ANP-10256NP
Revision 2

Methodology for Analysis of Control Rod Withdrawal Error for BWR Plants with ARTS

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U.S. Nuclear Regulatory Commission
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Nature of Changes

Changes implemented in Revision 1

Item	Page	Description and Justification
1.	Throughout	Changed Framatome ANP to AREVA NP Inc.
2.	v	Clarified the definition of MICROBURN-B.
3.	1-1	Clarified the NRC review expectations.
4.	1-2	Added items to the range of applicability description.
5.	2-1	Added Section 2.0 to describe previously approved methodologies for this event.
6.	3-1	Expanded the description of the methodology.
7.	3-3	Added Section 3.3 to compare proposed and previous methodologies.
8.	3-3	Added Section 3.4 to identify conservatisms of the analysis.
9.	3-5	Added a schematic of the calculation process.
10.	4-5	Added detail to description of LHGR evaluation.
11.	6-1	Added items to the range of applicability description.

Changes Implemented in Revision 2

Item	Page	Description and Justification
1.	All	Proprietary label removed

Contents

Tables

Figures

This document contains a total of 47 pages.

Nomenclature

ANFB	AREVA NP critical power correlation
AOO	anticipated operational occurrences
APRM	average power range monitor
ARTS	average power range monitor, rod block monitor, and technical specification improvement
ATRIUM-10	AREVA NP 10x10 reload fuel
BWR	boiling water reactor
CRWE	control rod withdrawal error
COLR	core operating limits report
EPU	extended power uprate
GE	General Electric
HPSP	high power setpoint
IPSP	intermediate power setpoint
LHGR	linear heat generation rate
LPRM	local power range monitor
LPSP	low power setpoint
MELLLA	maximum extended load line limit analysis
MELLLA+	maximum extended load line limit analysis plus
MCPR	minimum critical power ratio
MICROBURN-B	AREVA NP reactor physics code for BWR reactors
MICROBURN-B2	AREVA NP reactor physics code for BWR reactors
OLMCPR	operating limit MCPR
RBM	rod block monitor
RWL	rod worth limiter
SLMCPR	safety limit MCPR
SPCB	AREVA NP critical power correlation

1.0 Introduction and Summary

This report presents a method for the analysis of a Control Rod Withdrawal Error (CRWE) event in BWR/3, 4, and 5 plants that have installed ARTS (average power range monitor (APRM), rod block monitor (RBM), and technical specifications improvement). A statistical approach is used to determine the minimum critical power ratio (MCPR) results and a deterministic approach is used to determine the linear heat generation rate (LHGR) results. The CRWE MCPR results are presented as a function of the ARTS power dependent RBM setpoints.

The statistical methods are similar to the methods previously approved for BWR/6 reactors in References 6 and 7. The purpose of this topical report is to obtain NRC approval to use the statistical approach described herein for BWR/3,4 and 5 plants with ARTS.

The CRWE transient is hypothesized as an inadvertent reactor-operator–initiated-withdrawal of a single control rod from the core. Withdrawal of a single control rod has the effect of increasing local power and core thermal power which lowers the MCPR and increases the LHGR in the core limiting fuel rods. The CRWE transient is terminated by control rod blocks which are initiated by the RBM system or when the control rod is fully withdrawn, i.e., unblocked.

The statistical CRWE methods have been used to calculate CRWE results for two example BWR reactors with the ARTS RBM system. The analyzed conditions include actual and projected core designs that are representative of current and expected future operation of the reactors. For one of the reactors, the analyzed conditions include plant operation at 20% extended power uprate (EPU) and operation in the maximum extended load line limit analysis (MELLLA+) region of the power/flow map.

The range of applicability of the CRWE statistical methodology is summarized below:

- The statistical methodology for evaluation of the CRWE event is applicable for BWR/3, 4, and 5 plants that have installed the ARTS RBM system.
- The method is applicable to CRWE analyses performed with NRC approved reactor physics codes including the AREVA MICROBURN-B and MICROBURN-B2 codes (References 2 and 3).
- The method is applicable for CPR calculations performed with NRC approved critical power correlations including the AREVA SPCB and ANFB correlations (References 4 and 5) and future NRC-approved CPR correlations.

- Control rod withdrawal rates must be slow enough to approximate with the steady state reactor physics code. Rod withdrawal rates up to 4 inches per second are reasonably estimated with steady state reactor conditions.

- The power level increase for the control rod withdrawal event is dominated by the control rod worth. The reactivity increase during the time of the withdrawal may impact the steady state assumption. Total power increases for the event are limited to 10% of rated core thermal power.

The methodology will be performed on a plant-specific basis. A review of the plant specific analysis will be performed each cycle to determine the applicability of the plant specific results to that cycle. A new analysis will be performed if it is determined that the previous results are not applicable. The analysis performed for one plant may be applied to another plant if the plant characteristics are comparable.

The unique feature of this proposed methodology relative to previously approved methodologies for this event is the statistical treatment of LPRM failures in the determination of the rod block response.

2.0 Previous Methodologies

2.1 *BWR-3/4/5 Analysis*

Analysis of the control rod withdrawal event for BWR-3/4/5 plants has been approved in a previous topical report, XN-NF-80-19(P)(A). The method used the steady state reactor physics code. The following steps are demonstrated schematically in Figure 2-1.

1. The first step of the analysis is to perform a control rod step through of the cycle to identify core locations at which control rods will be inserted at full power with nearby assemblies close to the thermal operating limits.

2. Second, a core exposure and control rod is selected for the first control rod withdrawal calculation. For the start of the calculation, the control rod to be withdrawn is placed in the fully inserted position and the surrounding control rods are adjusted as required to place the fuel near the transient rod conservatively close to the operating limit.

3. The control rod is withdrawn in steps of six inches. At each step the critical power ratio (CPR) for each assembly and the rod block monitor (RBM) response for each channel are calculated. The RBM response calculations include the effects of failed or out of service LPRM detectors.

4. The ΔMCPR at each step of control rod withdrawal is calculated by subtracting the MCPR at the step from the starting MCPR with the control rod fully inserted. The ΔMCPR is calculated for each fuel type in the core.

5. The limiting condition is the minimum increase in the RBM response from the start of the control rod withdrawal which normally results from a failure of the LPRM detectors located close to the control rod being withdrawn.

6. The ΔMCPR occurs at the point where the limiting RBM response reaches the rod block trip setting.

7. Steps 2 through 6 are repeated for other control rods and core exposures to determine the maximum ΔMCPR for each fuel type as a function of rod block set point.

2.2 *BWR/6 Analysis*

BWR-6 plants do not have an RBM system and instead use a rod worth limiter (RWL) system which limits the control rod movement to one foot or two foot withdrawals depending on core power level. The methodology for BWR-6 plants was described in the topical report XN-NF-825(P)(A). The XN-NF-825(P)(A) methodology also uses a steady state reactor physics code to determine the power increase and corresponding decrease in MCPR for a control rod withdrawal. The XN-NF-825(P)(A) methodology however uses a statistical approach by evaluating a large number of operating conditions using nominal control rod patterns while the XN-NF-80-19(P)(A) methodology uses a conservative deterministic methodology. The population of conditions analyzed cover the full range of the core operating map for a series of reactor cycles to cover a wide range of control rod configurations and fuel types.

The objective of the BWR/6 RWE generic transient analysis was to determine statistically bounding (95% probability/95% confidence) values for changes in the core limiting MCPR from minimum CPR values calculated before and after a hypothesized RWE transient event. Cycle exposure points between 0 MWd/MT and the exposure corresponding to the peak core reactivity were modeled in this analysis. Using core exposure points from BOC to peak core reactivity ensured maximum control rod densities to give initial control rod positions between 0 and 16 notches. Deep control blade insertion give maximum rod worth and maximum changes in the CPR when withdrawn.

