General Electric Advanced Technology Manual

Chapter 4.3

Power Oscillations

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4.3   POWER OSCILLATIONS

Learning Objectives:

1. List the primary safety concern regarding unstable power oscillation.

2. List the major factors that can contribute to instability.

3. Explain the mechanism present at most BWR plants to guard against neutron flux oscillations.

4. Define the following terms:
    a. fuel time constant
    b. decay ratio

5. Explain why it is difficult to detect power oscillations.

4.3.1   Introduction

Boiling water reactors (BWRs) have complex dynamic responses that can result in the initiation of power oscillations.   Of the various types of oscillations, those generated from control systems response are the most common.   Controllers, such as the master recirculation flow controller, are typically more stable at the high end of their control band than at the low end.   To account for this problem, interlocks and procedures prevent automatic master flow control below some value (typically less than 45%).   Other control systems that effect BWR oscillations are the pressure control system and the feedwater control system.   Even with the constant modulation of the turbine control valves to regulate reactor pressure and feedwater pump steam supply valves or feedwater regulating valves to control feedwater flow, a sinusoidal oscillation can be observed in reactor power during steady state operation.   These oscillations are usually slow and small in magnitude.   Figure 4.3-1 was taken from an operating recorder in a BWR control room and illustrates the power oscillations that occur at many plants during normal power operation.   The amplitude of these observed oscillations has ranged from a few percent to fifteen percent. Oscillations that occur from control system responses are not normally divergent and do not challenge fuel safety limits.

Unstable power oscillations can occur during power operations or in conjunction with an Anticipated Transient Without Scram (ATWS).   The primary safety concern regarding unstable power oscillations during normal operations is the ability of the reactor protection system to detect and suppress oscillations before they can challenge the fuel safety limits (Minimum Critical Power Ratio).   The safety issues regarding power instability in conjunction with ATWS events are now being resolved within NRR.

The type of instability that can lead to divergent oscillations and challenge fuel safety limits is a thermal-hydraulic, neutronic generated, density-wave instability that occurs inside fuel bundles.   GE BWR plant and fuel design provide stable operation with margin within the normal operating domain.   However, at the high power/low flow corner of the power/flow operating map, the possibility of power oscillations exists.   The major factors that can contribute to instability are void fraction, fuel time constant, power level, power shape, feedwater temperature and core flow.   To provide assurance that the oscillations are detected and suppressed, technical specifications require that APRM and LPRM flux levels be monitored when in the region of possible power oscillation.   This requirement is based on the results of stability tests at operating BWRs.   A conservative decay ratio of 0.6 was chosen as the basis for determining the generic region for monitoring for power oscillation.   Decay ratio in this context is the measured stability of an oscillating system and is the quotient of the amplitude of one peak in an oscillation divided by the amplitude of the peak immediately preceding it.   The amplitude is measured relative to the average amplitude of the signal.   A stable system is characterized by a decay ratio of less than 1.0.   As a result of recent power oscillation events, and a desire to minimize the possibility of exceeding the minimum critical power ratio (MCPR) safety limit, the BWROG and the NRC have agreed in principle to three plausible options that are discussed in Section 4.3.3 on Mitigation of Power Instability.

Thermal-hydraulic-neutronic instabilities in BWRs have been known to exist since the early days of BWR research using prototype reactors.   Although this instability mechanism was identified early, the analysis methods needed to predict its effect are only now becoming available.   Appendix 1, Analysis Methods Used For BWR Stability Calculations, is, therefore, provided for additional information.

4.3.2 Discussion of Power Instability

The basic mechanism causing flow and power instabilities in BWRs is the density wave. The effect of a density wave is illustrated in Figure 4.3-2.   Coolant flows in the upward direction through the core and is guided by the channels that surround the matrix of fuel rods.   Local voiding within a fuel bundle may be increased either by an increase in the power at a constant inlet flow, by a decrease in the inlet flow at constant power, or by an increase in feedwater temperature.   This resulting localized concentration of voids will travel upward, forming a propagating density wave which produces a change in the localized pressure drop at each axial location as it travels upward.   The effective time for the voids to move upward through the core is referred to as the density wave propagation time.   In two-phase flow regimes, the localized pressure drop is very sensitive to the local void fraction, becoming very large at the outlet of the bundle where the void fraction is normally the greatest.   Because of this a significant part of the pressure drop is delayed in time relative to the original flow perturbation.

