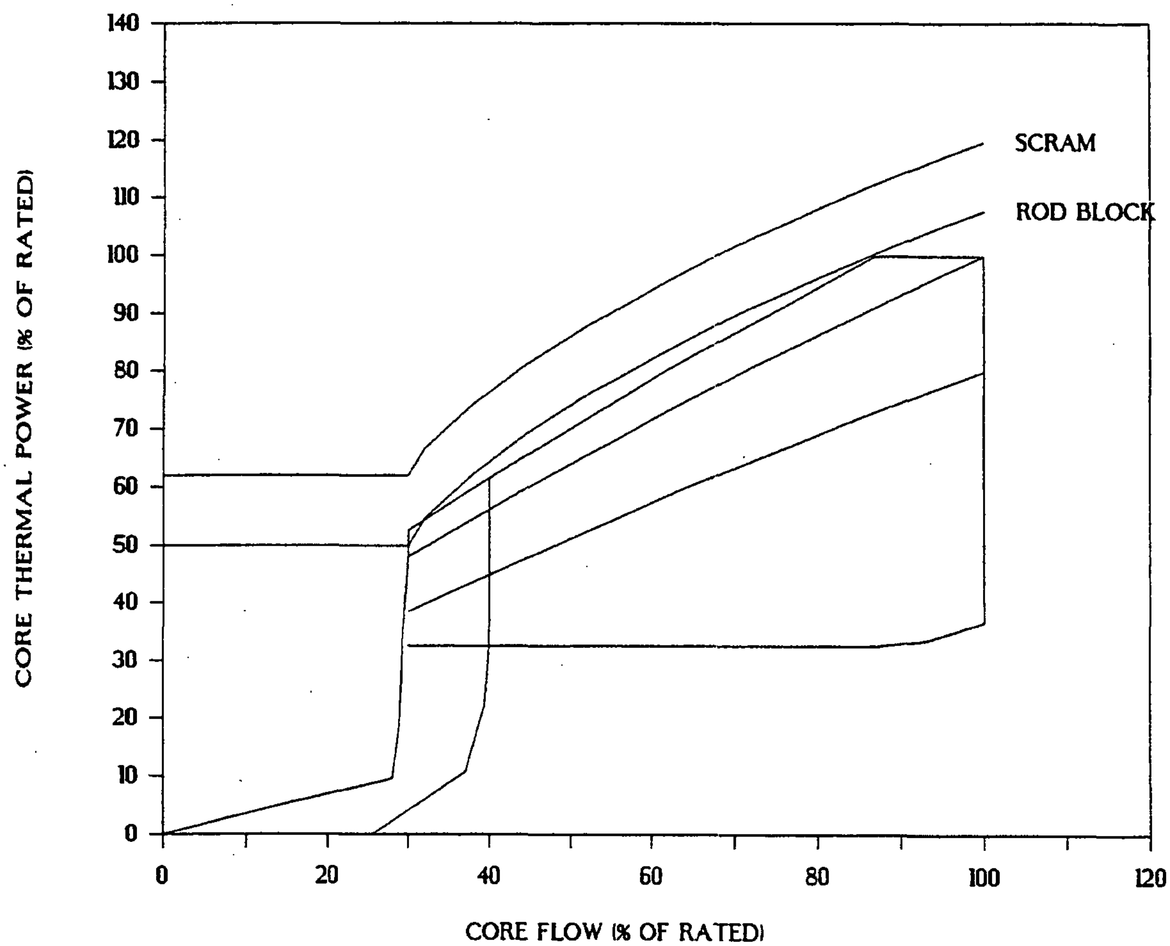


FIGURE A-4. FLOW-BIASED APRM NEUTRON FLUX SCRAM SYSTEM

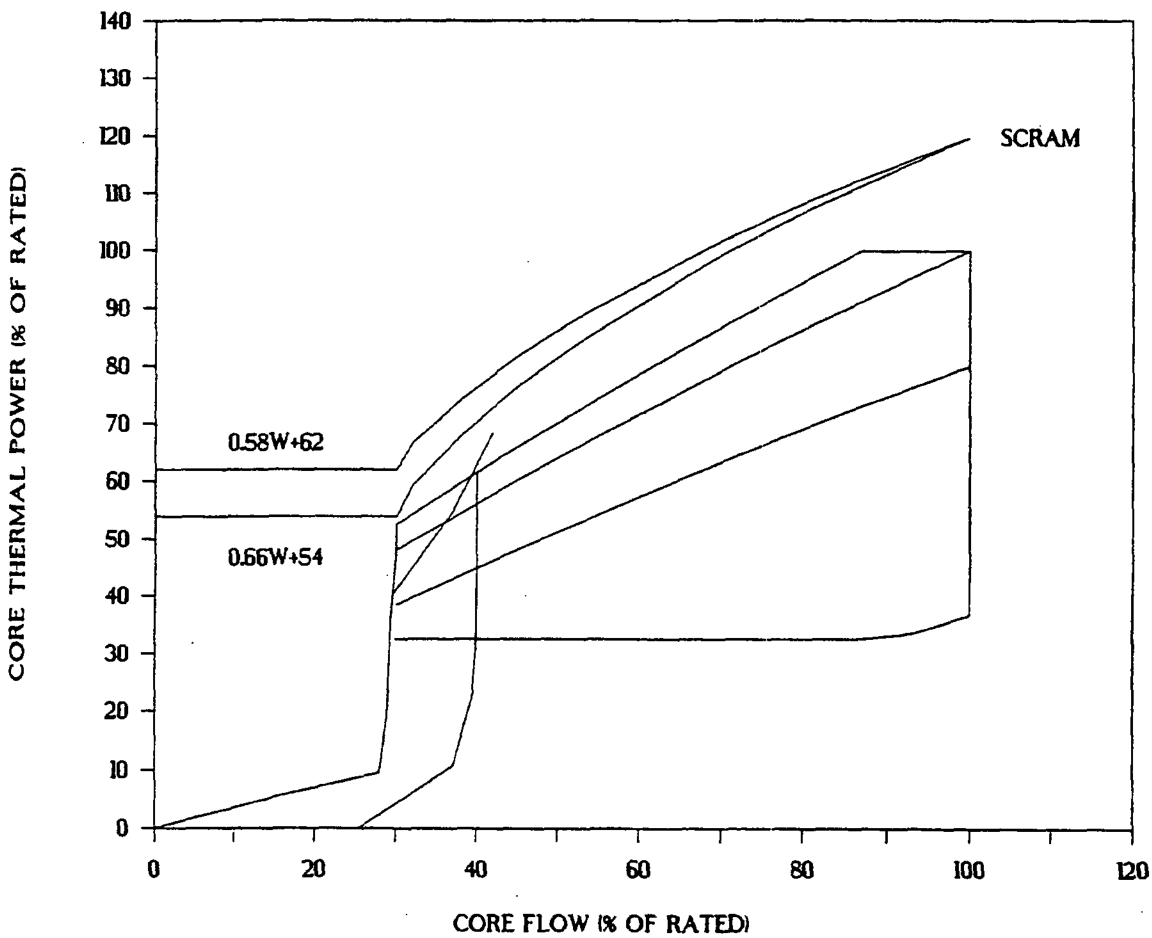


FIGURE A-5. PROTECTION FOR OPERATION INSIDE THE EXCLUSION REGION

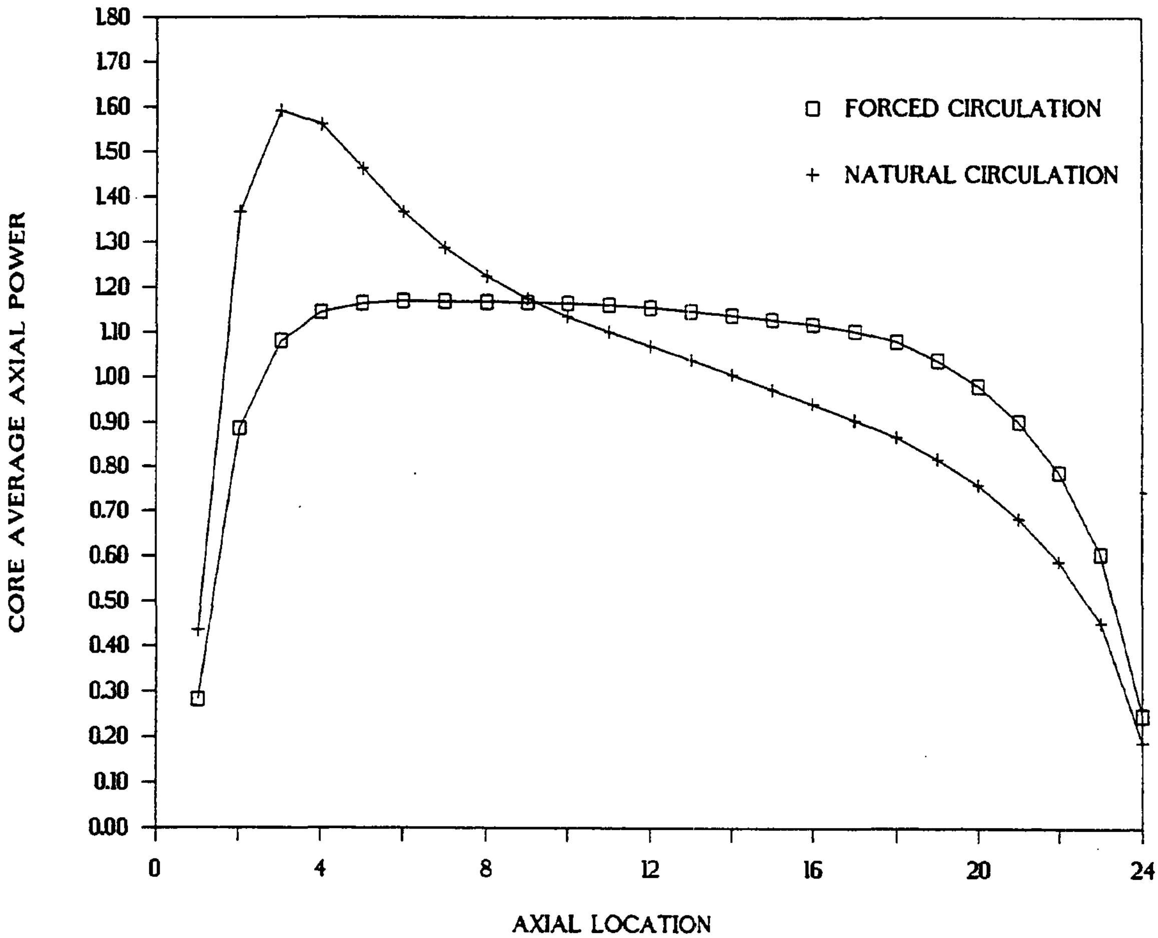


FIGURE A-6. DUANE ARNOLD CYCLE 10 EOC HALING AXIAL POWER DISTRIBUTIONS

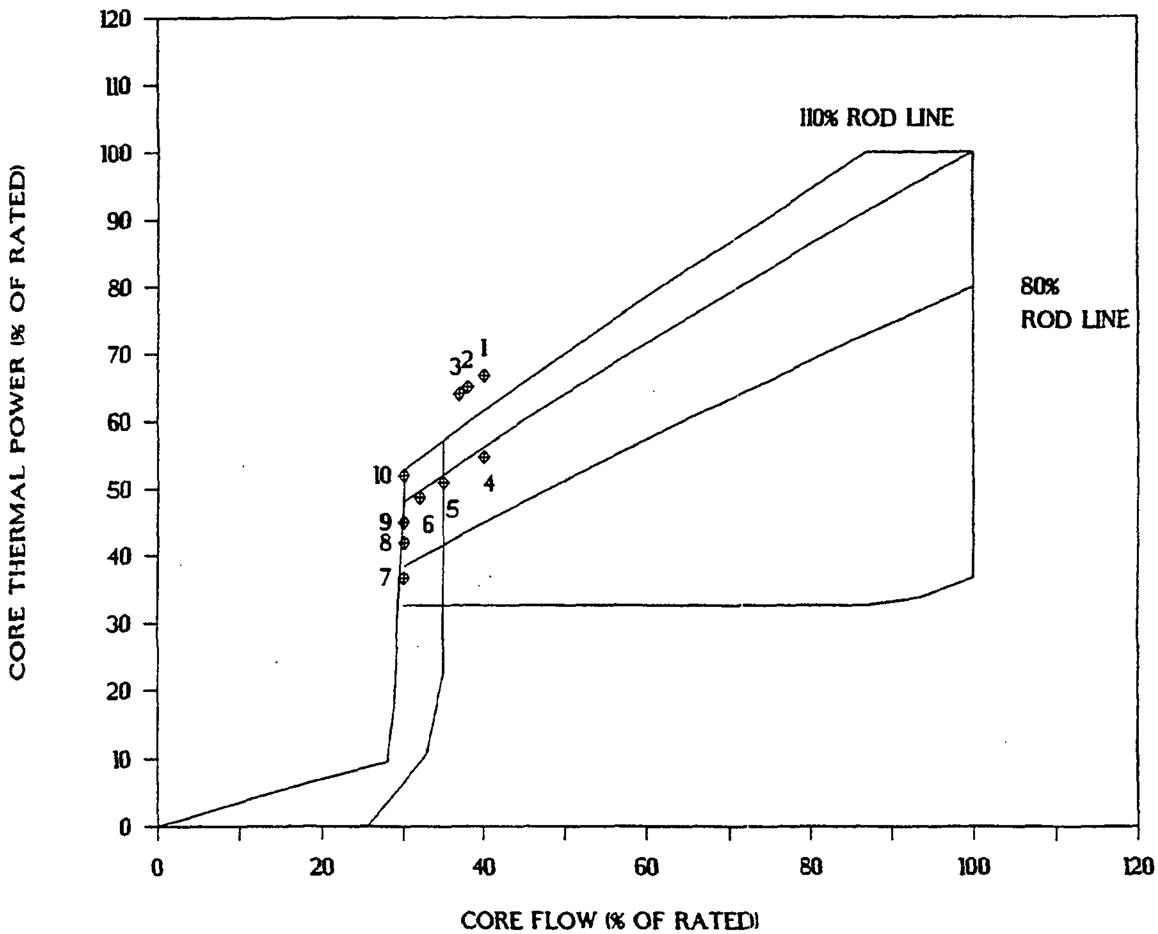


FIGURE A-7. DUANE ARNOLD CYCLE 10 REGION BOUNDARY DEFINITION ANALYSIS POINTS

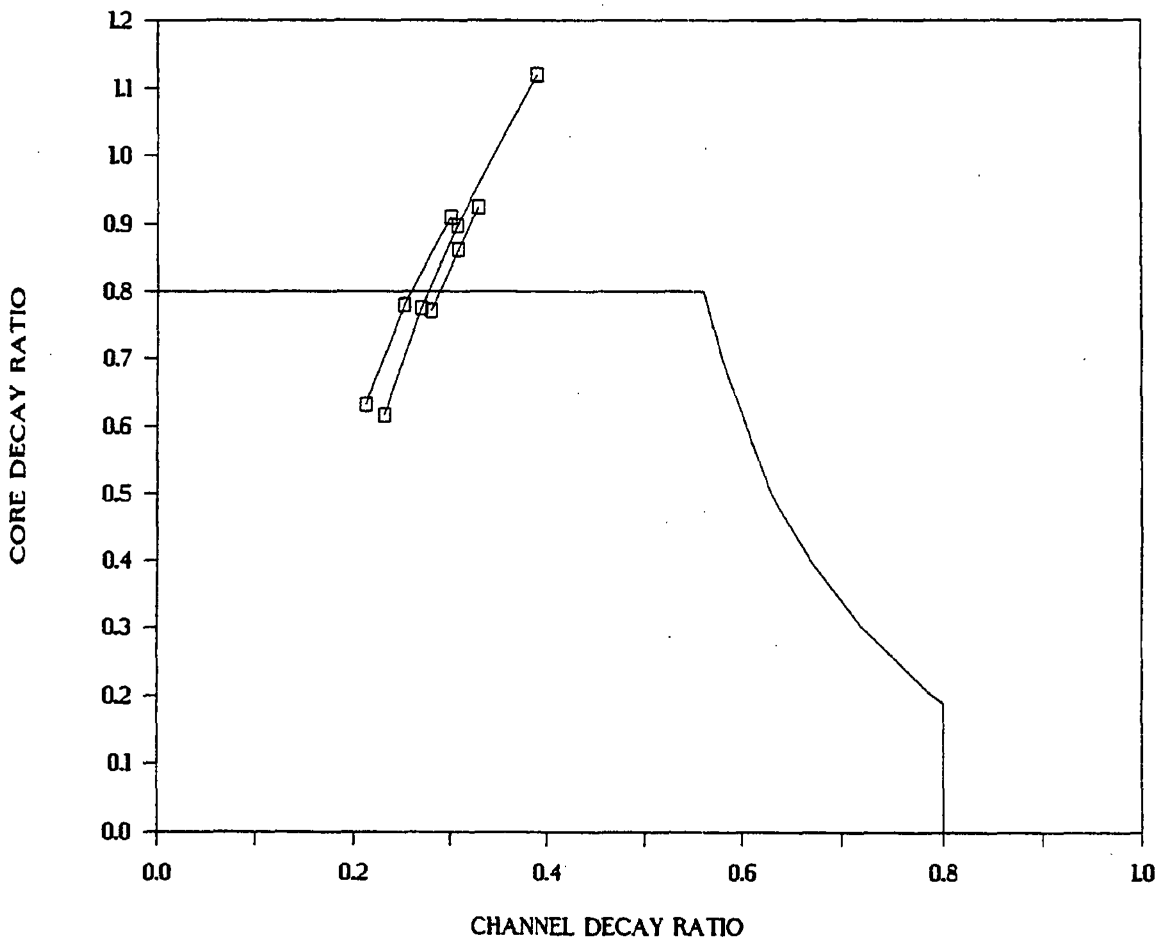