Changes in MCPR as a function of power level for one foot and two foot withdrawals were calculated from projected control rod patterns and operating conditions for multiple cycles that encompassed the entire core operating map including the extended operating domain. Over 600 data points were generated in order to define the initial MCPR as a function of power. The same techniques used to characterize the initial MCPR were applied to determine the 95/95 tolerance limit curves as a function of power for the delta MCPR corresponding to one and two foot withdrawals.

A schematic of this methodology is presented in Figure 2-2.

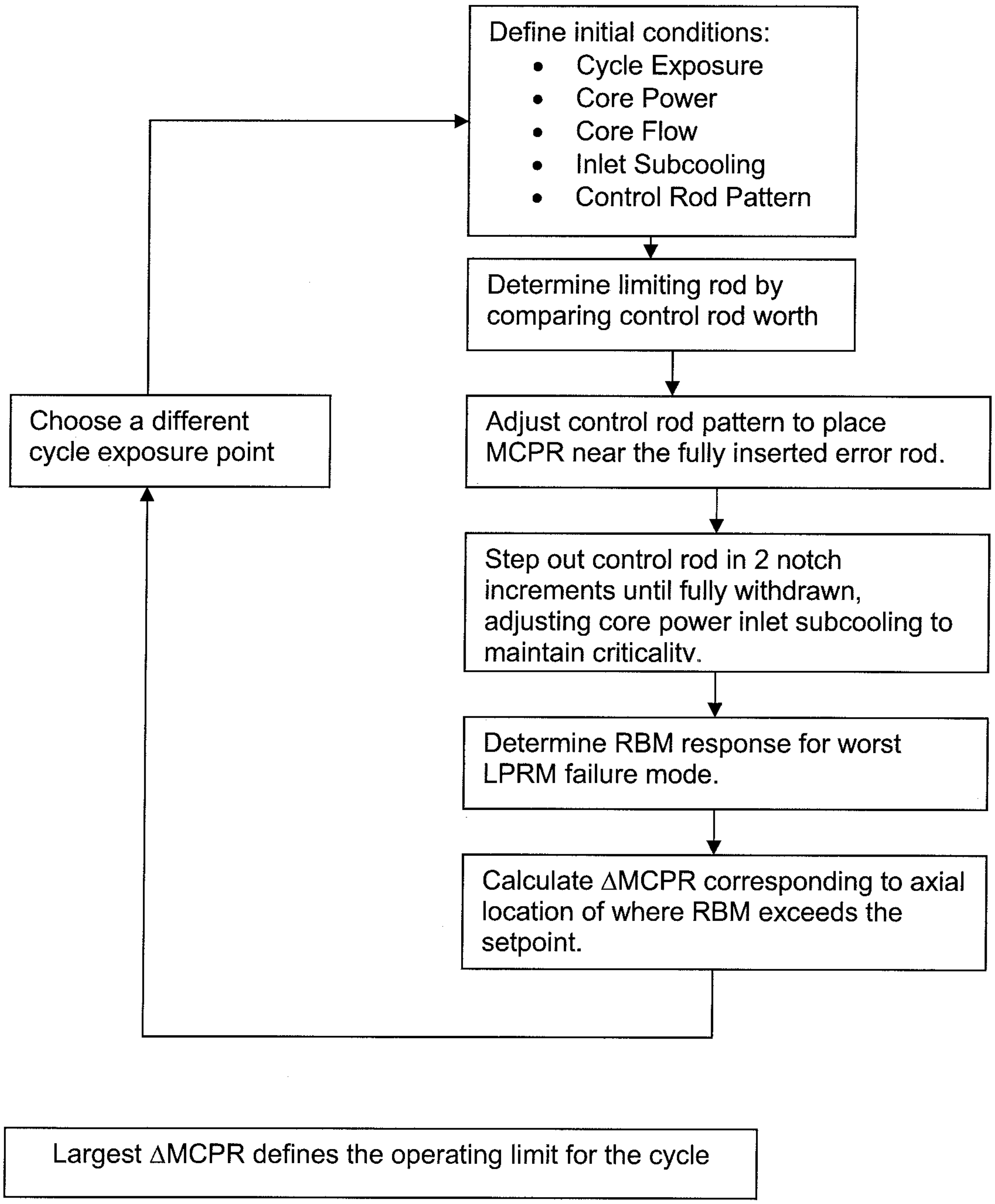

Figure 2-1 Schematic for Deterministic Methodology Applied to BWR/3,4 and 5 Plants

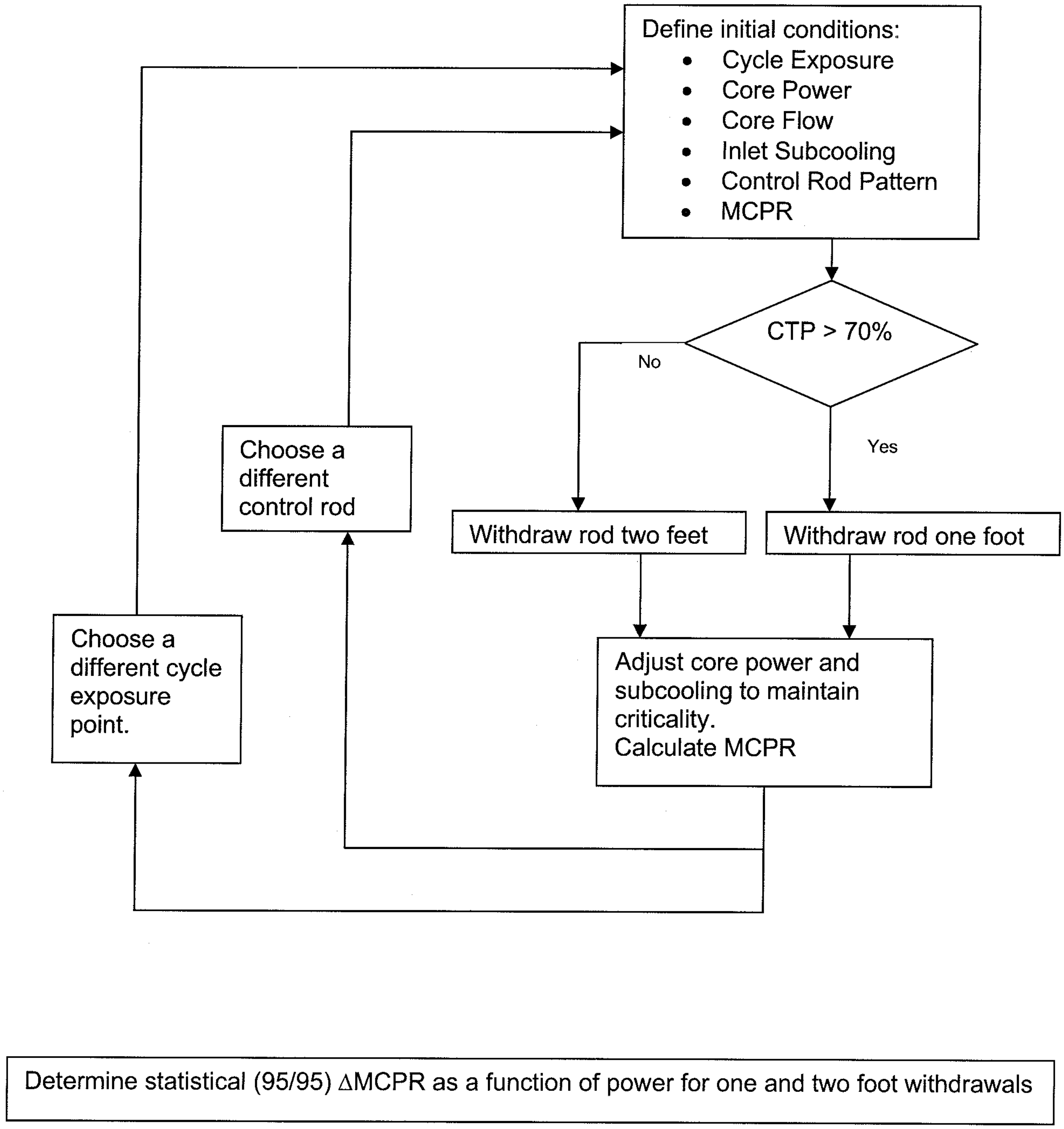

Figure 2-2 **Schematic for Statistical Methodology Applied to BWR/6 Plants**

3.0 Methodology Description

The CRWE methods for the BWR/3, 4, and 5 reactors that have installed the ARTS RBM system are described in this section. Figure 3.1 presents a schematic of the methodology described in more detail in this section.