If a sine wave perturbation of the inlet flow is used to illustrate this, Figure 4.3-3 is obtained.   The localized axial pressure drops are also sinusoidal within the linear range; however, they are delayed in time with respect to the initial perturbation, the sine wave in this case.   The total pressure drop across the bundle is the sum of the localized pressure drops.   If the bundle outlet pressure drop (the most delayed with respect to the initial perturbation) is larger than the inlet pressure drop, then the total bundle pressure drop may be delayed by as much as 180 degrees with respect to the inlet flow perturbation and be of the opposite sign.   This is the case in Figure 4.3-3, where an increase in inlet flow results in a decrease in the total bundle pressure drop.   Bundle flow with this density wave propagation time behaves as if it has a "negative" friction loss term.   This causes the bundle flow to be unstable, inlet flow perturbations to reinforce themselves (positive feedback), and oscillations grow at the same unstable frequency.   Bundle flow instability starts when the outlet (i.e., delayed) localized pressure drop equals the pressure drop at the inlet for a particular density wave propagation time.

Power generation is a function of the reactivity feedback and, depends strongly on the core average void fraction.   When a void fraction oscillation is established in a BWR, power oscillates according to the neutronic feedback and the core dynamics.   Most important to this discussion are the void fraction response to changes in heat flux, including the inlet flow feedback via the recirculation loop, and the reactivity feedback dynamics.

One important difference between the neutronic feedback dynamics and the flow feedback dynamics is the fuel time constant.   Before the power generated in the fuel can affect the moderator density, it must change the fuel temperature and transfer heat to the coolant.   The fuel in BWRs responds relatively slowly with a time constant between 6 and 10 seconds.   The delay times for unstable density wave oscillation and void reactivity feedback are not the same.   Differences in the delay times add additional phase delays and can cause the void feedback to reinforce the density wave oscillations (effectively positive feedback).   Decreasing the time response of the fuel generally has a destabilizing effect.   Smaller response times can be a problem even if only a small portion of the fuel has the decreased time response, as was the case in the WNP2 event, because the most unstable bundles dominate the response.

When conditions within a reactor are such that it could become unstable (eg: high power and low flow), any perturbation in the inlet conditions can start the unstable oscillations. A moment before the instability event starts, the reactor is in a relatively steady condition with some particular power and flow.   Initially the reactor will behave linearly and the oscillations will grow exponentially.   As the oscillation becomes larger, the nonlinearities in the system begin to grow in importance.   These nonlinearities have the effect of increasing the negative power feedback in the reactor.   When a sufficiently large reactivity bias is reached, an equilibrium is established, and a limit cycle oscillation

remains.   The amplitude of the resulting limit cycle oscillation will depend on various parameters and can be many times greater than rated full power.

BWRs can experience unstable power oscillation either in a single bundle (localized) or core wide.   In the case of core wide oscillations, the entire core can oscillate together or part of the core can be increasing in power while another part is decreasing in power (out of phase).   The out of phase oscillation is important because it is more difficult to detect. BWRs monitor local power at various radial and axial locations with the use of Local Power Range Monitors (LPRMs).   The LPRMs consist of up to 172 stationary in-core detectors which are arranged in radially located assemblies of four detectors each, separated at axial intervals of three feet.   The LPRMs in turn provide information to the Average Power Range Monitoring (APRM) System.   In general, a set of individual LPRMs provide information to a single APRM channel.   APRMs sample power both radially and axially in the core, and, therefore, may not indicate the worst case out-of-phase oscillation since the oscillation may be masked by the cancellation between out of phase LPRMs that provide signals to the same APRM channel.