FIGURE A-8. DUANE ARNOLD CYCLE 10 DECAY RATIOS

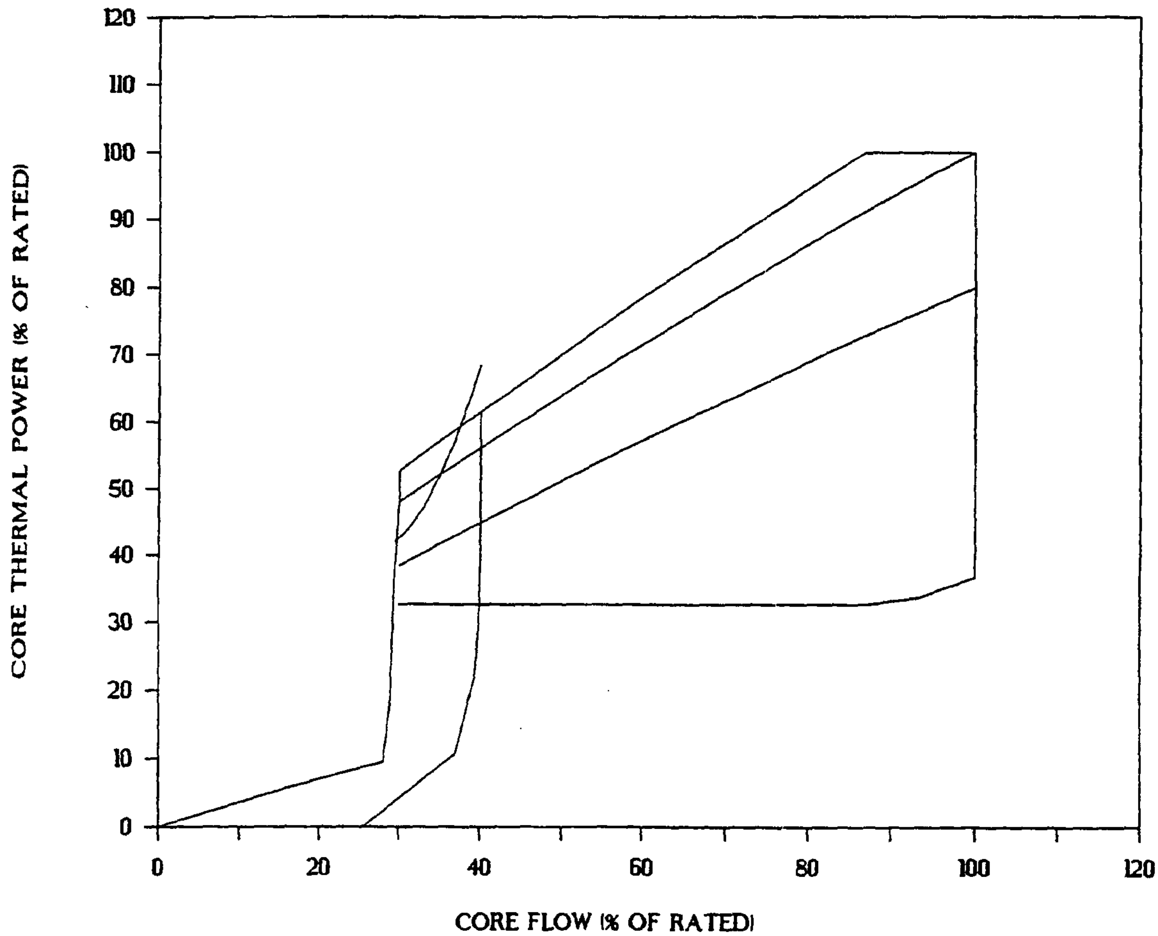


FIGURE A-9. DUANE ARNOLD CYCLE 10 REGION BOUNDARY DEFINITION

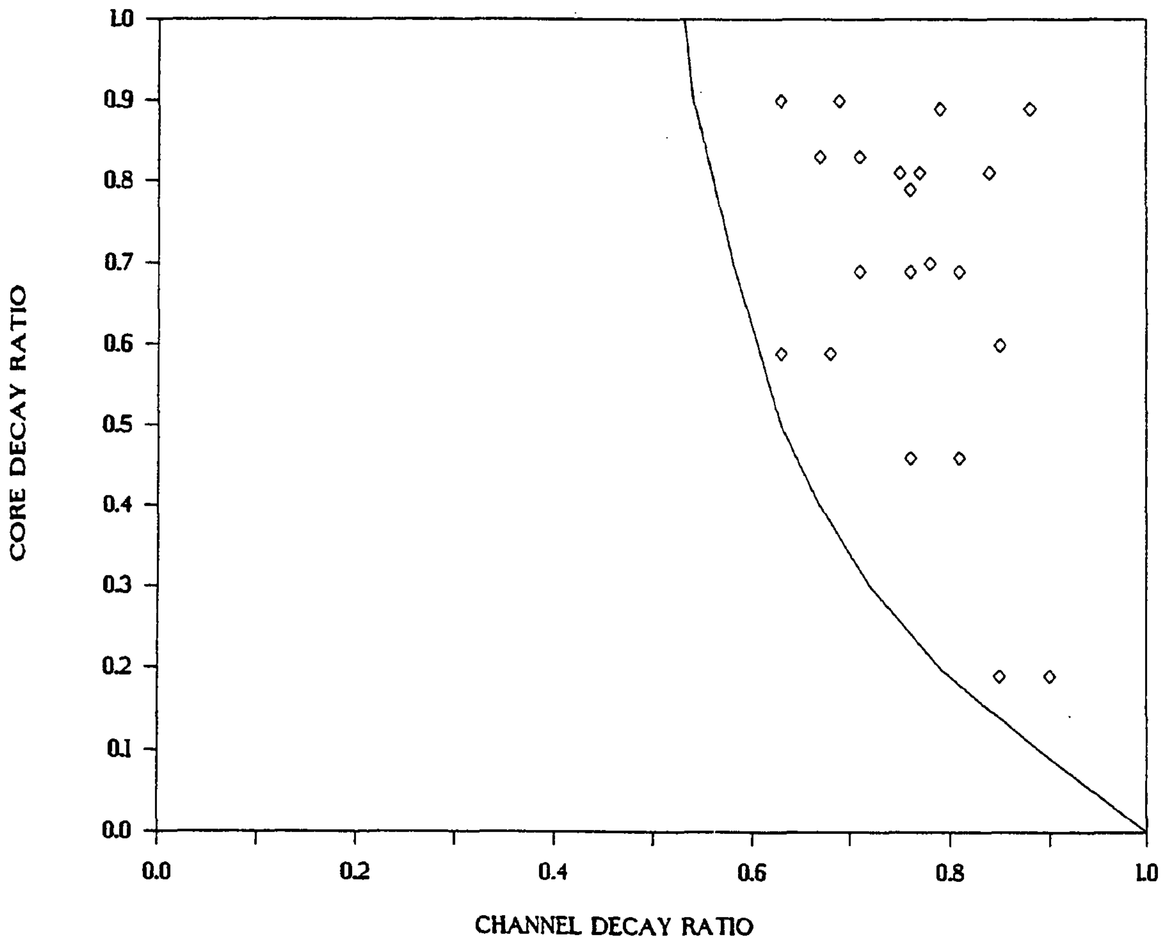


FIGURE A-10. CORE/CHANNEL DECAY RATIOS RESULTING IN CALCULATED REGIONAL OSCILLATIONS

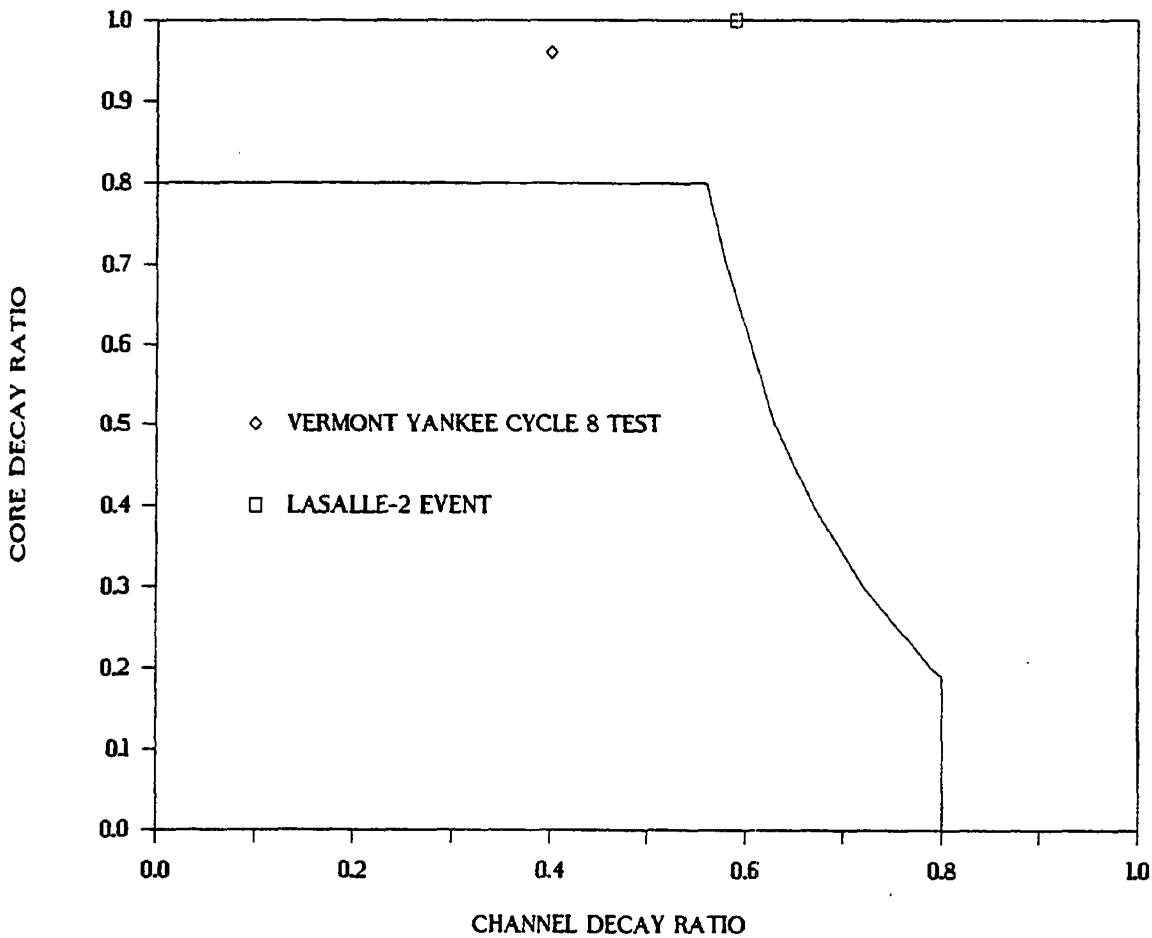


FIGURE A-11. STABILITY CRITERIA

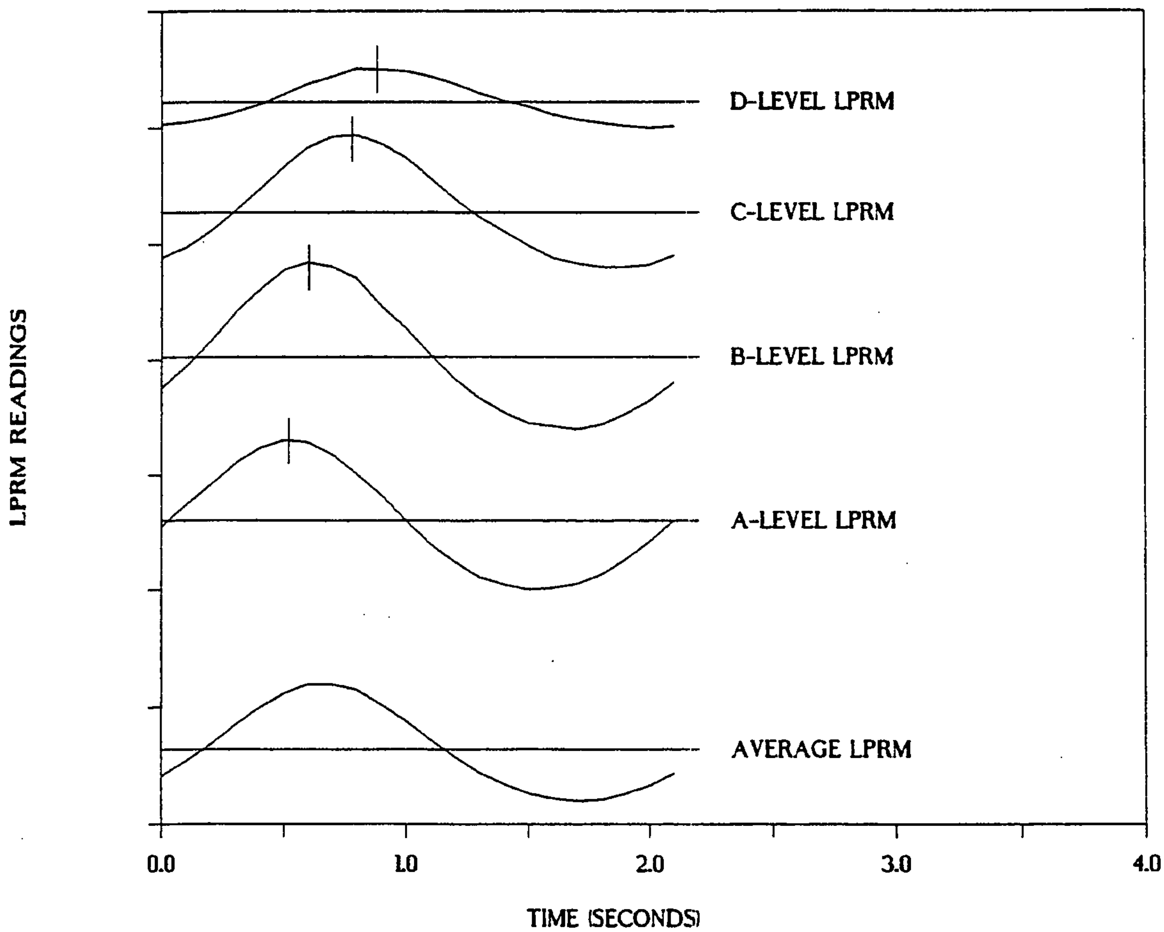


FIGURE A-12. AXIAL VARIATION IN LPRM READINGS DURING AN INSTABILITY

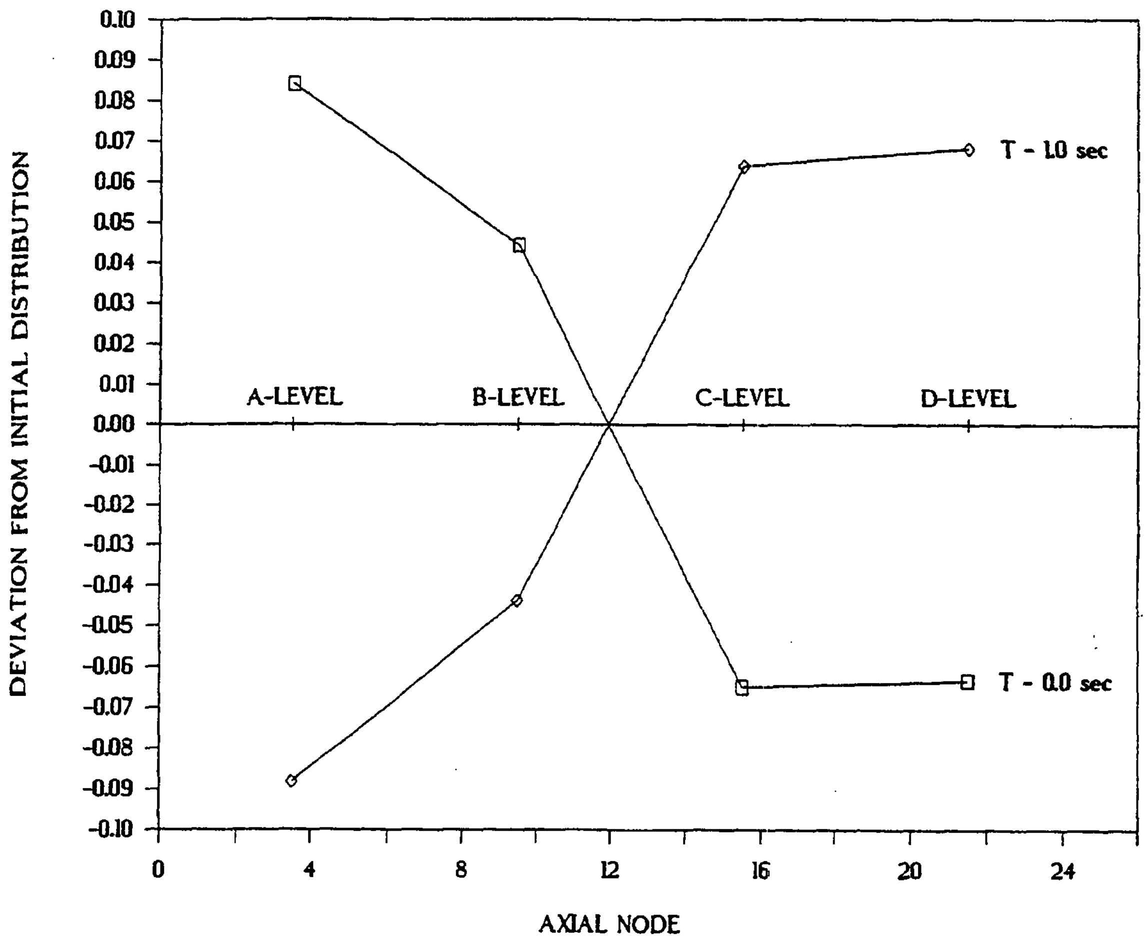