The NRC review of References 6 and 7 approved the statistical CRWE method and results for BWR/6 reactors.

3.1 *CRWE Calculation Method*

The CRWE calculations are performed with either the MICROBURN-B2 or MICROBURN-B reactor physics codes documented in References 2 and 3. MICROBURN-B2 and MICROBURN-B are steady-state reactor physics codes that determine the global neutron flux distribution by solving the three-dimensional, two group neutron diffusion equation based upon the homogenized lattice nuclear data generated by the lattice/spectrum code CASMO-4. The codes model all assemblies in the core with radial and axial mesh sizes corresponding to the radial dimension of the assembly, i.e. one node per assembly radially and ~25 nodes per assembly axially. A two-phase thermal-hydraulic model is capable of calculating nodal coolant flow and density distributions under conditions ranging from cold shutdown to hot full power operation (including extended power uprate and other extended flow domains at power uprate conditions (MELLLA+ being an example)). Control blade insertion is modeled explicitly. The rod withdrawal is modeled in a series of steady-state solutions with a heat balance to determine the hydraulic changes associated with the reactor power changes. Steady-state solutions are justified because the rate of power increase for a rod withdrawal is slow compared to the time constants for heat transfer and delayed neutrons.

The choice of which code to use for the CRWE analysis, MICROBURN-B or MICROBURN-B2, is dependent upon which code is being used to support plant operation in other analysis areas.

The initial conditions for the CRWE calculations are based on the cycle projection step-through rod patterns and exposure history for each cycle. At the selected core state points for analysis (cycle exposure, power, flow, and rod pattern), the rod withdrawal error rods are selected based on MCPR limiting fuel assemblies located near inserted control rods. Additional conservative calculations are also performed with the error rod and symmetrical rods initially located in the fully inserted position.

During the CRWE transient, the reactor operator is assumed to ignore the LPRM and RBM alarms and continue to withdraw the control rod until the control rod motion is stopped by the RBM rod block. To model the rod withdrawal, full core reactor calculations are performed at approximately every six inches of rod travel from the starting control rod position to the fully withdrawn control rod position. As the control rod is withdrawn, the local and reactor power increases are calculated with the reactor physics code. The power increases result in a decrease in the MCPR, an increase in the LHGR, and an increase in the RBM response in the area of the core where the control rod is withdrawn.

The MICROBURN-B2 and MICROBURN-B codes calculate the MCPR, LHGR, and RBM response at each control rod position and write the results to output files. All of the control rod withdrawal results for a specific initial power and fuel type are combined to perform a 95/95 statistical analysis of the MCPR results. The effect of LPRM detector failure (bypass) on the RBM response is modeled by selecting the number of LPRM failures using a 15% LPRM failure probability. The 15% failure rate is atypically high based on actual LPRM failure experience. The number of failed LPRM detectors and the LPRM configuration are assumed to be random.

The unblocked CRWE results are used to verify that the transient LHGR limits are not exceeded and to determine MCPR values where the RBM system is not required for reactor operation.

3.2 *Statistical Method*

The data for the 95/95 statistical analysis are the maximum calculated ΔMCPR/initial MCPR values as each control rod of interest is withdrawn from the core. For each CRWE calculation, 100 simulated ΔMCPR/initial MCPR results are generated by randomly varying the location and number of in service LPRM detectors in a RBM channel.

An examination of the example reactor data base showed that the results were slightly skewed towards smaller ΔMCPR/initial values. Based on this observation, a one sided distribution free tolerance limit method is used to calculate the 95/95 statistical results for the ΔMCPR/initial MCPR values as a function of RBM setpoint for each analyzed power level, fuel type, and reactor. The statistical method is summarized in Reference 8. The CRWE event is dependent on several variables including the reactor operating state point (power, flow, cycle exposure), the control rod pattern, the error rod selection, the RBM channel in service assumption, and the LPRM in service assumption. Because of the large number of independent analysis variables, the 95/95 statistical approach is appropriate for the MCPR analysis of the CRWE event. The statistical value is the initial MCPR necessary for 95% confidence that the SLMCPR will not be violated in 95% of possible rod withdrawals. The CRWE operating limit $MCPR_{95/95}$ values are calculated from the 95/95 ΔMCPR/initial MCPR values using the following formula:

The mean, standard deviation, and 95/95 ΔMCPR/initial MCPR values are tabulated in Section 3.0 with the CRWE operating limit $MCPR_{95/95}$ results.

3.3 *Comparison to previous Methodologies*

The methodology described in this report is similar to the methodologies presented in XN-NF-80-19(P)(A) and XN-NF-825(P)(A) in that it uses a steady state reactor physics code to evaluate the MCPR at various reactor conditions. As in the XN-NF-80-19(P)(A) methodology the RBM response is calculated at each axial position as the control rod is withdrawn to identify the ΔMCPR associated with that axial position. The methodology uses nominal control rod patterns from multiple cycles in the same way as the XN-NF-825(P)(A) methodology. The primary difference in this method relative to the XN-NF-825(P)(A) method is the use of the RBM response to determine the final position of the control rod by statistically analyzing the LPRM detector failure configuration. The ARTS implementation defines various RBM settings for different power ranges rather than the single setpoint used in the XN-NF-80-19(P)(A) methodology. These ranges are specifically evaluated in the new methodology.

The statistical technique of defining the 95/95 value is a distribution free tolerance limit rather than the normal distribution assumed in the XN-NF-825(P)(A) methodology.

3.4 *Conservatisms*

As discussed above, additional cases were added to the calculation database to increase the control rod worth of the error control rod by fully inserting the symmetric control rods. This not only increases the control rod worth but increases the change in localized power surrounding the error control rod at the termination of the event.

Shallow rods in the normal control rod patterns are not included in the population of error rods since they have been shown to result in significantly lower delta CPR's. Thus for all possible control rod withdrawal events the population of this analysis is biased to those resulting in higher delta CPR's.

The random failure rate for the LPRM failures in a rod block configuration was chosen at a very conservative rate of 15% failure probability.