Bottom-peaked power shapes are more unstable because they tend to increase the axially averaged void fraction.   This causes void perturbation to start at a lower axial level, and produces a longer delay time for the density wave which will be more unstable. Radial power shape is important because the most unstable bundles tend to dominate the overall response.   Lower void velocities result in longer delay times for the density wave which will be more unstable.   Increasing the subcooling of the feedwater inlet flow has two effects.   First, it will tend to increase the operating power (a destabilizing effect) and second, it raises the boiling boundary (a stabilizing effect).   In most cases the total effect is destabilizing.   The fuel isotopic composition has an indirect effect on the density reactivity coefficient with the effect depending on the burnup and being difficult to predict. Generally increased burnup causes the density reactive coefficient to become less negative, which will tend to destabilize the core.

Many of these effects can accrue as a result of a single cause.   As an example, fuel burnup will change the fuel isotopic composition as well as the axial power shape. Additionally, changes in other parameters can effect these factors.   Increasing reactor pressure will decrease the core average void fraction and stabilize the reactor. Increasing the core inlet restriction (flow orificing) will increase the single phase component of the pressure drop across the core which retards dynamic increases in the flow rate (a stabilizing effect).   Therefore, the effects of all parameters must be taken into account when evaluating mitigation strategies.

4.3.3 Mitigation of Power Instability

General Design Criteria (GDC) 10, 12, and 20 of 10 CFR 50, Appendix. A, require that protection systems be designed to assure that specified acceptable fuel design limits are not exceeded as a result of power oscillations that are caused by thermal-hydraulic instabilities.   Minimum Critical Power Ratio (MCPR) is the primary fuel design limit that is being protected during potential instabilities.

The BWROG submitted to the U.S. Nuclear Regulatory Commission Topical Report NEDO31960, "Long-Term Stability Solutions Licensing Methodology," (Reference 7) for staff review.   Long-term solutions described in this report consist of conceptual designs for automatic protection systems developed by the BWROG with its contractor, the General Electric Company.   The automatic protection systems are designed to either prevent stability related neutron flux oscillations or detect and suppress them if they occur.   This report also described methodologies that have been developed to establish set points and demonstrate the adequacy of the protection systems to prevent violation of Minimum Critical Power Ratio limits in compliance with 10CFR50, Appendix A, GDC 10 and 12.

Because of the variety of plant types, the need to accommodate differing operational philosophies, and owner-specific concerns, several alternative solutions are being pursued.   For some BWR/2s, existing systems and plant features already provide sufficient detection and suppression of reactor instabilities.   This capability is limited primarily to those plants having quadrant average power range monitors (APRMs), it is referred to as Option II, and has been agreed upon by BWROG and the NRC. However, for most of the BWRs, new or modified plant systems may be necessary.   A summary of the three most promising BWR owner group long-term solutions is provided below.

4.3.3.1 Solution Description Option I-A

Regional Exclusion, Option I-A, assures compliance with GDC-12 by preventing the occurrence of instability.   This is accomplished by preventing entry into a power/flow region where instability might occur.   An example of an exclusion region (I) is shown in Figure 4.3-4 along with the restricted (II) and monitored (III) regions.   Upon entry into the exclusion region, an Automatic-Safety-Feature (ASF) function will cause the region to be exited.   The ASF may be a full scram or a selected rod insert (SRI).   For plants choosing SRI as their primary ASF, a full scram automatic backup must take place if the exclusion region is not exited within a reasonable period of time (a few seconds).

For plants choosing to implement this option (full scram or SRI), the existing flow-biased scram cards will be replaced.   The new microprocessor-based cards will provide three independent functions: (1) a scram signal (that will be processed by the existing

flow-biased scram system) if the exclusion region is entered, and (2) an alarm (directed to an existing alarm panel) if the restricted region is entered, and (3) automatic monitoring (using the period-based algorithm of solution III) within the monitored region to detect instabilities should they occur.