FIGURE A-13. CHANGE IN AXIAL POWER SHAPE DURING AN INSTABILITY

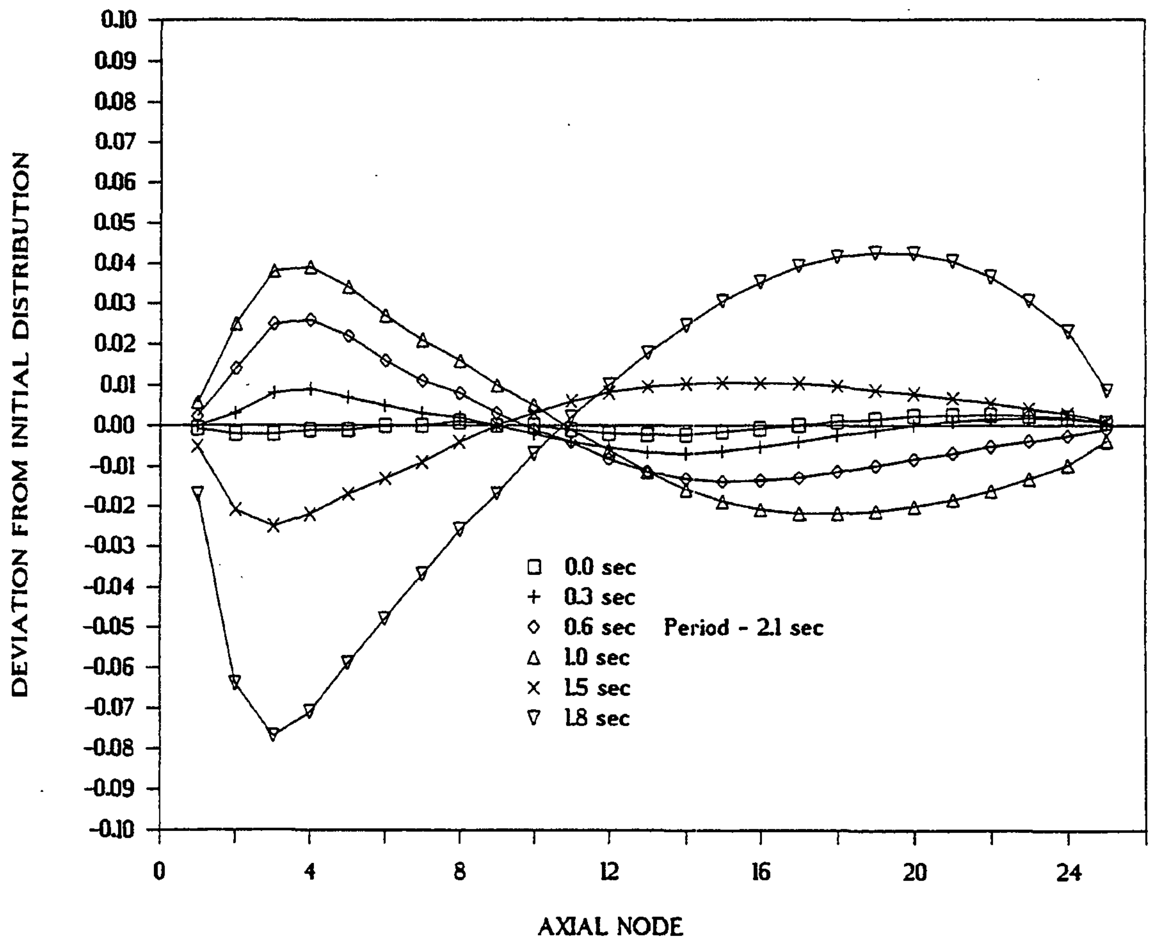


FIGURE A-14. TRAC-G PREDICTION OF THE AXIAL HARMONIC DURING AN INSTABILITY

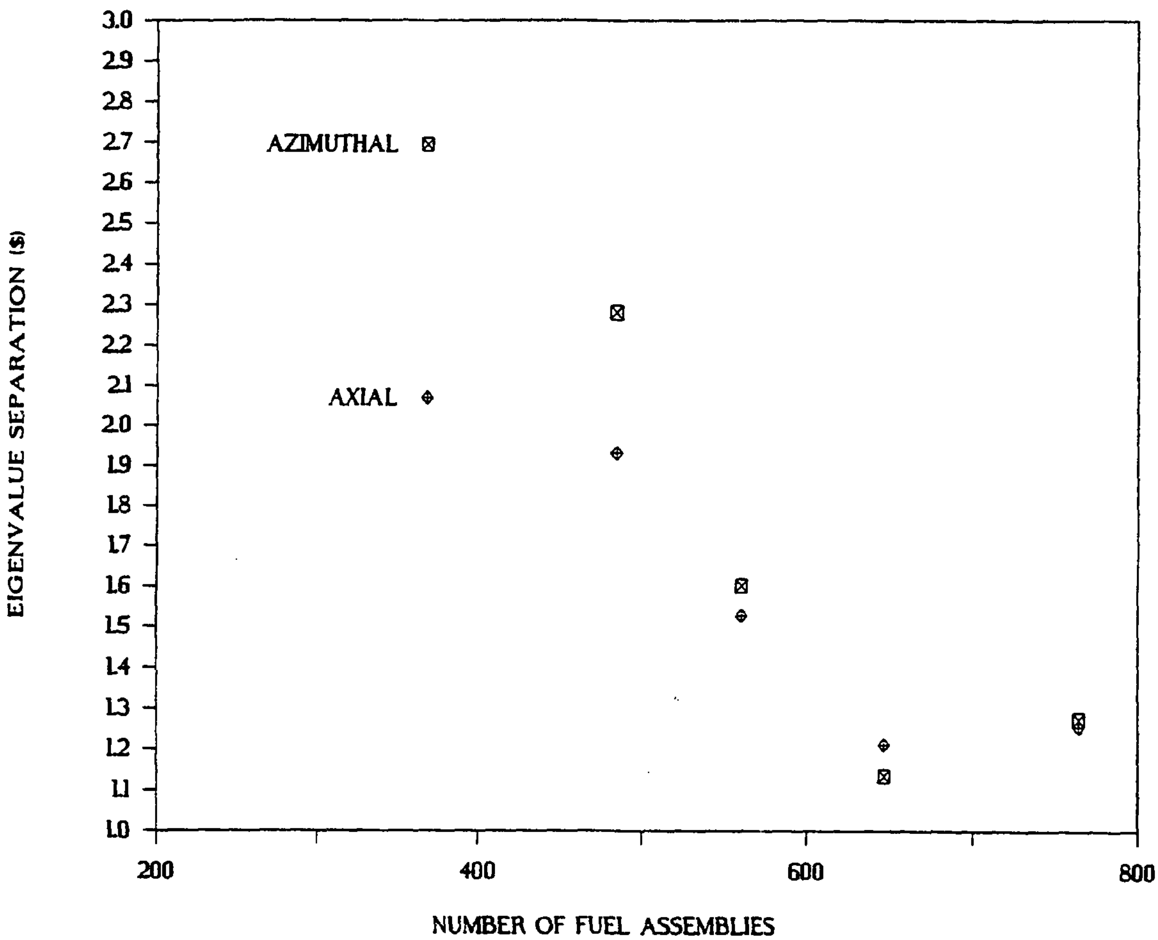


FIGURE A-15. EIGENVALUE SEPARATION AS A FUNCTION OF CORE SIZE

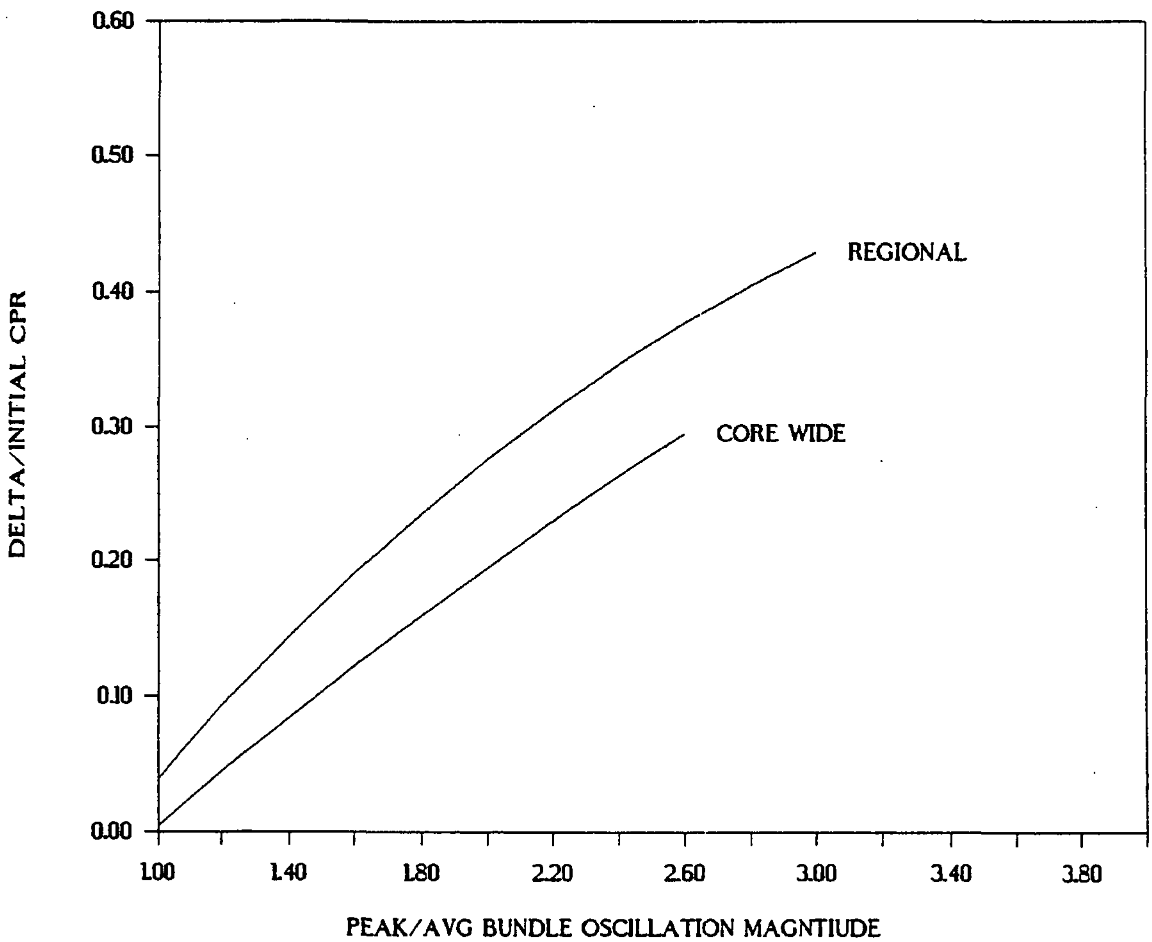


FIGURE A-16. MCPR PERFORMANCE DURING DIFFERENT OSCILLATION MODES

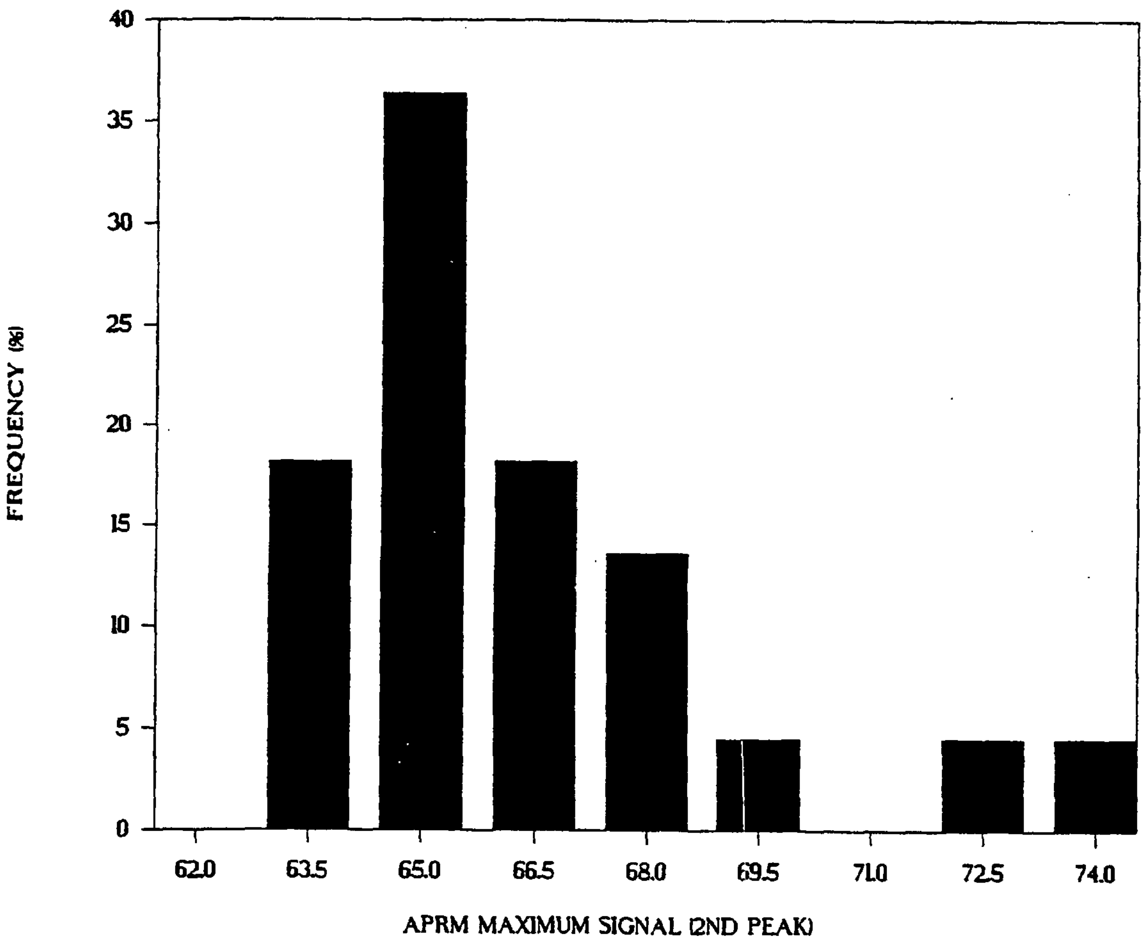


FIGURE A-17. EXAMPLE SETPOINT OVERSHOOT DISTRIBUTION