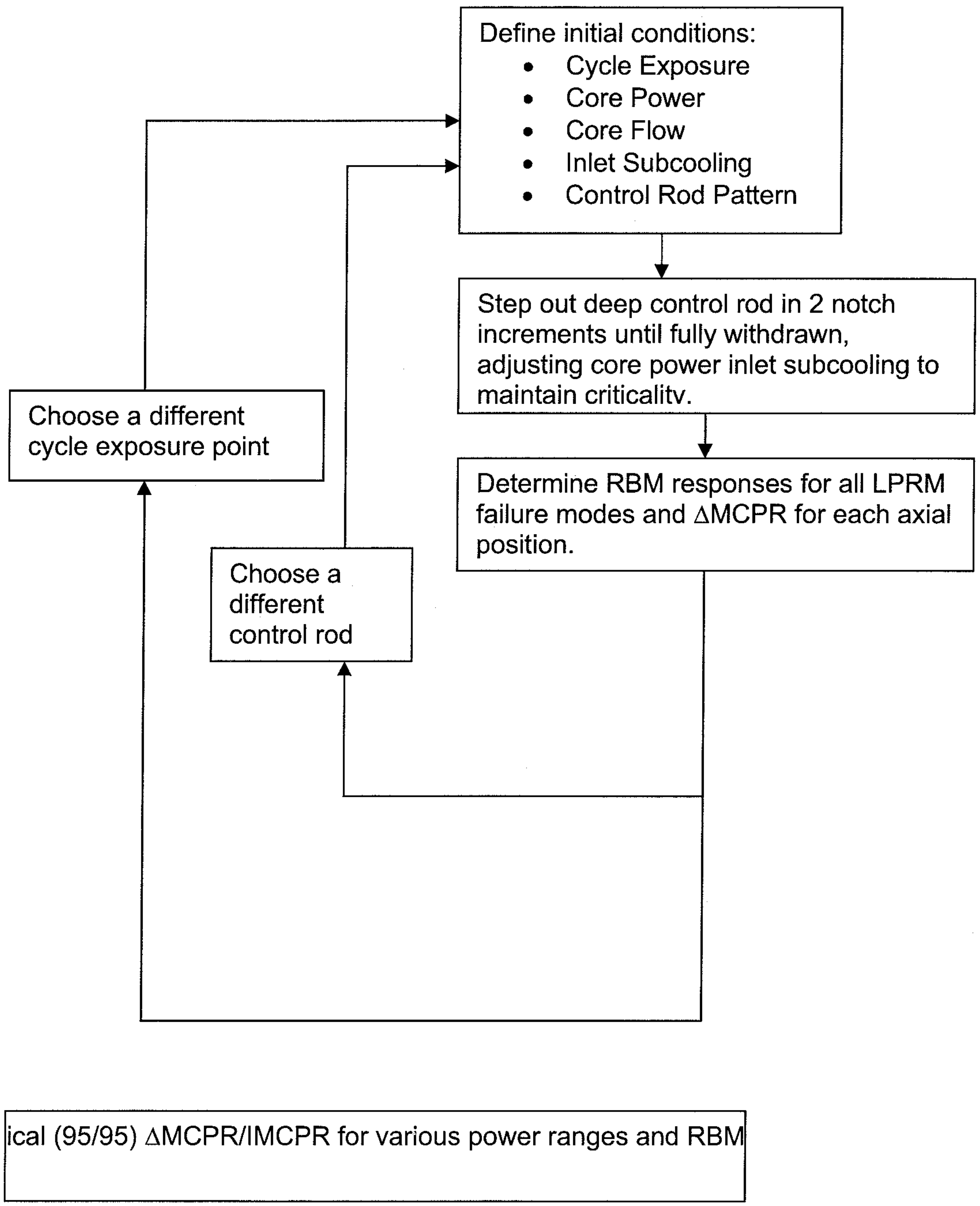

Figure 3-1 Schematic Description of CRWE Methodology for ARTS Plants

4.0 Sample Problem Control Rod Withdrawal Results

The ARTS RBM system uses power dependent RBM trips and different LPRM inputs compared to the original RBM system. The LPRM assignments associated with ARTS make the RBM system more sensitive to control rod withdrawals. For all control rods surrounded by four LPRM strings, each RBM channel uses the input from two D level LPRMs, four C level LPRMs, and two B level LPRMs as shown on Figure 4.1. The A level LPRM detectors (located near the core bottom) that are less sensitive to control rod withdrawals are not used in the ARTS RBM system. The control rod and LPRM reactor core locations and typical ARTS RBM LPRM detector assignments for Reactor B are shown in Appendix A on Figures A.1 and A.2 as an illustration.

Representative CRWE analyses have been performed for two example reactors that use the ARTS RBM system. Reactor B is a D Lattice BWR/4 with a core size of 764 assemblies and partial symmetric reload fuel. Reactor C is a C Lattice BWR/4 with a core size of 408 assemblies (the C Lattice fuel is symmetric). The analyses are for selected CRWE cases in transition and equilibrium cycles. The transition cycles have mixed cores of ATRIUM-10 and GE fuel. The equilibrium cycles have a full core of ATRIUM-10 fuel. The control rod withdrawal calculations are performed in full core geometry with either the MICROBURN-B2 or MICROBURN-B reactor physics codes. The initial conditions for the CRWE calculations are based on the cycle projection step-through control rod patterns. The control rod withdrawal error control rods are selected based on limiting fuel assemblies located near inserted control rods.

The CRWE calculations are performed at representative reactor conditions of 100% power/nominal flow, 65% power/45% flow (64% power/45% flow for Reactor C), and 40% power/45% flow. The nominal flow for the 20% EPU condition represents operation in the MELLLA+ region of the power/flow map (see Figure 4.2).

For the 65% and 40% power cases, the assumed flow of 45% of rated represents the flow on reactor startups. The 100% power and 65% power CRWE calculations are performed with equilibrium xenon. The 40% power CRWE calculations are performed with no xenon.

The RBM response for the CRWE is calculated from either the MICROBURN-B2 or MICROBURN-B output files. The effect of LPRM detector failure (bypass) on the RBM response is modeled by selecting the number of LPRM failures using a 15% LPRM failure probability. Up to one-half of the LPRM detectors in an RBM channel are allowed to be bypassed before the channel is required to be inoperative. For each RBM channel with eight LPRM inputs, the bypass of zero through four detectors is considered in the statistical analysis. The number of failed LPRM detectors and the LPRM configuration is assumed to be random.

4.1 *MCPR Results*

For each CRWE calculation, the MICROBURN-B2 and MICROBURN-B calculated maximum MCPR change (ΔMCPR) divided by the respective initial MCPR for each fuel type is calculated as a function of control rod withdrawal position. The maximum ΔMCPR/initial MCPR (Δ/I) output is combined with the RBM output to produce mean ΔMCPR/initial MCPR and standard deviation values as a function of RBM setpoint. The RBM setpoint values given in this report are the analytical unfiltered values. The effect of the RBM signal filter on the CRWE is presented in Section 4.0.

The CRWE $MCPR_{95/95}$ Operating Limit values that bound 95% of the results with 95% confidence is calculated as a function of RBM setpoint and fuel type at the different analyzed reactor conditions using the following formula:

4.1.1 ATRIUM-10 Fuel MCPR Results

A range of cases were evaluated for Reactor B including 119 MICROBURN-B2 control rod withdrawal cases at 100% power/nominal flow, 66 control rod withdrawal cases at 65% power/45% flow, and 40 control rod withdrawal cases at 40% power/45% flow. A similar range of cases were evaluated for Reactor C including 69 MICROBURN-B rod withdrawal cases at 100% power/nominal flow, 28 control rod withdrawal cases at 64% power/45% flow, and 16 control rod withdrawal cases at 40% power/45% flow. A control rod withdrawal case is the withdrawal of one control rod in the core at a specific reactor condition (rod pattern, power, flow, cycle, and cycle exposure). For each MICROBURN-B2 or MICROBURN-B control rod withdrawal case, 100 simulated control rod withdrawal cases are generated by randomly varying the location and number of failed LPRM detectors in the RBM channel which is in service with the other RBM channel bypassed.

The statistical output summaries for the ATRIUM-10 fuel in Reactor B are shown in Tables 3.1, 3.2, and 3.3 for the three power/flow conditions analyzed. The statistical output summaries for the ATRIUM-10 fuel in Reactor C are shown in Tables 3.4, 3.5, and 3.6 for the three power/flow conditions analyzed. The ΔMCPR/initial MCPR values are averaged over all rod withdrawal cases and all simulated (failed LPRM) cases to obtain the mean values shown in the tables.