Entry into the monitored region is unrestricted.   This region only defines a region outside which the monitoring algorithm is not active.   The main purpose is to avoid false alarms from the automated monitor when operating at very low powers during startup. Intentional entry into the restricted region is only permitted if certain stability controls are in place.   These stability controls deal primarily with power distributions and may be implemented by monitoring a parameter defined as the boiling boundary.   The purpose of these controls is to assure that plant conditions that are sensitive to stability are bounded by the assumptions of the exclusion region boundary analysis.

4.3.3.2 Solution Description Option I-D

Regional Exclusion with Flow-Biased APRM Neutron Flux Scram, Option I-D, assures that BWRs with tight fuel inlet orificing (less than 2.43 inches) and an unfiltered, flow-biased scram comply with GDC-12 by providing an administrative boundary for normal operations in the vicinity of the region where instability could be expected to occur. During normal operation, the boundary of the exclusion region is administratively controlled, and operation within the region is to be avoided.   If an unexpected operational event results in entry into the exclusion region, action to exit the region must be taken immediately.   Oscillations that do occur in this situation should be automatically detected and eliminated by the flow-biased APRM neutron flux scram.   This scram is based on a comparison of the unfiltered APRM signal to a set point that varies as a function of core flow.   When the unfiltered APRM neutron flux signal exceeds the flow-biased set point, a scram signal is generated.   An example of the administratively controlled region and the instability region is shown in Figure 4.3-5.

4.3.3.3 Solution Description Option III

Local Power Range Monitor (LPRM) based Oscillation Power Range Monitor (OPRM), Option III, is a microprocessor-based monitoring and protection system that detects a thermal hydraulic instability and initiates an alarm and ASF before safety limits are exceeded.   The OPRM does not affect the design bases for the existing APRMs because it operates in parallel with and is independent of the installed APRM channels.

Each of the four OPRM channels consist of a microprocessor unit that takes amplified LPRM signals from available locations in the APRM cabinets.   These LPRM signals are grouped together such that the resulting OPRM response provides adequate coverage of expected oscillation modes.   If a thermal-hydraulic oscillation is detected, a scram signal will be sent to the ASF designed to suppress oscillations before the MCPR safety limit is

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exceeded.   The OPRM would be installed and maintained as a reactor protection system (RPS) parameter input in the standard one-out-of-two-twice actuation scram logic for all BWRs (except Clinton which has a solid-state logic RPS).

The algorithms proposed for use in the automatic detection solutions, I-D and III are: High-Low-High Algorithm, Growth Algorithm and Period-Based Algorithm.   The High-Low-High Algorithm establishes a setpoint at some value above 100% power.   In order to cause a scram the signal must pass through the setpoint with a positive slope followed by passing through the setpoint with a negative slope and then pass the setpoint a second time with a positive slope.   When the setpoint is set well above the random fluctuations that occur in reactor operation, this algorithm will prevent scrams that would otherwise result from single spikes.   The Growth Algorithm is designed to detect the presence of oscillations as they grow above the level of normal random noise.   If the amplitude of an oscillation is greater than the previous oscillations amplitude by a predetermined amount, a scram signal will be generated.   The Period-Based Algorithm is the most sensitive of the automatic detection solution algorithms.   It detects the "periodicity" of the signal by maintaining statistical data of the intervals between consecutive peaks.   When the "periodicity" is high, the reactor is considered to be approaching instability.