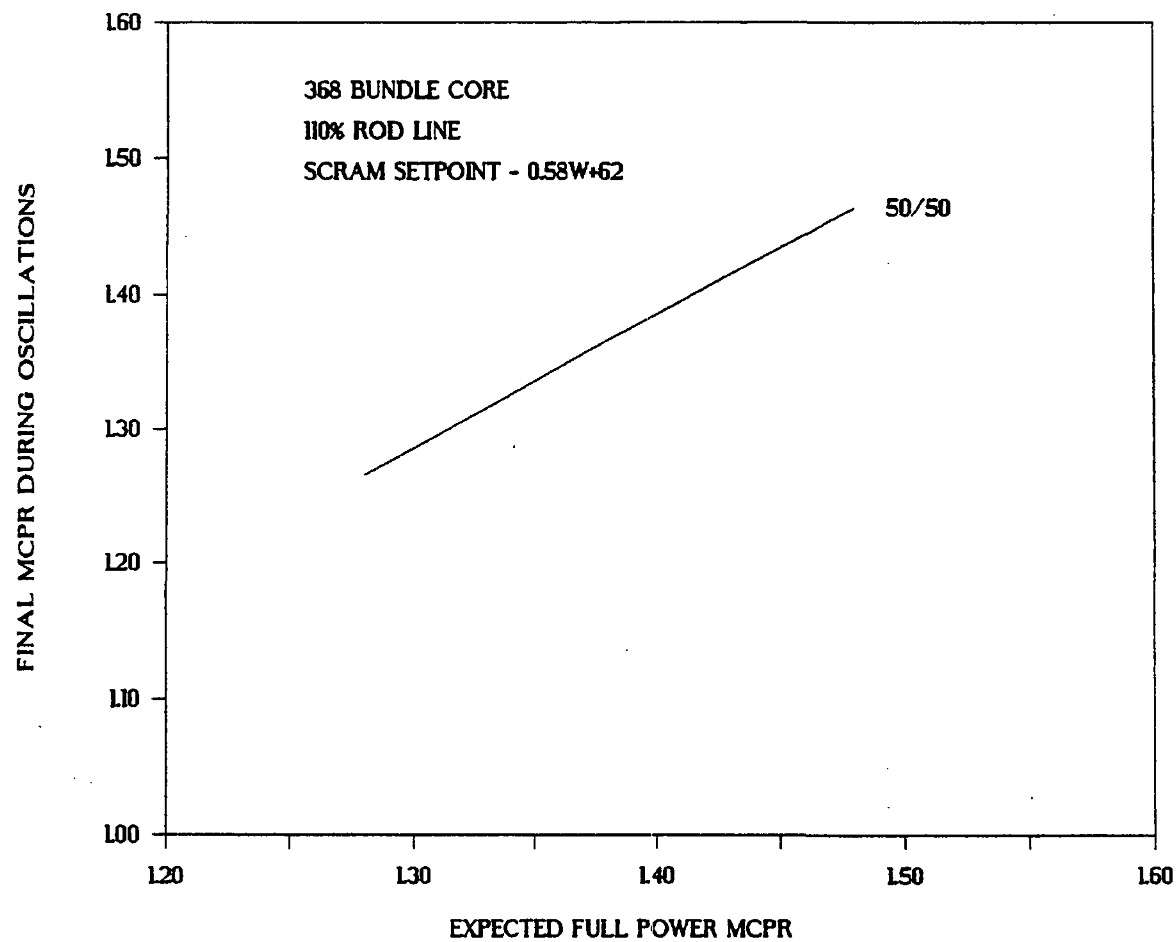
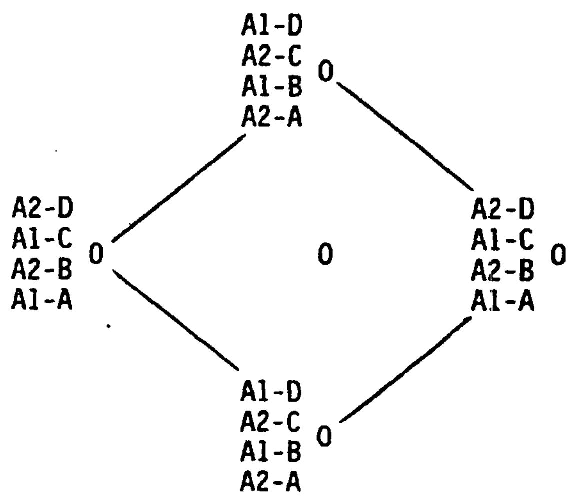
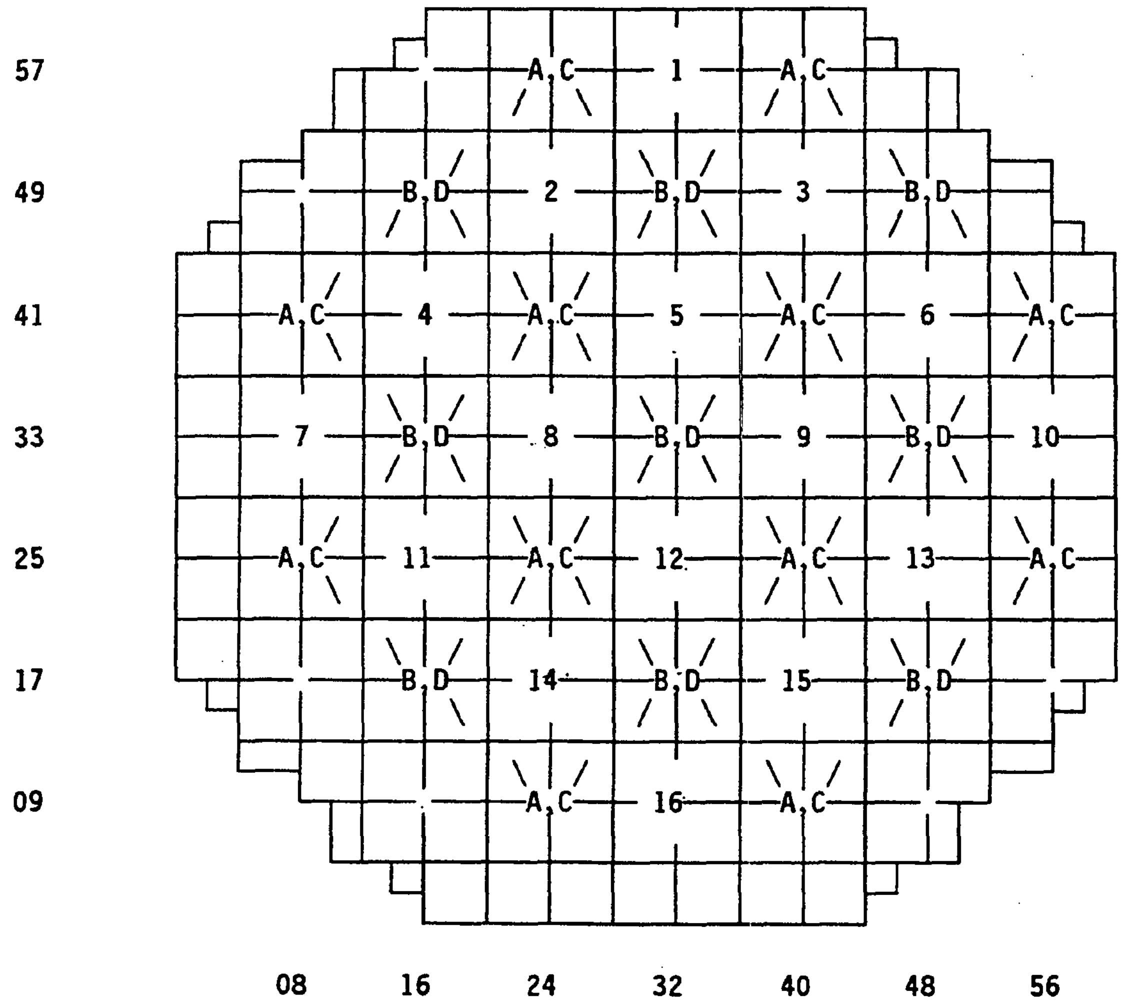


FIGURE A-18. SENSITIVITY TO INITIAL MCPR DURING REGIONAL OSCILLATIONS



O = LPRM string
A, B, C, D = LPRM detector levels (A is bottom detector)
A1, A2, B1, B2 = OPRM channel assigned to respective LPRM detector

FIGURE A-19. BASIC "DIAMOND" ASSIGNMENT SCHEME

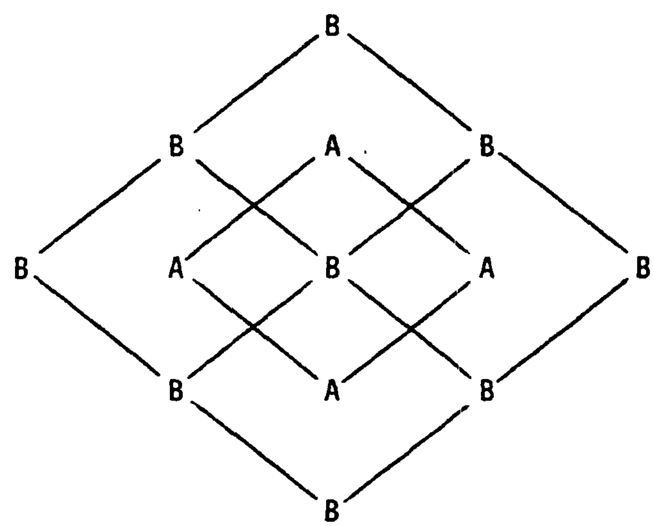



```
     x,y
   /     \
x,y   n   x,y
   \     /
     x,y
```