The $MCPR_{95/95}$ Operating Limit values for Reactor B and C in the tables are for an SLMCPR value of 1.09. The $MCPR_{95/95}$ results for other SLMCPR values can be obtained by multiplying the $MCPR_{95/95}$ values in the tables by the ratio SLMCPR/1.09.

4.1.2 GE Fuel MCPR Results

The CRWE MCPR results for the GE fuel are calculated for the first transition cycle where the GE fuel is in the second cycle of irradiation. A total of 24 MICROBURN-B2 control rod withdrawal cases were analyzed at 100%power/nominal flow, 24 control rod withdrawal cases were analyzed at 65% power/45% flow, and 14 control rod withdrawal cases were analyzed at 40% power/45% flow for the GE fuel in Reactor B. A total of 10 control rod withdrawal cases were analyzed at 100% power/nominal flow, 10 control rod withdrawal cases were analyzed at 64% power/45% flow and 8 control rod withdrawal cases were analyzed at 40% power/45% flow for the GE fuel in Reactor C. A total of 100 simulated failed LPRM cases were generated for each control rod withdrawal case.

The statistical output summaries for the GE fuel in Reactor B are shown in Tables 3.7, 3.8, and 3.9 for the three power/flow conditions analyzed. The statistical output summaries for the GE fuel in Reactor C are shown in Tables 3.10, 3.11, and 3.12 for the three power/flow conditions analyzed. An SLMCPR value of 1.09 was used to generate the GE fuel results in the tables.

4.1.3 MCPR Summary Results

The CRWE $MCPR_{95/95}$ Operating Limits have been determined from Tables 3.1 through 3.12 for the ATRIUM-10 and GE fuel in the two example reactors as a function of percent rated power and RBM setpoint for an SLMCPR of 1.09. Representative ARTS analytical unfiltered RBM setpoints are shown in Table 3.13. The full power results for Reactor B are for power levels from 105% to 120% of the original rated reactor power. The maximum CRWE $MCPR_{95/95}$ results for Reactor B are shown in Table 3.14 for the representative RBM setpoints from Table 3.13. The full power results for Reactor C are for rated power. The maximum CRWE $MCPR_{95/95}$ results for Reactor C are shown in Table 3.15 for the same representative RBM setpoints. The difference between the CRWE $MCPR_{95/95}$ values for the two reactors is small.

The CRWE results combined with the SLMCPR and the $MCPR_p$ limits for the cycle can be used to select the RBM setpoints that are documented in the core operating limits report (COLR) for the CRWE event.

The upgraded performance of the ARTS RBM system makes the CRWE event non-limiting for selected RBM setpoints compared to other AOO events. The LPRM assignments make the ARTS RBM system more sensitive to control rod withdrawals. The CRWE MCPR operating limits can be compared with the limiting cycle specific transient $MCPR_p$ limit and SLMCPR to verify that the CRWE is a non-limiting event for a specific set of RBM setpoints. For example, representative $MCPR_p$ and CRWE $MCPR_{95/95}$ limits are shown on Figure 4.3 for Reactor B. A comparison of the curves on Figure 4.3 shows that the CRWE $MCPR_{95/95}$ limit is bounded by the $MCPR_p$ limit for the representative RBM setpoints of 114% (high power), 119% (intermediate power) and 124% (low power). At low reactor powers, the CRWE event is far from being MCPR limiting as shown on Figure 4.3 and the RBM system is not required to be in service below the RBM low power setpoint.

4.1.4 <u>Unblocked MCPR Results</u>

The unblocked CRWE MCPR results are calculated to determine MCPR values where the RBM system is not required to be in service. The RBM system operability requirements are included in the reactor specific COLR for a given cycle. The maximum unblocked MCPR results for the example Reactor B and C CRWE calculations rounded up to two places are shown in Table 3.16.

Based on the results shown in Table 3.16, the recommended MCPR values for RBM bypass for the example reactors (1.09 SLMCPR) are as follows:

The assumed single loop SLMCPR is 1.11 and greater than 90% power is not attainable with single loop operation. For SLMCPR values different than 1.09 (two loop) and 1.11 (single loop), the MCPR values for RBM bypass need to be multiplied by the ratio of the SLMCPR values (SLMCPR/1.09).

4.1.5 Cells with Fewer LPRM Strings

Selected cases were evaluated to verify that the CRWE MCPR results for control rods surrounded by four RBM LPRM strings are applicable to control rods that are surrounded by fewer than four RBM LPRM strings. The control rods with less than four LPRM strings in an RBM channel are located near the core periphery where the missing string is located away from the control rod position. A core map for Reactor B showing the typical assignment of LPRM detector strings to the RBM system is shown in Appendix A on Figure A.2.

To evaluate the effect of the number of LPRM strings on the CRWE MCPR results, representative MCPR results were tabulated for one through four LPRM strings input to a RBM channel. The resulting mean and standard deviation ΔMCPR/initial MCPR results are shown in Table 3.17. All MCPR results in the tables are for an RBM setpoint of 108%. The Case 1 and 2 results for three LPRM strings in Table 3.17 are for different geometries (the Case 1 three LPRM assembly geometry is shown on Figure A.2).

As shown in Table 3.17, the ΔMCPR /initial MCPR results are about the same for the one through four LPRM strings. These results show that the CRWE MCPR results for four LPRM strings are valid for a rod withdrawal near the core periphery with fewer than four LPRM strings input to the RBM system.

4.2 *LHGR Results*

The limiting LHGR values for the CRWE event in the two example reactors were calculated with the MICROBURN-B2 or MICROBURN-B codes and an evaluation was performed to confirm that the transient LHGR limits were not exceeded. None of the cases evaluated in these sample cases exceeded the transient LHGR limit during the postulated event. For most events the control rod block occurs while the control rod is still deeply inserted and the corresponding LHGR is relatively low. The overall core power increase is less than 10% and the transient LHGR limit is typically 15% above the steady state LHGR limit. The most limiting cases for LHGR concerns are the unblocked cases. For the Reactor B and C cases, the maximum calculated unblocked CRWE LHGR results are shown in Table 3.18. In all cases the LHGR did not exceed the transient LHGR limit.

Table 4.1 CRWE MCPR Results for Reactor B ATRIUM-10 Fuel at 100% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.2 CRWE MCPR Results for Reactor B ATRIUM-10 Fuel at 65% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.3 CRWE MCPR Results for Reactor B ATRIUM-10 Fuel at 40% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.4 CRWE MCPR Results for Reactor C ATRIUM-10 Fuel at 100% Power Conditions - SLMCPR 1.09

Table 4.5 CRWE MCPR Results for Reactor C ATRIUM-10 Fuel at 64% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.6 CRWE MCPR Results for Reactor C ATRIUM-10 Fuel at 40% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.7 CRWE MCPR Results for Reactor B GE Fuel at 100% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.8 CRWE MCPR Results for Reactor B GE Fuel at 65% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.9 CRWE MCPR Results for Reactor B GE Fuel at 40% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.10 CRWE MCPR Results for Reactor C GE Fuel at 100% Power Conditions - SLMCPR 1.09

Table 4.11 CRWE MCPR Results for Reactor C GE Fuel at 64% Power Conditions - SLMCPR 1.09

* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.12 CRWE MCPR Results for Reactor C GE Fuel at 40% Power Conditions - SLMCPR 1.09

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* Mean ΔMCPR/initial MCPR.

† Standard deviation.

‡ 95/95 ΔMCPR/initial MCPR.

Table 4.13 Representative ARTS RBM Instrumentation Setpoints

Power Setpoint	Analytical Trip Level Setting*
LPSP	30.0
IPSP	65.0
HPSP	85.0

Analytical RBM Trip Setpoints (Unfiltered)		
LTSP	ITSP	HTSP
118	112	108
121	116	111
124	119	114
127	122	117

Function	Definition
LPSP	Low power setpoint; rod block monitor system trips automatically bypassed below this level
IPSP	Intermediate power setpoint
HPSP	High power setpoint
LTSP	Low trip setpoint
ITSP	Intermediate trip setpoint
HTSP	High trip setpoint

* Analytical setpoint in % of reference power level.