Although not part of the BWROG proposed long term solutions, several "Decay Ratio" monitor designs have been developed and used.   These on-line monitors can show operators how close the plant is to being unstable and have the same general principles of operation.   They use the random fluctuations in the neutron population (reactor noise) to determine the current reactor decay ratio at any given time.   The algorithm that is used (determination of the effective decay ratio by using the automatic correlation of the signal) must be time averaged to reduce the fluctuation inherent in this method and to increase its accuracy. Although these are on-line systems, the signal from the monitors is delayed by the averaging time (usually about 2 minutes).   The Advanced Neutron Noise Analysis (ANNA) system by Siemens is used at WNP-2 and is the only such system used in the U.S.   At the present, the monitor at WNP-2 is only used for startup operations.   The NRC has granted WNP-2 permission, through a technical specification change, to operate in the old exclusion region C provided the decay ratio monitoring system (ANNA) is in operation.   The system was not in use during the oscillation events that occurred at WNP-2.   The CASMO system by ABB-Atom and the SIMON system by EuroSim are in use at some foreign BWRs.   In Sweden, decay ratio monitors are used at all times since the plants operate in a load following mode and routinely drop flow very close to the exclusion region.   Reports indicate that the use of these monitors has prevented many reactor scrams and oscillation events.   However, due to their high sensitivity, false alarms are not unusual, and the monitors may indicate high decay ratios when stable conditions exist.

4.3.4 Historical Perspective

Evaluation of the probability of thermal-hydraulic instability in BWRs has been an ongoing study by General Electric starting with the first power production plants.   Early testing consisted of moving a control rod one notch position while monitoring reactor performance.   For BWR/3s, 4s, 5s, and 6s with high power density cores, a pressure disturbance technique was used to cause power instability.   The pressure disturbance was accomplished using one of the four turbine control valves.   The signal used to control the perturbation amplitude was adjusted to obtain an APRM neutron oscillation within 15% of the steady state signal.

Tests following the instability scrams (one each in 1982 and 1983) at the Caorso Nuclear Power Station (Italian plant), indicated the possibility of power oscillation at high power and low flow conditions.   These tests also indicated an out-of-phase neutron flux oscillation and showed that half of the core was oscillating 180   out of phase with respect to the flux oscillation in the other half of the core (as sensed by the LPRMs).   These tests also showed that APRMs would not be as sensitive to such a phenomenon.   While the LPRMs indicated oscillations of 60% of peak-to-peak power, APRMs indicated oscillations of only 12%.

On February 10, 1984, General Electric issued Service Information Letter (SIL) 380, Revision 1, which discussed the BWR core thermal-hydraulic stability problems that could exist in different variations in all BWRs.   The SIL provided a list of recommended actions and identified the high power, low flow corner of the power-to-flow map as the region of least stability and one which should be avoided.   If this region of instability was entered, guidance was to insert control rods to reduce reactor power below the 80 percent rod pattern line and monitor LPRMs and APRMs for oscillation.

Generic Letter 86-02 was issued January 1986 to inform licensees of the acceptance criteria for thermal-hydraulic stability margin required in GDC 10 and GDC 12.   The objective of the letter was to account for these criteria in future licensing submittals and in safety evaluations in support of 10 CFR 50.59 determinations.   It also stated that plants may have to change technical specifications to comply with SIL 380, Rev. 1.

On March 9, 1988 the Unit 2 reactor at the LaSalle Station was operating at 84% steady state power and 76% flow when an instrument technician made a valve lineup error that caused both recirculation pumps to trip.   As a result of the rapid power decrease, the EHC system reduced steam flow to the main turbine causing a reduction of extraction steam.   The rapid decrease in extraction steam caused severe perturbations in feedwater heater levels which eventually caused isolation of the heater strings. Feedwater temperature decreased 45 F in 4 minutes as a result of this significant reduction in feedwater heating, causing an increased power-to-flow ratio and further reducing the margin to instability.   Between 4 and 5 minutes into the event, the APRMs

were observed to be oscillating between 25 and 50% power every 2 to 3 seconds accompanied by oscillating LPRM up scale and down scale alarms. The unit automatically scrammed at the 7 minute mark from a fixed APRM scram signal of 118%.