n = OPRM cell number

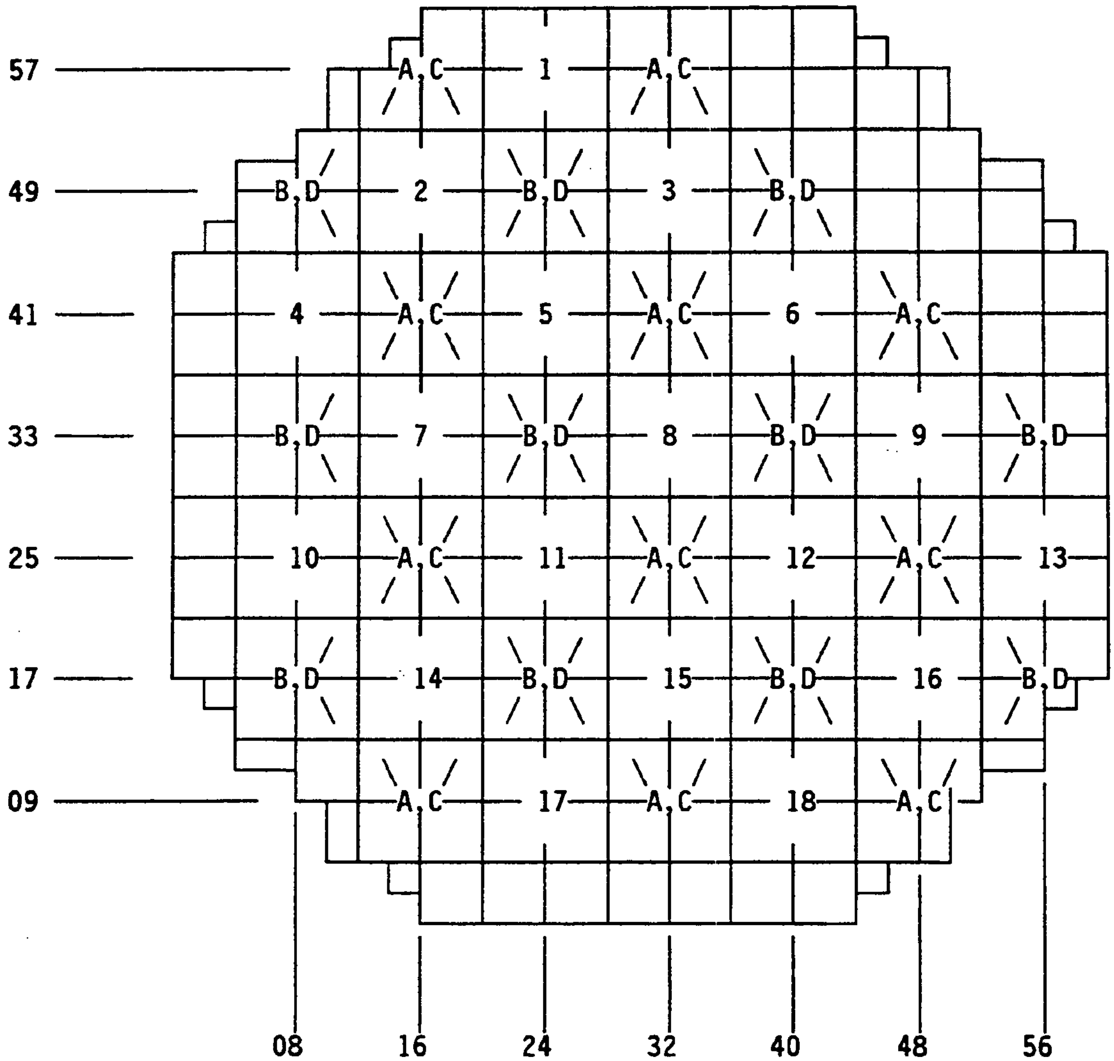
x,y = LPRMs assigned to OPRM-A1 channel (other LPRMs in string assigned to OPRM-A2 channel)

FIGURE A-20. 764 BUNDLE LPRM ASSIGNMENTS TO OPRM A1(A2)



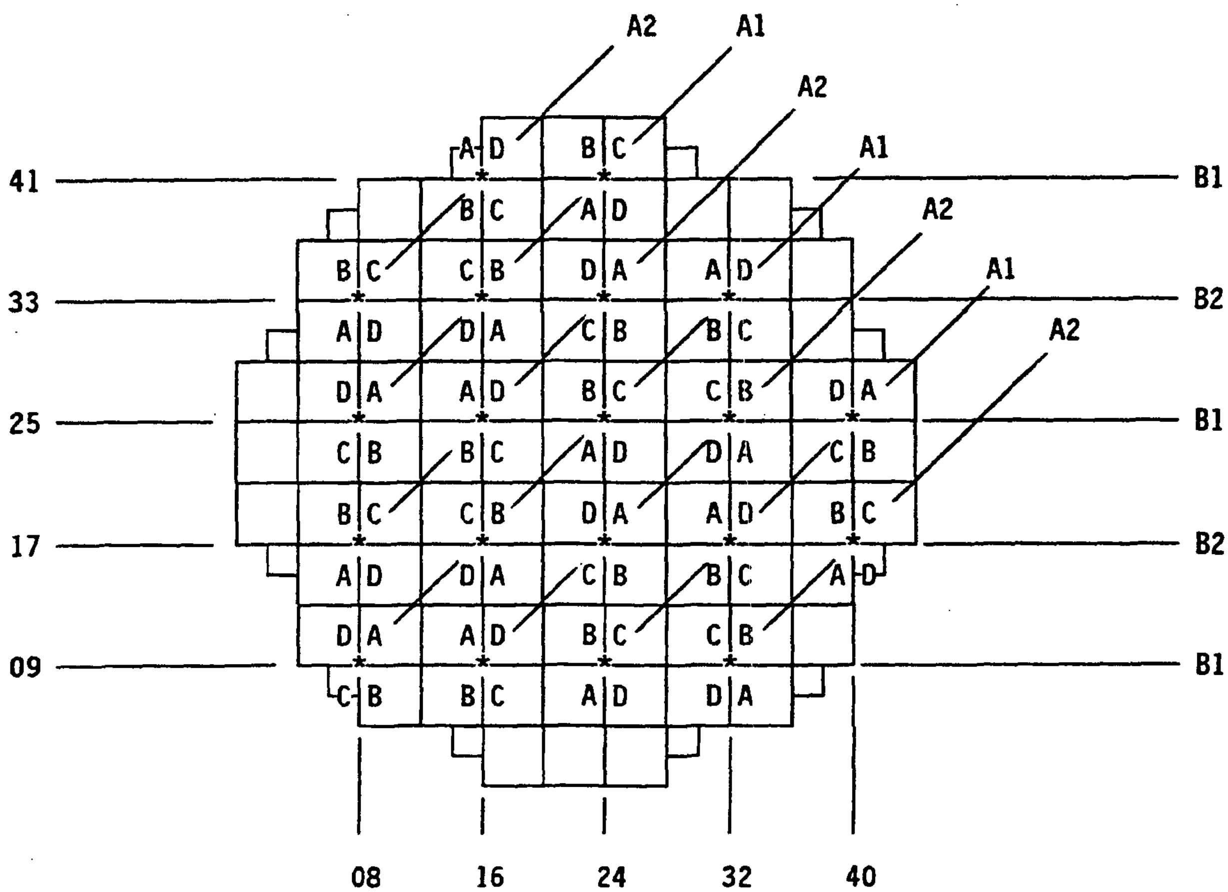
Note: A and B indicate LPRM string locations.

FIGURE A-21. EXAMPLE OF A AND B DIVISION OPRM OVERLAP



x,y x,y n x,y x,y	n	=	OPRM cell number
	x,y	=	LPRMs assigned to OPRM-B1 channel (other LPRMs in string are assigned to OPRM-B2 channel)

FIGURE A-22. 764 BUNDLE LPRM ASSIGNMENTS TO OPRM B1(B2)

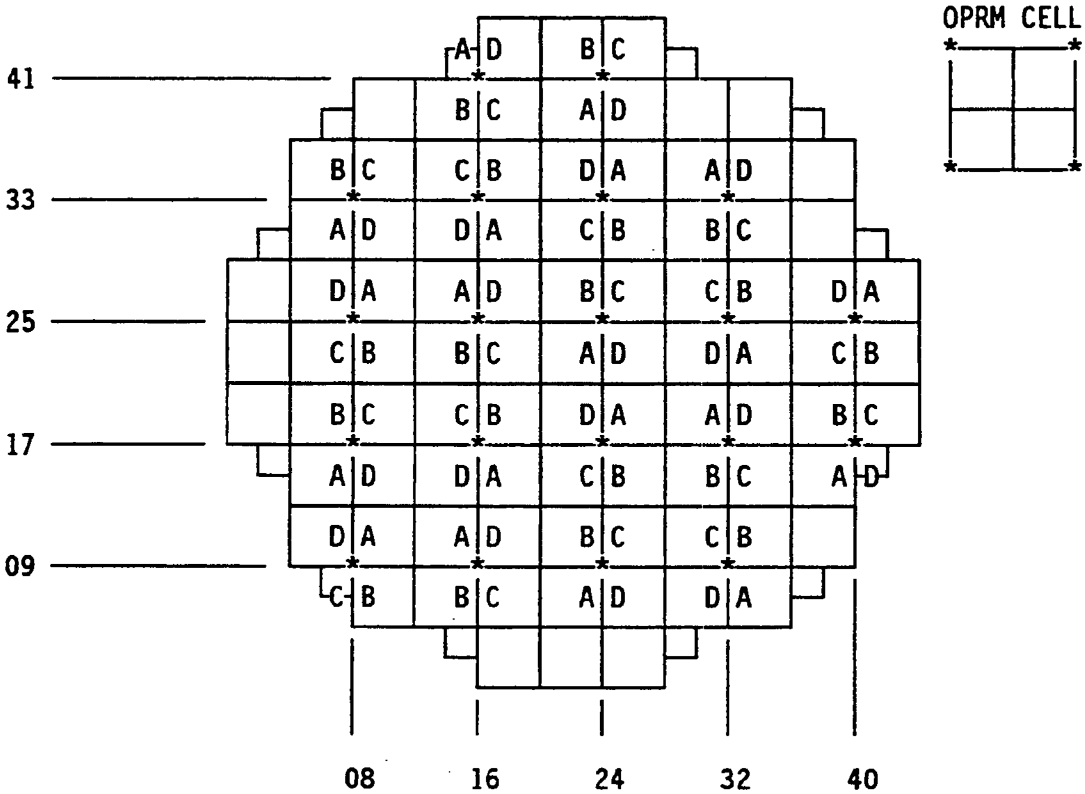


A1
A D
*—— B1
B C

Upper left/lower right letter = Input for OPRM Channels identified (A1/B1 in example)

Upper right/lower left letter = Input for OPRM Channels not identified (A2/B2 in example)

FIGURE A-23. SMALL CORE LPRM ASSIGNMENT SCHEME - TRIANGLE DESIGN

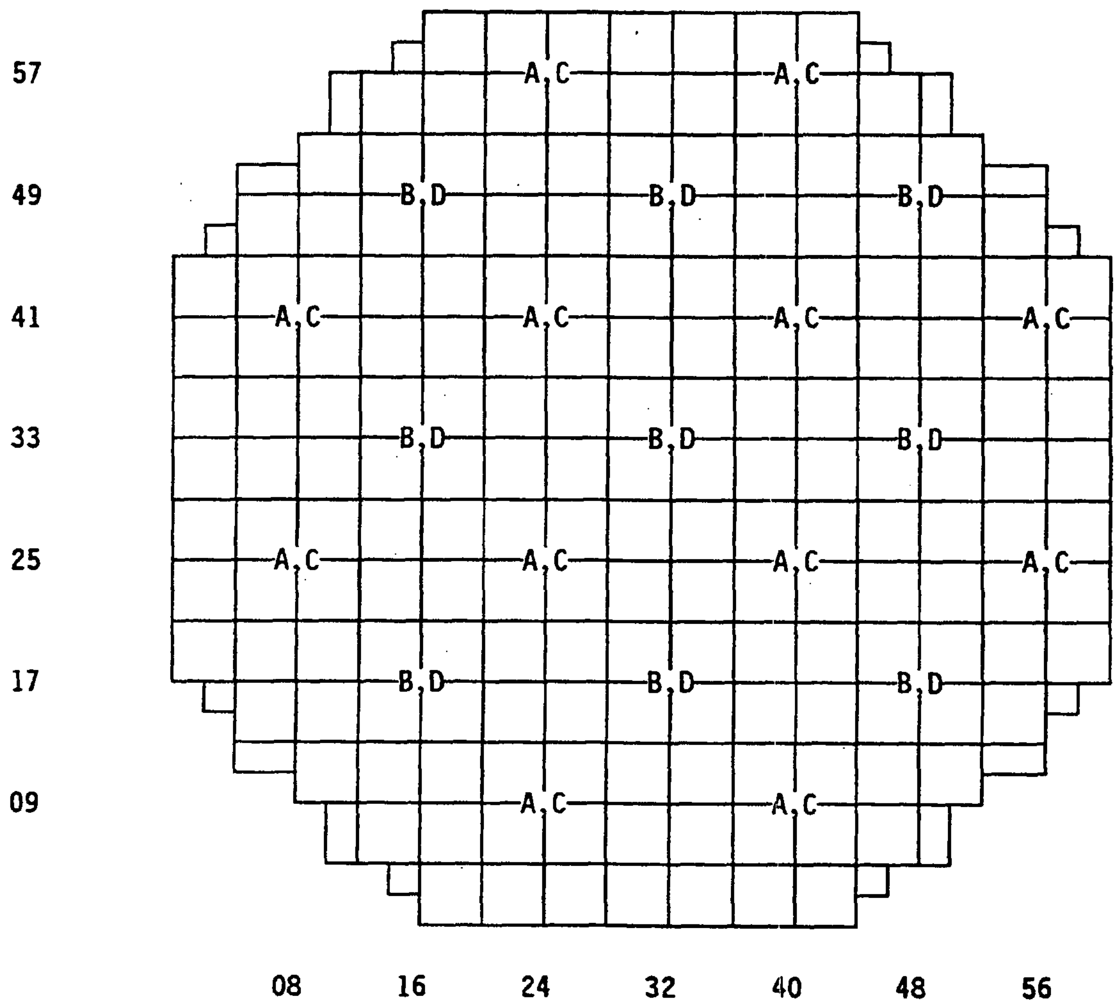


A|B
—*—
D|C

LPRMs providing input to OPRM Channels A1, A2, B1, and B2

Upper left letter = Input for OPRM Channel A1
Lower left letter = Input for OPRM Channel B1
Lower right letter = Input for OPRM Channel A2
Upper right letter = Input for OPRM Channel B2

FIGURE A-24. SMALL CORE LPRM ASSIGNMENT SCHEME - SQUARE DESIGN



x,y = LPRMs assigned to OPRM-A1 channel (other LPRMs in string assigned to OPRM-A2 channel)