Table 4.14 Maximum CRWE MCPR Operating Limit Results for Reactor B with 1.09 SLMCPR

Table 4.15 Maximum CRWE MCPR Operating Limit Results for Reactor C with 1.09 SLMCPR

* Bounding results for ATRIUM-10 and GE fuel in Reactor B.
† Analytical unfiltered RBM setpoints (see page 4-1 for filter effects).
‡ Bounding results for ATRIUM-10 and GE fuel in Reactor C.

Table 4.16 Maximum Unblocked CRWE MCPR Results for Reactors B and C with 1.09 SLMCPR

Table 4.17 CRWE Analysis Results for Peripheral Rod Groups (108% RBM Setpoint)

Table 4.18 Maximum CRWE LHGR Results

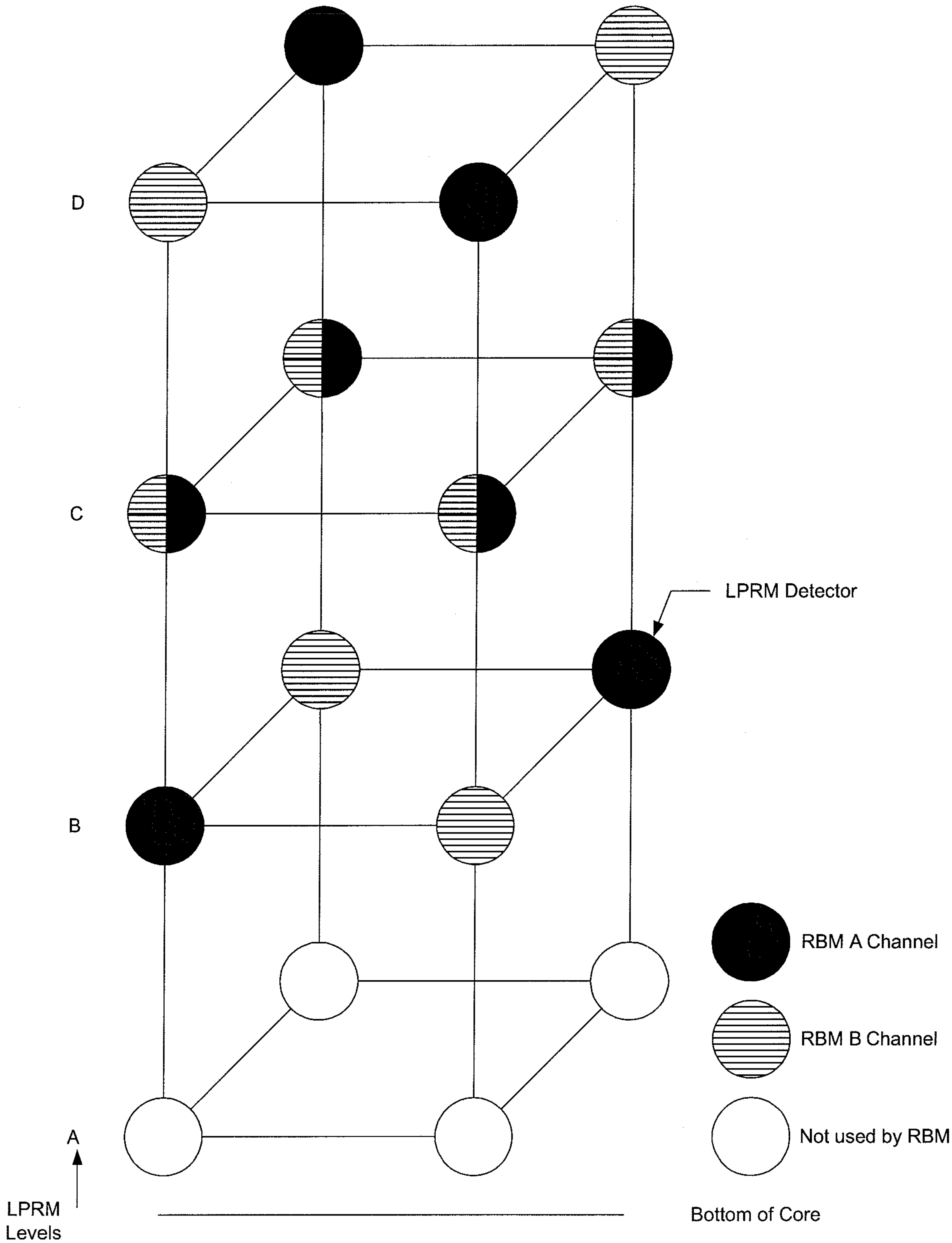

Figure 4-1 Rod Block Monitor LPRM Assignments for ARTS

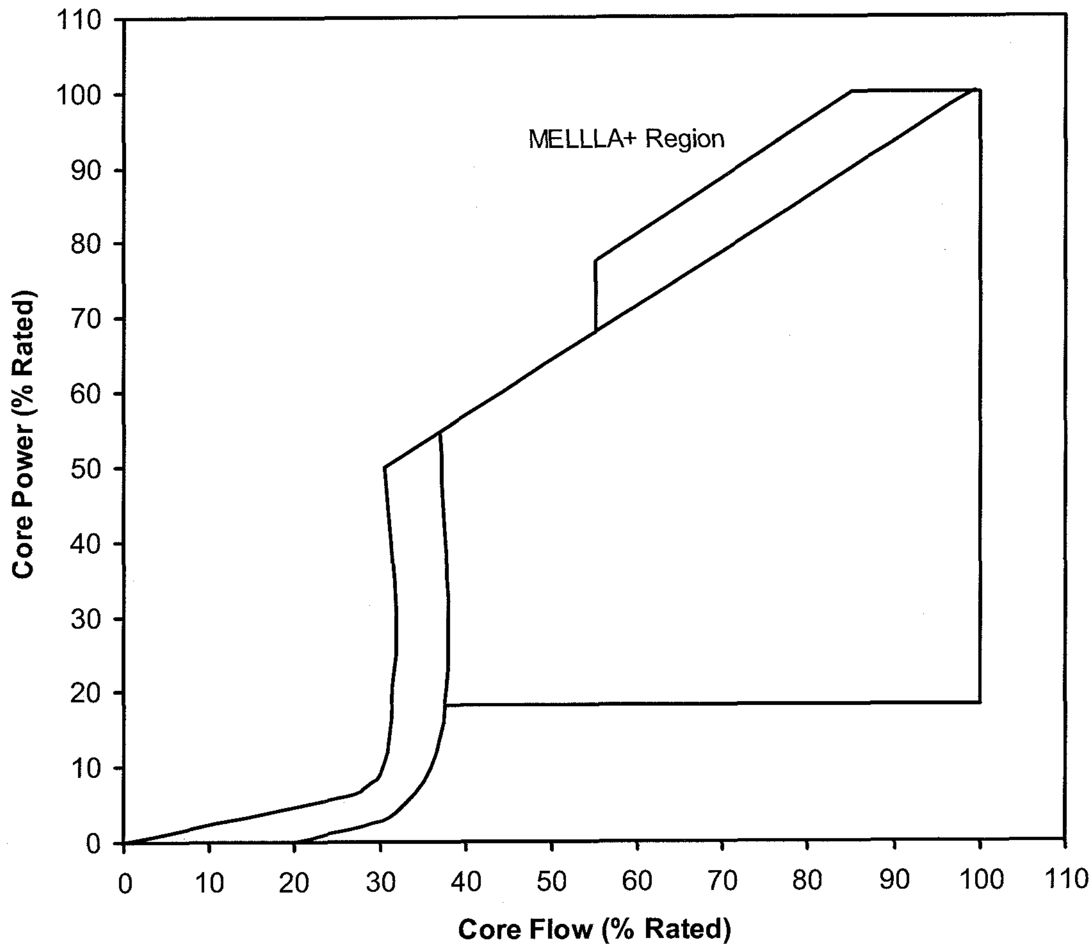

Figure 4-2 Reactor B Power/Flow Map for MELLLA+ Operation

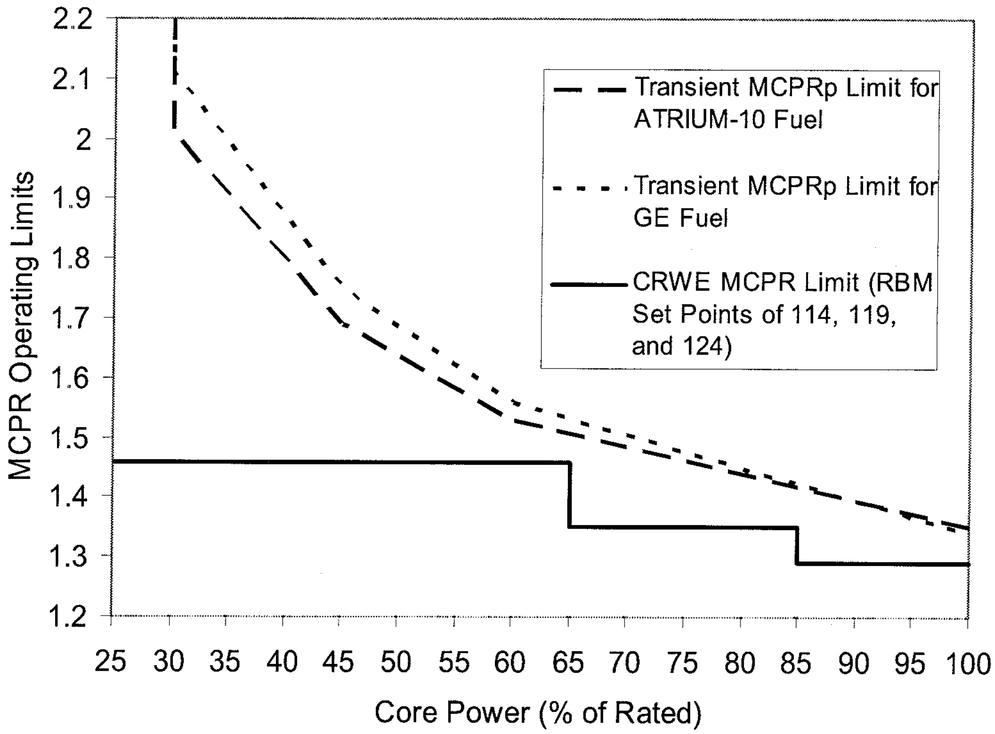

Figure 4-3 Representative MCPR$_p$ and CRWE MCPR Limits for Reactor B

5.0 Effect of RBM Signal Filter on CRWE

Optional capability is included to filter the ARTS RBM signal to reduce signal noise levels. The filter time constant is adjustable up to a maximum value of 0.5 ± 0.05 seconds. The design of the control rod drive system is for a normal speed of 3 ± 0.6 inches/second. When the filter is utilized, the filtered signal lags the unfiltered signal. For a ramp input, the asymptotic time lag will equal the time constant of the filter.

AREVA has evaluated the effect of the signal filter on the ARTS RBM setpoints. The RBM response data were evaluated with and without a time constant filter. The data base included results at all analyzed power levels. The evaluation was performed for a maximum filter time constant of 0.55 seconds and a maximum control rod withdrawal speed of 3.6 inches/second. The maximum control rod withdrawal speed results in the maximum control rod withdrawal distance as a function of RBM setpoint with filter. The difference between the filtered and unfiltered setpoints is subtracted from the analytical setpoint values to assure that the CRWE results are valid. The evaluated setpoint reduction results are summarized statistically in Table 4.1. For each evaluated condition, the uncertainty on the setpoint result is small.

The AREVA analysis supports the following setpoint reduction values for filter lag effects:

- For RBM setpoints ≤ 108%, subtract 0.6%.
- For 108% < RBM Setpoints ≤ 116%, subtract 0.8%.
- For 116% < RBM Setpoints ≤ 124%, subtract 1.0%.
- For 124% < RBM Setpoints ≤ 127%, subtract 1.2%.

These setpoints are "Analytical Limits;" other adjustments are required for inaccuracy, calibration, and drift effects to obtain the "Nominal Trip Setpoint." The RBM setpoint reductions for filter time lag effects are independent of the fuel type.

Table 5.1 RBM Signal Filter Setpoint Adjustment

Power Level (%)	RBM Channel	Number of times evaluated	RBM Setpoint (%)	Mean Signal Difference Where Unfiltered Signal equals Setpoint	Standard Deviation of Difference
100	1	1056	108	0.0048	0.0009