On December 30, 1988 NRC Bulletin 88-07, Supplement 1, dealing with power oscillations in BWRs was issued. The purpose of this supplement was to provide additional information concerning power oscillations in BWRs and to request that licensees take actions to ensure that the safety limit for minimum critical power ratio (MCPR) was not exceeded. In addition, within 30 days of receipt of Supplement 1, all BWRs were required to implement the GE interim stability recommendations derived for GE fuel. The supplement also specified that plants with ineffective automatic scram protection shall manually scram the reactor if both recirculation pumps should trip. Adequate automatic scram protection is available at plants with a flow biased APRM scram with no time delay. Inadequate automatic scram protection is provided at plants with a fixed APRM high flux scram and a separate thermal APRM, time delayed, flow-biased scram.

During the startup of cycle 13, of the Ringhals-I plant in Sweden in 1989, an unexpected out of-phase oscillation occurred with a peak-to-peak amplitude of about 16 percent. The event was initiated when high neutron flux power level triggered an automatic pump run back from 79 percent power to 68 percent power. An analysis following the event appeared to indicate that the slope of the flow control line was altered by the new fuel cycle and that an increase in recirculation flow resulted in greater-than-expected increases in power.

The Caorso nuclear power station (a BWR/6 located in Italy) experienced an unexpected instability event in 1991. The event occurred during a reactor startup, using GE-7 fuel, and with plant conditions of minimum pump speed, minimum flow control valve position, and a rod pattern line of nearly 80 percent. Actual power and flow values were uncertain but were estimated to be in the range from 38 to 40.8 percent power and from 30.7 to 31.3 percent flow. This event demonstrated that oscillations below the 80 percent rod line are possible and suggested that the regions defined in NRC Bulletin 88-07 may not have been restrictive enough. This event occurred during a startup and was attributed to extreme bottom-peaking of the axial power shape. The feedwater heaters were still cold when the event occurred with a feedwater temperature of approximately 150oF and 56 BTU/lb of subcooling. An interesting effect occurred during the event. The power oscillations continued to grow in amplitude while core power was clearly decreasing as the operator inserted the control rods. The corrective action to avoid repetition of this event was to modify the plant startup procedures to require a hot feedwater temperature before power could be increased above 30 percent power.

On August 15, 1992, Washington Nuclear Power Unit-2 experienced power oscillations during startup. The reactor core for cycle 8 consisted of mostly Siemens fuel (9*9-9x)

that has a higher flow resistance than the GE 8*8 fuel.   While on the 76% rod line following a power reduction with flow, a power oscillation was observed by the operators who then initiated a scram.   An Augmented Inspection Team (AIT) found, by analyses using LAPUR code, that a major contributor was the core loading.   The analyses indicated that a full core load of 9*9-9x fuel would be less stable than the old 8*8 fuel and that the mixed core was less stable than a fully loaded core of either type.   This event indicated that the boundaries of the instability region defined in the BWROG interim corrective actions may not include all possible areas of instabilities.

4.3.5 Analysis Methods Used For BWR Stability Calculations

Predictive calculations of BWR stability are too complex to allow for simple calculations and require computer codes to simulate the dynamic behavior of the reactor core.   The family of codes that has been used to represent and to predict the stability of commercial BWRs can be subdivided in two main categories: frequency-domain and time-domain codes.   Among the frequency domain codes are LAPUR, NUFREQ, and FABLE. Time-domain codes are more widely used and include RAMONA-3B, TRAC-BF1, TRAC-G, RETRAN, EPA, SABRE, TRAB, TOSDYN-2, STANDY, and SPDA.

LAPUR was developed at the Oak Ridge National Laboratory (ORNL) for the NRC and is currently used by NRC, ORNL, and others.   LAPUR's capabilities include both point kinetics and the first subcritical mode of the neutronics for out of phase oscillations.   The thermalhydraulic part is modeled to consider up to seven flow channels with inlet flows coupled dynamically at the upper and lower plena to satisfy the pressure drop boundary condition imposed by the recirculation loop.   LAPUR's main result is the open- and closed-loop reactivity-to-power transfer function from which a decay ratio is estimated. Its current version is LAPUR-5.