Total number of A1 and A2 cells = 21

FIGURE A-25. 764 BUNDLE LPRM ASSIGNMENTS TO OPRM-A1 FOR THE 2 LPRMs = 1 CELL EXAMPLE

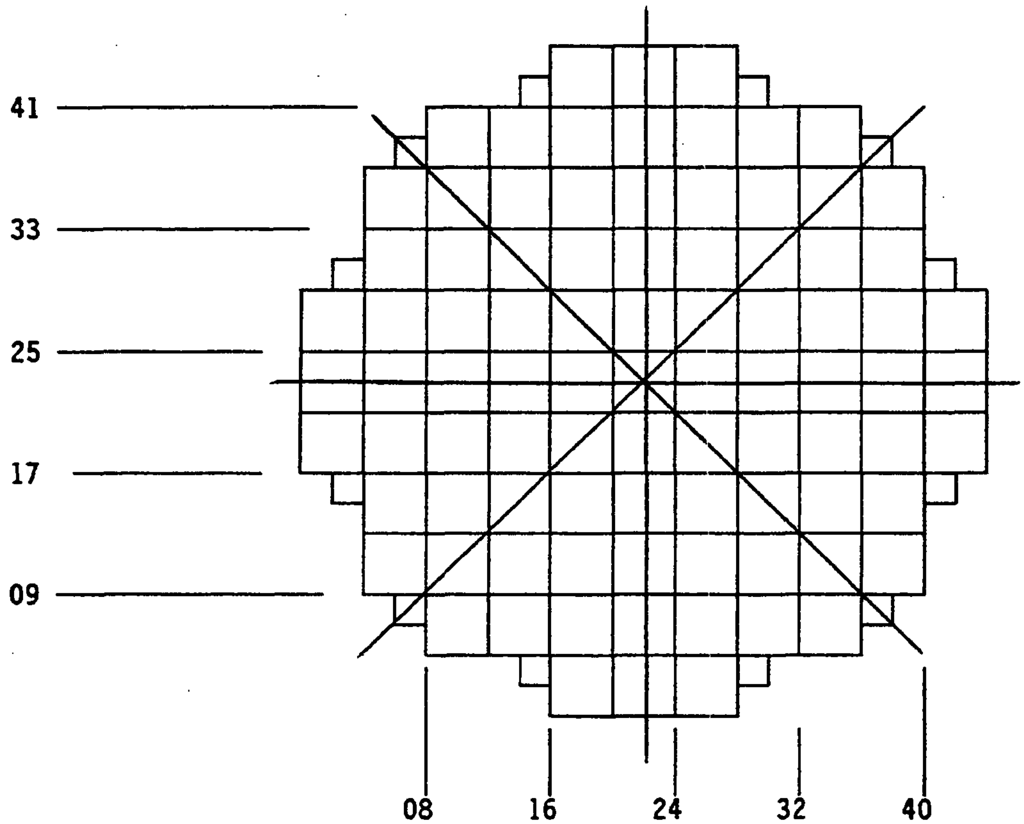


FIGURE A-26. SMALL CORE OCTANT BOUNDARIES

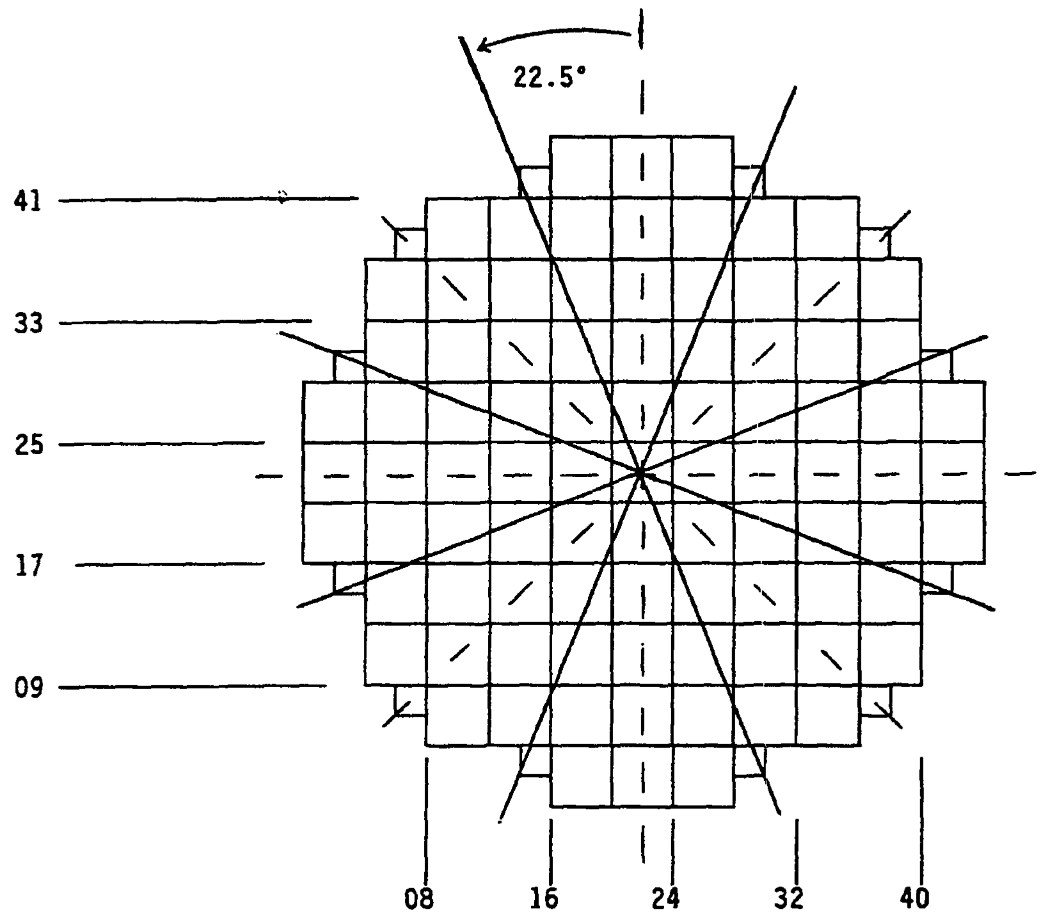
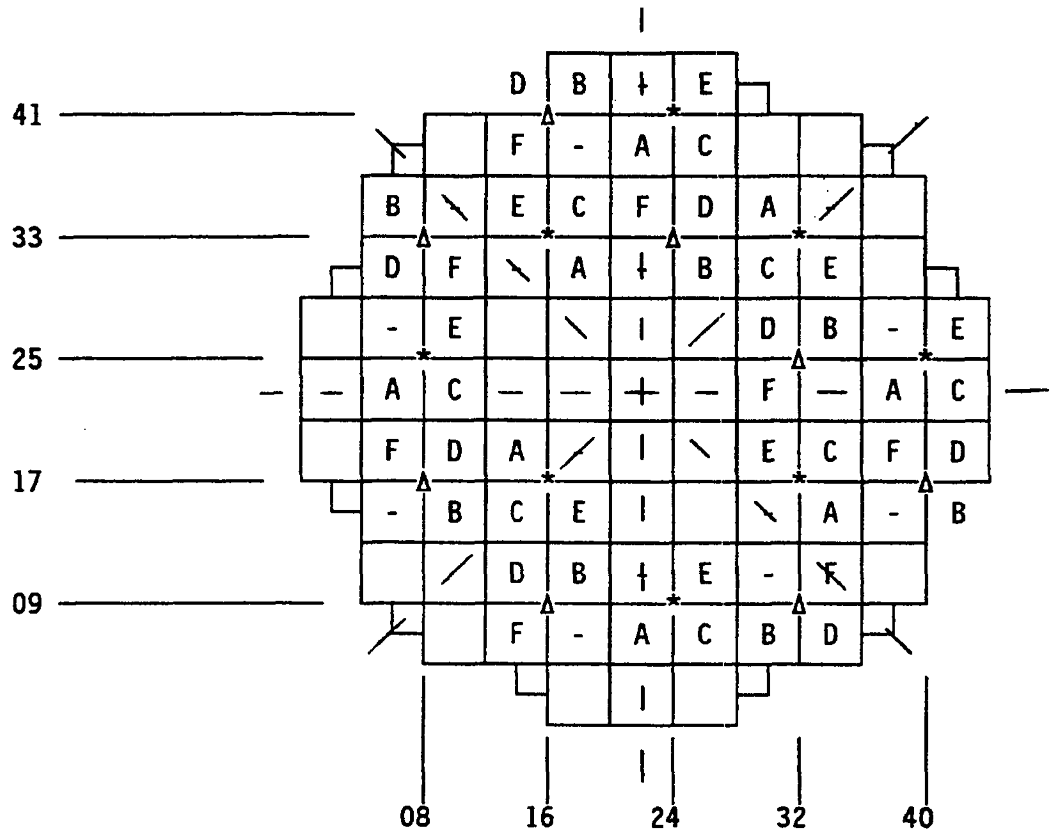
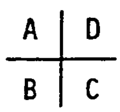


FIGURE A-27. SMALL CORE ALTERNATIVE OCTANT BOUNDARIES





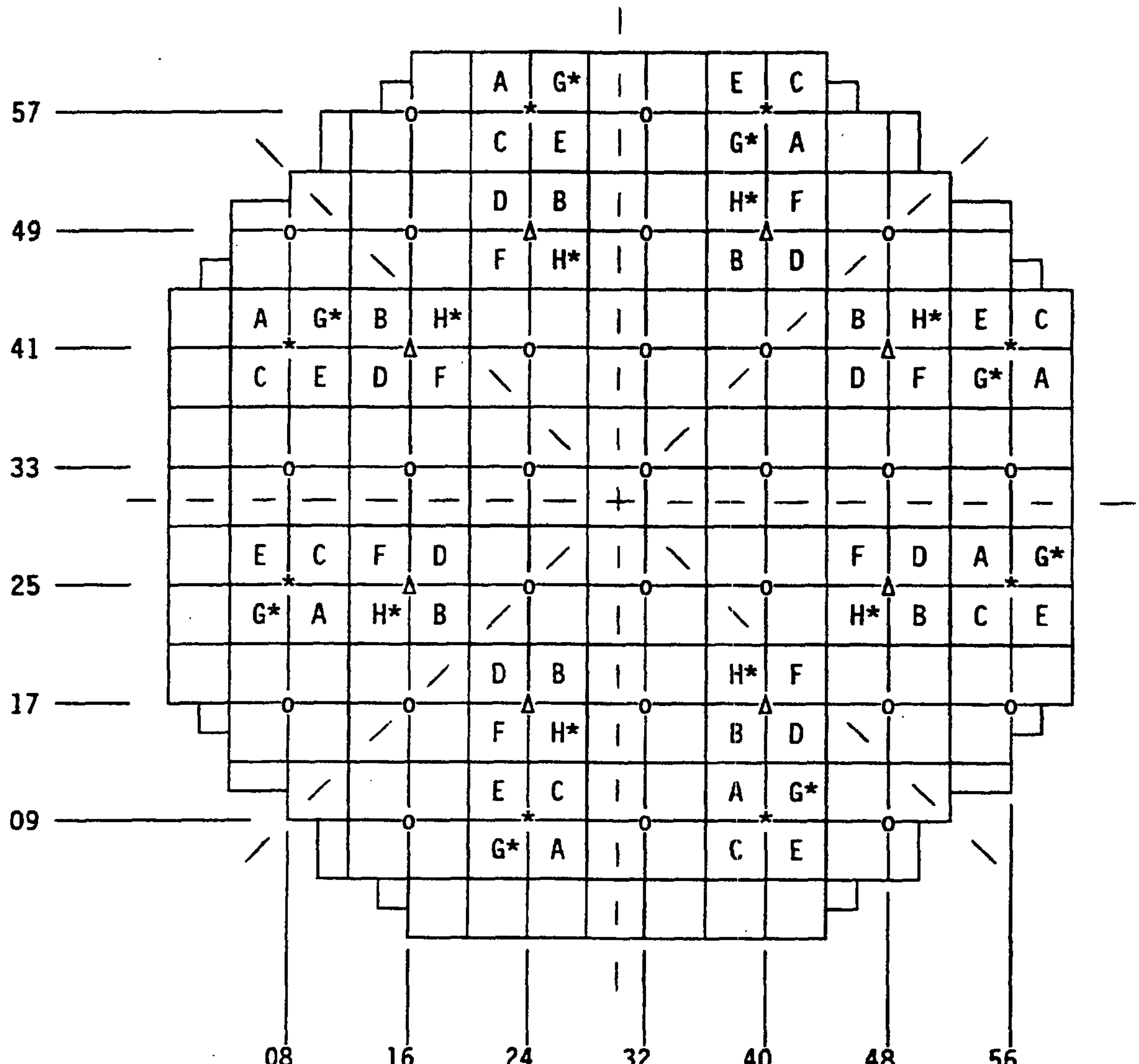
LPRMs providing input to LBS channels

Upper left letter = LBS Channel Assignment for LPRM "A" (bottom)
Lower left letter = LBS Channel Assignment for LPRM "B"
Lower right letter = LBS Channel Assignment for LPRM "C"
Upper right letter = LBS Channel Assignment for LPRM "D" (top)

* = RPS "A" LPRM String

Δ = RPS "B" LPRM String

FIGURE A-28. SAMPLE SMALL CORE LBS ASSIGNMENT SCHEME



A | D
B | C

LPRMs providing input to LBS channels

Upper left letter = LBS Channel Assignment for LPRM "A" (bottom)
Lower left letter = LBS Channel Assignment for LPRM "B"
Lower right letter = LBS Channel Assignment for LPRM "C"
Upper right letter = LBS Channel Assignment for LPRM "D" (top)

G*, H* for BWR/6 only, all other plants have no assignments for these LPRMs

* = RPS "A" LPRM String

Δ = RPS "B" LPRM String

FIGURE A-29. SAMPLE LARGE CORE LBS ASSIGNMENT SCHEME

APPENDIX B

OSCILLATION DETECTION ALGORITHMS

Detection and suppression systems rely on the recognition of oscillations by monitoring LPRM signals. An evaluation of the LPRM signals (or groups of LPRM signals) that determines whether the signal variations are indicative of an instability is called an "oscillation detection algorithm". These algorithms are important in ensuring the early detection of instabilities and subsequent initiation of an ASF before exceeding the MCPR Safety Limit. However, the algorithms must also be able to discriminate against spurious signals to avoid unnecessary initiations of the ASF.

Stability-related neutron flux oscillations in a BWR exhibit known characteristics (e.g., a dominant frequency in the range of 0.3 to 0.7 Hz) that will be used in the design of trip algorithms. As the stability threshold is crossed, oscillations begin and grow to a limit cycle (constant oscillation magnitude). The rate of growth of the oscillations is determined by the instability of the conditions, how rapidly the condition is entered, and also perturbations in the system. In general, the approach to a limit cycle in the magnitude range of interest (prior to trip) is relatively gradual. The periodic nature of oscillations and the fact that oscillation growth must occur before any fuel impact is possible allows an algorithm to be designed that will be able to discriminate between stability-related neutron flux oscillations and other neutron flux variations that are expected to occur in the plant.

The algorithms monitor the trip system channel responses and provide a trip signal if a growing oscillation with sufficient magnitude is detected. The trip system channels may be comprised of individual LPRM signals or groups of signals called cells. Each individual LPRM signal or cell value is evaluated against the algorithm.

The determination that the channel response is oscillating requires the calculation of a time dependent signal average. The characteristic frequency of the oscillations can also be used to confirm that the oscillation peaks and

minima are occurring at intervals which are consistent with the expected frequency range. The determination that the oscillation magnitude is growing requires at least two peaks of oscillation before a trip is actuated. These features have the potential to introduce time delays in the trip logic. Any time delay caused by an algorithm is factored into the basis for the trip setpoints. In general, various oscillations are evaluated relative to the oscillation magnitude reached when the logic initiates a trip. A range of oscillation growth rates is evaluated to confirm that the setpoints are adequate to protect the MCPR Safety Limit.

The basic approach to establishing trip system setpoints is first to determine an acceptably low setpoint such that expected plant evolutions do not result in initiation of an ASF and then confirm that the setpoint provides margin to the MCPR Safety Limit. Therefore, evaluation of plant operating data against potential trip algorithms is the first step in the determination of the setpoints. Basic characteristics of oscillations are already known from test data and operating experience. The trip algorithm can be conceptually designed based on this information. Digitally recorded plant data during various expected plant maneuvers are then analyzed by the trip algorithm to determine the margin to trip for these transients. Desired trip margins are established based on trip avoidance and previous experience with other trip systems.