100	2	1056	108	0.0052	0.0011
40	1	767	118	0.0081	0.0023
40	2	766	118	0.0078	0.0022

6.0 Range of Applicability

The range of applicability of the CRWE statistical methodology and results in this report are summarized below:

6.1 *Methodology*

The following are the requirements for the statistical CRWE method:

- The statistical methodology presented in Section 3.0 for evaluation of the CRWE event is applicable for BWR/3, 4, and 5 plants that have installed the ARTS RBM system.
- The method is applicable to CRWE analyses performed with NRC approved reactor physics codes including the AREVA MICROBURN-B and MICROBURN-B2 codes (References 2 and 3).
- The method is applicable for CPR calculations performed with NRC approved critical power correlations including the AREVA SPCB and ANFB correlations (References 4 and 5) and future NRC-approved CPR correlations.
- Control rod withdrawal rates must be slow enough to approximate with the steady state reactor physics code. Rod withdrawal rates up to 4 inches per second are reasonably estimated with steady state reactor conditions.
- The power level increase for the control rod withdrawal event is dominated by the control rod worth. The reactivity increase during the time of the withdrawal may impact the steady state assumption. Total power increases for the event are limited to 10% of rated core thermal power.

6.2 *CRWE Results for Example Reactors*

The CRWE OLMCPR and LHGR results in this report are applicable for the following conditions in the two analyzed reactors:

- Maximum cycle length of 24 months
- All AREVA ATRIUM™-10 fueled cores
- ATRIUM-10 plus exposed GE fueled cores
- Reactor Power up to 20% EPU
- 85% Minimum Reactor Flow at 100% EPU power (MELLLA+)
- 75% Minimum Reactor Flow at 100% power (MELLLA)

7.0 References

1. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, *Exxon Nuclear Methodology For Boiling Water Reactors – Neutronic Methods For Design and Analysis,* Exxon Nuclear Company, March 1983.

2. EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, Siemens Power Corporation, October, 1999.

3. XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, *Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology,* Advanced Nuclear Fuels Corporation, November 1990.

4. EMF-2209(P)(A) Revision 2, *SPCB Critical Power Correlation*, Framatome ANP, September 2003.

5. EMF-1125(P)(A) Supplement 1 Appendix C, *ANFB Critical Power Correlation Application for Co-Resident Fuel*, Siemens Power Corporation, August 1997.

6. XN-NF-825(P)(A), *BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$,* Exxon Nuclear Company, May 1986.

7. XN-NF-825(P)(A) Supplement 2, *BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain,* Exxon Nuclear Company, October 1986.

8. Paul N. Somerville, "Tables for Obtaining Non-Parametric Tolerance Limits," Annals of Mathematical Statistics, Volume 29, No. 2, pages 599-601, June 1958.