In addition to the expected plant evolutions that are not required to initiate a trip, digitally recorded data from BWR instability events are evaluated using the trip algorithms. This evaluation confirms that the trip algorithm can accurately identify the occurrence of oscillations and readily detect the necessary characteristics for initiating a trip. The following section discusses examples of oscillation detection algorithms.

B.1 HIGH-LOW-HIGH DETECTION ALGORITHM

The oscillation recognition algorithm is intended to discriminate between true stability-related neutron flux oscillations and other flux variations that may be expected during plant operation. The algorithm design has two primary objectives. The first is to provide a sufficiently low amplitude trip setpoint such that margin is maintained to the MCPR Safety Limit. Second, the

algorithm must be capable of identifying stability-related neutron flux oscillations and discriminating against "false" signals from other expected plant evolutions. This design objective is essential for maintaining reliable power operation and, simultaneously, minimizing unnecessary challenges to the ASF. It is not the objective of the algorithm to avoid tripping due to single failures associated with the neutron monitoring system. For example, the redundancy in the OPRM channel design (i.e., multiple OPRM cells per channel, and four OPRM channels which can be used in a one-out-of-two-taken-twice trip logic) provides the required single failure protection.

To provide a "smart" algorithm, some information must be known about the characteristics of oscillations as well as other expected variations in sensed neutron flux. These characteristics form the basis for selecting the specific algorithm features. The following is a discussion of some of the characteristics that are considered important in the design of the algorithm.

The natural frequency of density wave oscillations in a BWR is directly related to the void transport time. This parameter is well known and does not significantly vary from plant to plant. The natural frequencies that have been observed during instabilities at jet pump BWRs are listed in Table B-1. The expected range of oscillation frequencies is 0.4 to 0.6 Hz based on the plant data. These instability events and tests cover a wide range of power, flow, power distributions, and core inlet subcooling conditions. For conservatism, the algorithm was designed to protect against oscillations with a frequency in the range of 0.3 to 0.7 Hz.

At the onset of an instability event, the oscillations are very nearly sinusoidal in time with a magnitude that dominates the normal steady state noise conditions. Therefore, the oscillations are very "clean" and distinctive and show a smooth trend in time. The increasing magnitude of oscillations can be expressed in the form of a decay ratio or growth rate. Oscillations that are growing initially have a relatively constant growth rate and then reach an equilibrium magnitude (i.e., limit cycle). Examples of limit cycle and growing oscillations are shown in Figures B-1 and B-2.

BWRs exhibit normal neutron flux noise that is the result of perturbations in the reactor system which affect the void reactivity. The noise shows the same dominant frequency as an instability. Plant LPRM data for a condition with a decay ratio of approximately 0.8, as determined by noise analysis, are shown in Figure B-3. Although the same characteristic frequency is evident in the noise signal, the coherence of the signal in time is not as high as for the unstable conditions. The noise signal does exhibit periods of growing amplitude for several successive peaks, although the magnitude is limited.

The neutron flux response during expected plant transients (pressure perturbation and recirculation pump start) is shown in Figures B-4 and B-5. The initial portion of the transient shows an oscillatory behavior with a decaying trend. This type of response is expected for short duration events which introduce a step or rapid change in reactivity. It is the timing and decaying trend of these transients that is important with respect to the algorithm design.

Based on the described characteristics of oscillations and expected plant transients/maneuvers, an initial design of the algorithm has been selected. The basic design concept is to evaluate the signal (e.g., LPRM or OPRM cell) relative to its time-averaged value to determine any oscillatory behavior. The known frequency range of the oscillations is used to detect successive oscillation peaks and valleys such that peaks occurring too close together or too far apart in time are screened. To avoid ASF initiations for initially large amplitude perturbations that decay in time, two successive peaks will be required to produce a trip, with the second peak larger than the first by a specified amount. A maximum trip level is also provided for the second peak.

The signal (e.g., LPRM or OPRM cell) value is filtered to remove high frequency noise components relative to the oscillation frequency (> 5 Hz). The resulting filtered signal is referred to as the conditioned signal value. The conditioned signal value is then filtered (approximately a six second time constant) to produce a time-averaged value indicative of the thermal power in the region where the signal measurement is taken. This value is used as a baseline for comparison with the conditioned value. A relative signal value is

then determined by dividing the conditioned signal value by the time-averaged value. This relative signal value is then used in the algorithm.

The relative signal value (Figure B-6) is first compared to a threshold trip level, S_1 (e.g., $S_1 = 1.10$, which corresponds to an instantaneous signal value 10% higher than the time-averaged value). If the threshold trip level is exceeded, timers are initialized to begin a search for the first peak (P_1) and also for the next valley. Once S_1 is exceeded, the relative signal value is compared against the minimum threshold, S_2 (e.g., $S_2 = 0.93$). If the signal value goes below the minimum threshold level in the allowable time window, timers are then initialized to begin the search for the next peak. The next peak can now trip the associated channel if the peak exceeds one of two trip setpoints.

If the previous conditions have been met, the relative signal value is compared to a trip setpoint, S_3, which is based on the previously detected peak (P_1). The purpose of the S_3 trip setpoint is to initiate a trip for oscillations with very high growth rates. The S_3 trip setpoint is calculated as:

$$S_3 = (P_1 - 1.0) * DR_3 + 1.0 \qquad \text{(B-1)}$$

where

DR_3 = Growth rate factor for S_3 trip setpoint calculation.

Additionally, the relative signal value is compared against an absolute maximum trip setpoint (S_{max}) to protect against very slowly growing oscillations that have a growth rate less than DR_3. One of the two trip setpoints (S_3 or S_{max}) must be exceeded in the allowable time window for a trip to be initiated. If any of the criteria of the algorithm are not satisfied in the required time intervals, the logic is reset and no trip occurs. The relationship between the various setpoints is shown in Figure B-6.

B.2 PERIOD BASED DETECTION ALGORITHM

An alternative detection algorithm that is based on recognition of the thermal-hydraulic oscillation period can be used to support the microprocessor-based systems. This algorithm is based on the observation that the neutron flux of an unstable core will oscillate with a well defined period and that the neutron flux of a stable core will be characterized by random noise.

Detection of the inception of thermal-hydraulic instability will be confirmed by several consecutive equal periods which will also result in an alarm signal. The oscillation amplitude will then be compared against a trip setpoint. Meeting both conditions (both a sustained period and an increasing signal amplitude) will result in a channel trip signal.

The algorithm is based on the following elements:

(1) Real time signal analysis (i.e., LPRM signal response) with sufficiently high signal sampling and computation frequency (large compared to the thermal-hydraulic oscillation frequency).

(2) Recognition of signal peaks and minima.

(3) Screening of oscillation frequency to be within the expected thermal-hydraulic oscillation range (i.e., 0.3 to 0.7 Hz).

(4) Time-averaging of signals to provide a time-dependent baseline.

The channel alarm and trip functions will be produced as follows:

(1) Identify first two signal peaks and calculate the time interval between them. This time interval is defined as the base period (T_0).

(2) If T_0 is within the expected range (i.e., 1.4 to 3.4 seconds), compare subsequent periods to T_0 with tight tolerance (~10% of T_0). Subsequent

periods consist of the time interval between either two consecutive peaks or two consecutive minima.

(3) If T_0 is confirmed N times (e.g., N=4 to 6), produce channel alarm signal, and

(4) If the last peak is above the trip setpoint (e.g., peak/average = 1.1), produce channel trip signal.

If any of the conditions associated with the period confirmation are not met, T_0 is reset to the latest calculated period (based on either two peaks or two minima). T_0 may be adjusted to reflect the average of all confirmed periods associated with an initial base period. This averaging will eliminate failed confirmations due to period variations caused by discrete sampling intervals. Confirmation of T_0, which already resulted in an alarm (step 3), continues until either confirmation fails (e.g., stable condition, noise disturbance) or the second condition (step 4) is met, leading to channel trip signal. The number of confirmations for the alarm and trip functions may differ.

The algorithm is expected to be very responsive to the inception of instability, and to be able to produce a trip signal at a very low oscillation amplitude with a high level of reliability. This algorithm is expected to provide an early alarm for state conditions with high decay ratio prior to any significant oscillation amplitude growth, and therefore may provide time for appropriate operator suppression action.

This algorithm may be applied to analyze a LPRM signal rather than a LPRM-group averaged signal with no increase in spurious scram probability due to LPRM spikes since the period confirmation requirement will not be met as a result of the LPRM spike and T_0 will reset. LPRM and trip channel redundancy can ensure that no reduction in instability recognition capability will result from a spurious spike in a single LPRM.

Table B-1

OBSERVED OSCILLATION FREQUENCIES

Plant	Power/Flow (%/%)	Frequency (Hz)
Vermont Yankee (1981)	51/31	0.43
Caorso (1983)	52/30	0.48
Leibstadt (1984)	73/40	0.58
Leibstadt (1984)	46/29	0.48
Leibstadt (1984)	56/31	0.46
Leibstadt (1984)	53/30	0.45
Leibstadt (1984)	51/31	0.46
Leibstadt (1984)	46/29	0.46
LaSalle-2 (1988)	45/29	0.45

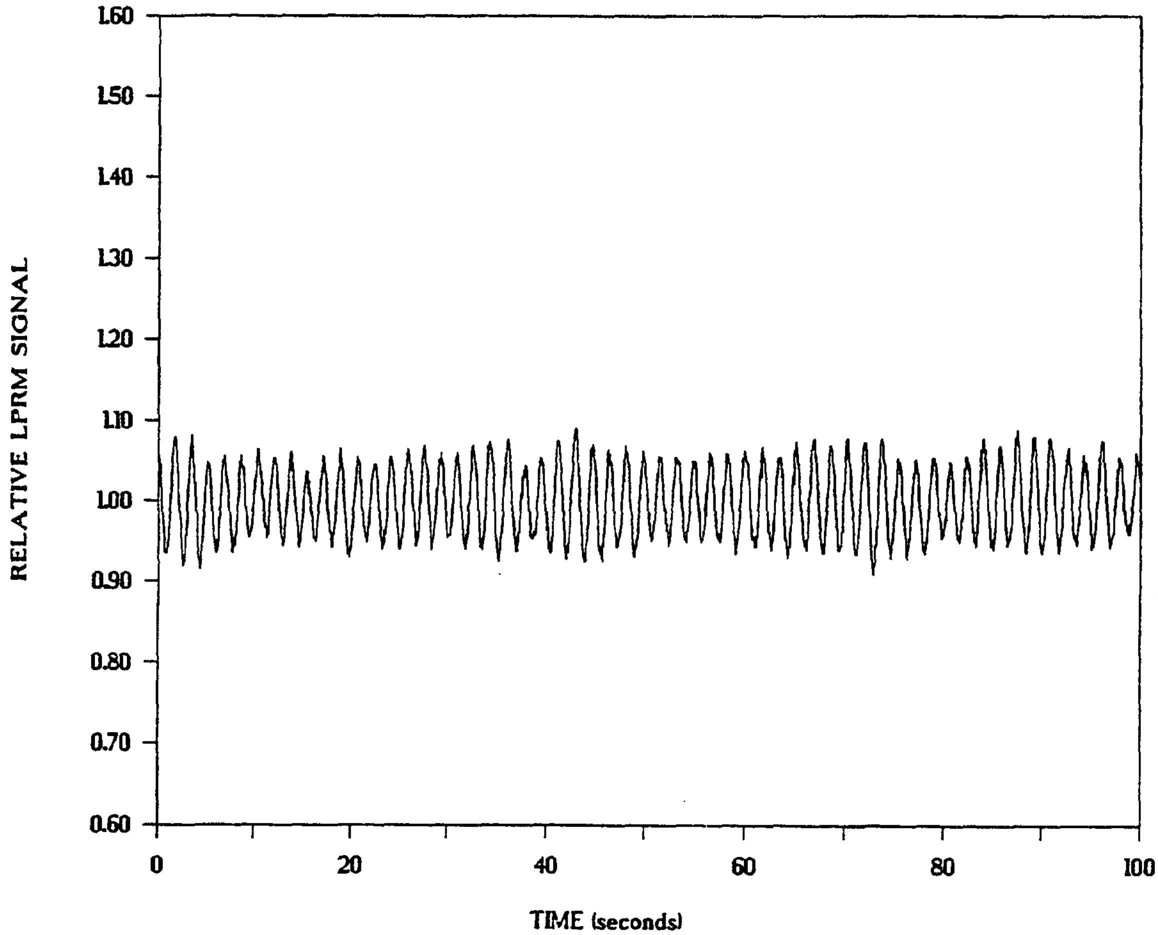


FIGURE B-1. LIMIT CYCLE NEUTRON FLUX OSCILLATION

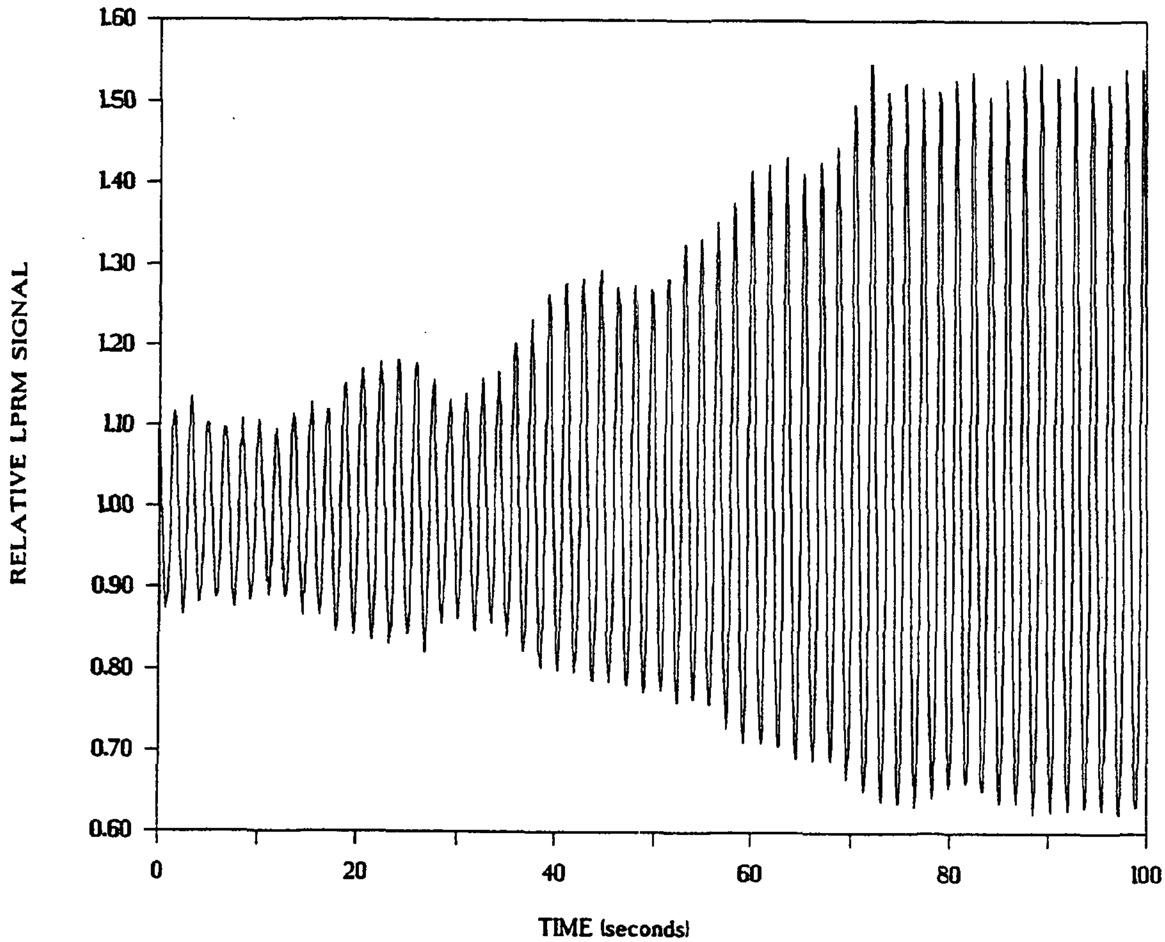


FIGURE B-2. GROWING OSCILLATION SIGNAL

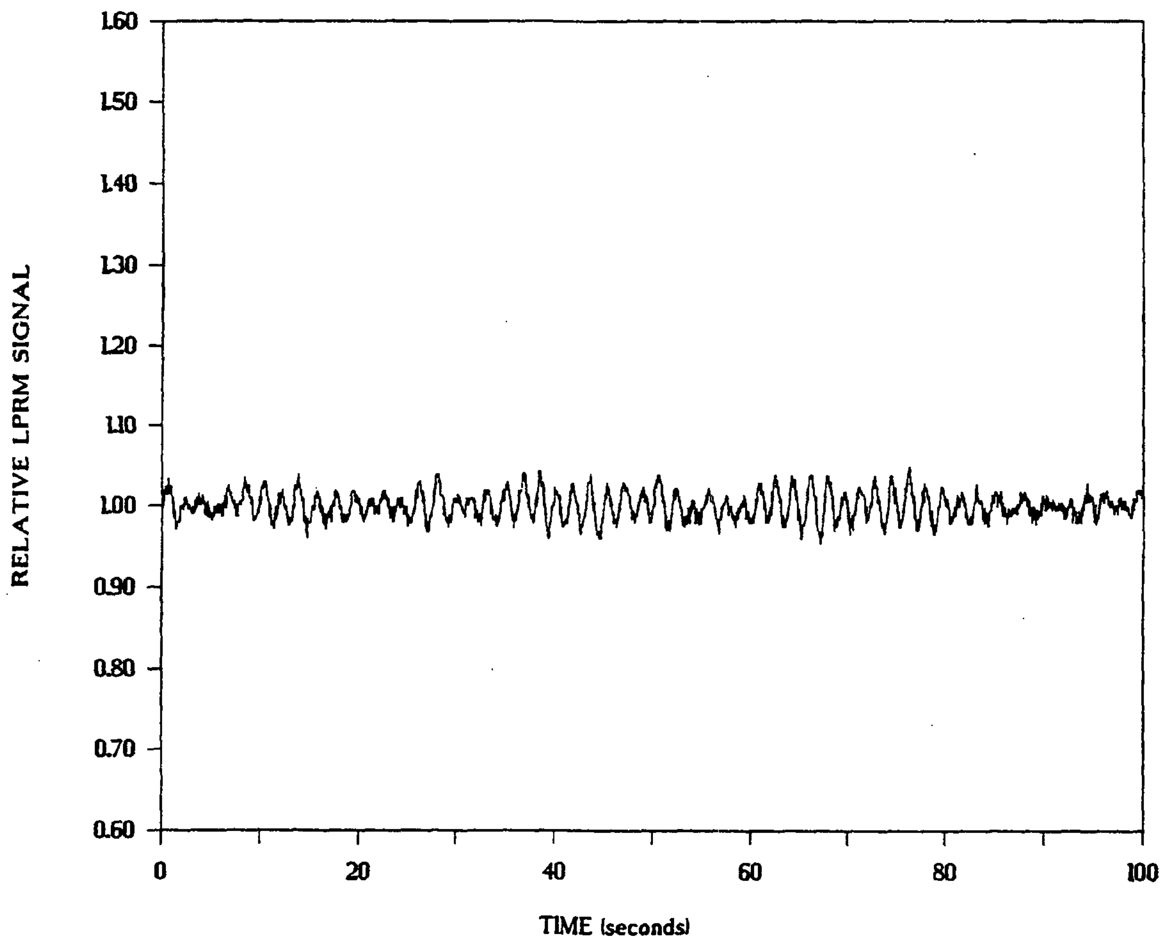


FIGURE B-3. LPRM NOISE RESPONSE - DECAY RATIO = 0.8

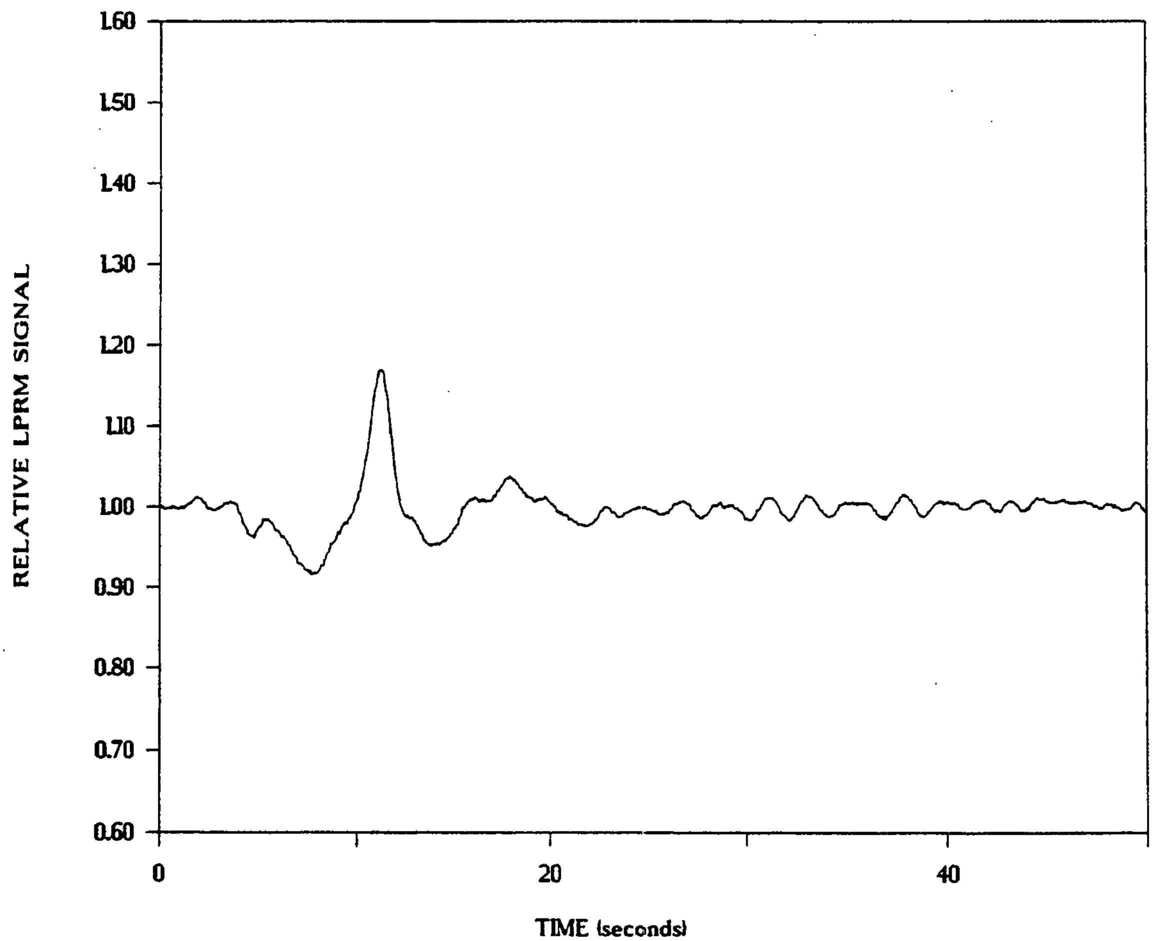


FIGURE B-4. LPRM RESPONSE DURING PRESSURE PERTURBATION

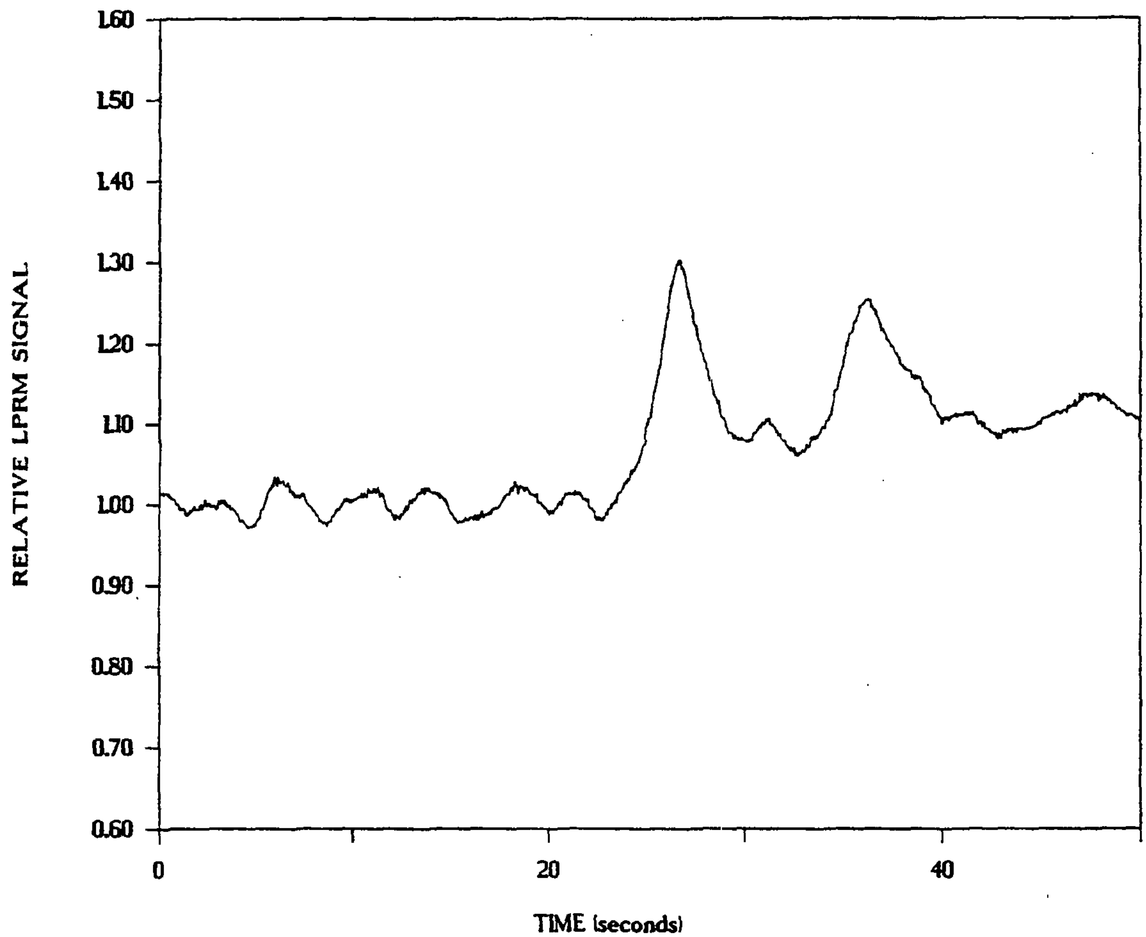


FIGURE B-5. LPRM RESPONSE DURING START OF RECIRCULATION PUMP

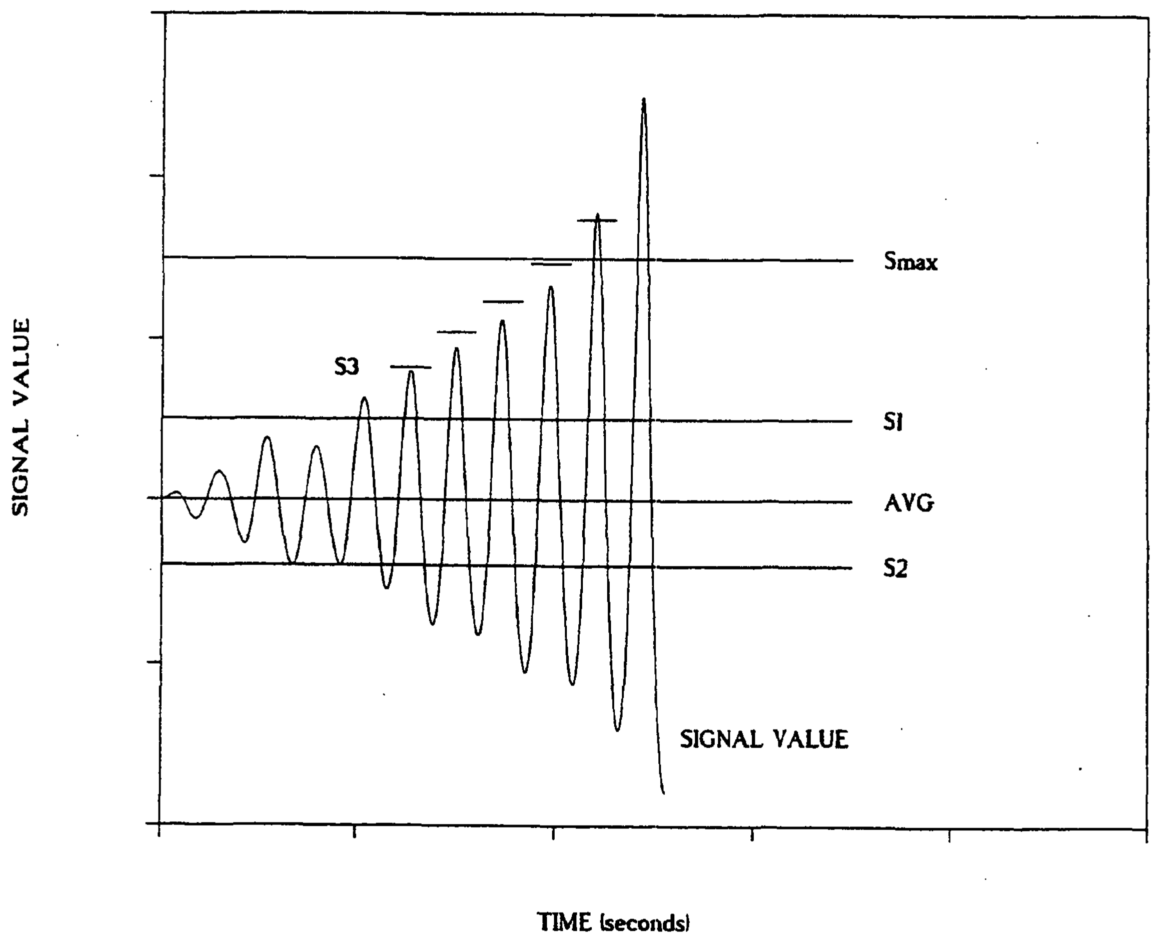


FIGURE B-6. SAMPLE DETECTION ALGORITHM SETPOINTS