Appendix A Reactor B LPRM Core Layout

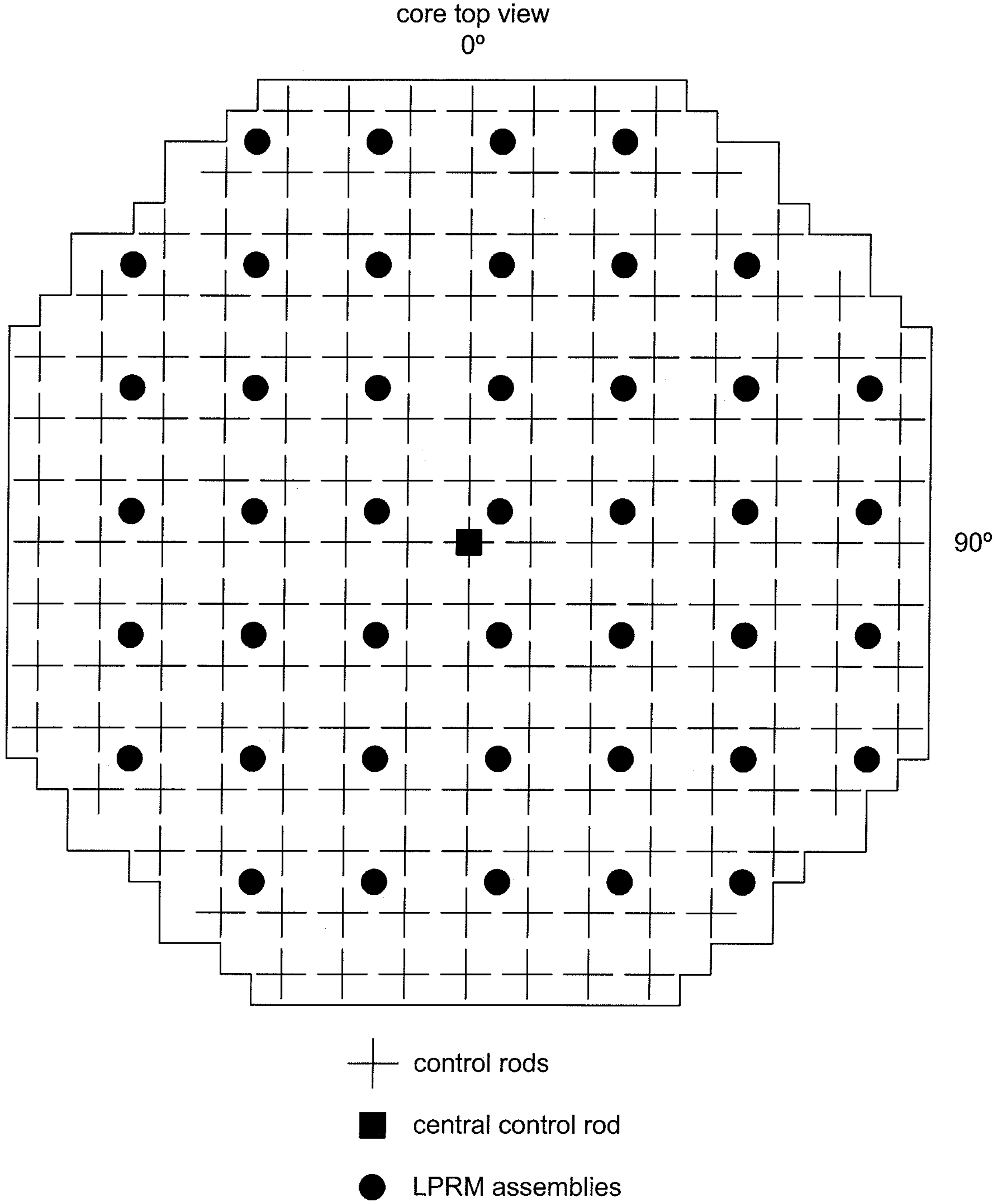

Figure A.1 Reactor B LPRM and Control Rod Locations

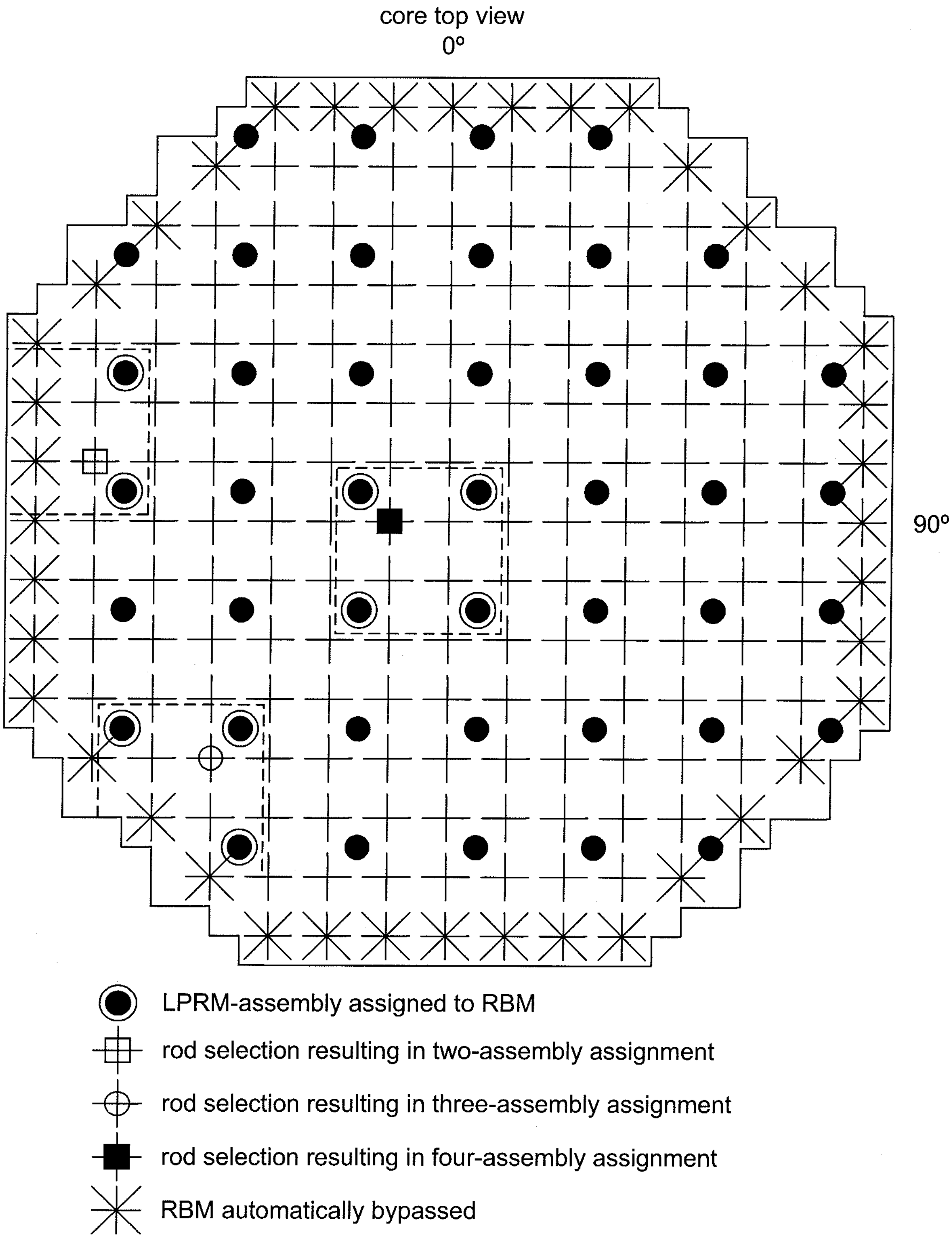

Figure A.2 Reactor B Assignment of LPRM Assemblies to RBM's