NUFREQ is a set of codes called NUFREQ-N, NUFREQ-NP, and NUFREQ-NPW that calculate reactor transfer functions for the fundamental oscillation mode. The main difference between them is their ability to model pressure as an independent variable (NUFREQ-NP) so that the pressure perturbation tests can be reproduced. NUFREQ-NPW is a proprietary version currently used by Asea Brown Boveri (ABB); its main feature is an improved fuel model that allows modeling of mixed cores.

FABLE is a proprietary code used by General Electric (GE) which can model up to 24 radial thermal-hydraulic regions that are coupled to point kinetics to estimate the reactor transfer function for the fundamental mode of oscillation.

RAMONA is a code that was developed by ScandPower; it is currently used by Brookhaven National laboratory (BNL), ScandPower, and ABB.   The RAMONA-3B version was developed by BNL and has a full three dimensional (3D) neutron kinetics model that is capable of coupling to the channel thermal-hydraulics in a one-to-one basis.

Typically, when using time-domain codes, the thermal-hydraulic solution requires orders of magnitude more computational time than the neutronics codes. Because of the large expense associated with the computational time, thermal-hydraulic channels are often averaged into regions to reduce computational time. RAMONA-3B uses an integral momentum solution that significantly reduces the computational time and allows for the use of as many computational channels as necessary to accurately represent the core.

TRAC has two versions currently used in BWR stability analysis. TRAC-BF1 is the open version used mostly by Idaho National Engineering Laboratory (INEL) and Pennsylvania State University, while TRAC-G is a GE-proprietary version. TRAC-BF1 has one dimensional neutron kinetics capabilities (as well as point kinetics). TRAC-G has full 3D neutron kinetics capability (as well as one dimensional and point kinetics), and GE has incorporated most of its proprietary correlations. The numerics in TRAC-G have also been improved with respect to those in TRAC-BF1 to reduce the impact of numerical diffusion and integration errors. Typically TRAC runs are very expensive in computational time; to minimize this time, most runs are limited to the minimum number of thermal-hydraulic regions that will do the job (typically 20).

RETRAN is a time-domain transient code developed by the Electric Power Research Institute (EPRI). It has one dimensional and point kinetics capability and is a relatively fast-running code since it models a single, radial, thermal-hydraulic region and uses the so-called three equation approximation (i.e, it assumes equilibrium between phases). A big advantage of RETRAN over other more detailed tools is that it is capable of running in a desk-top personal computer environment.

Engineering Plant Analyzer (EPA) is a combination of software and hardware that allows for real time simulation of BWR conditions including most of the balance of plant. It was developed for NRC and is located at BNL. EPA's software for BWR stability simulations (named HIPA) models point kinetics with mainly an average thermal-hydraulic region; a hot channel is also modeled but does not provide significant feedback to affect the global results. HIPA uses modeling methods similar to those of RAMONA-3B and, in particular, it uses the integral momentum approach to speed up the thermal-hydraulic calculations. An interesting feature of HIPA is its ability to use time dependent axial power shapes to compute the reactivity feedback. The nodal power shape is varied according to the local void fraction as a function of time based on some polynomial fits that are input to HIPA.

SABRE is a time domain code developed and used by Pennsylvania Power and Light for transient analyses that include BWR instabilities. SABRE uses point kinetics for the neutronics and a single thermal-hydraulic region.

TRAB is a one dimensional neutronics code with an average thermal-hydraulic region. It was developed and used in the Finish Center for Radiation and Nuclear Safety and has

been benchmarked against RAMONA-3B calculations and a stability event in the TVO-I plant.

TOSDYN-2 has been developed and used by Toshiba Corporation.   It includes a 3D neutron kinetics model coupled to a five-equation, thermal-hydraulic model and models multiple parallel channels as well as the balance of plant.

STANDY is a time domain code used by Hitachi Ltd.   It includes 3D neutron kinetics and parallel channel flow across at most 20 thermal-hydraulic regions.   STANDY is a vessel model only and does not include the balance of plant.

SPDA, a combination of RELAP5 and EUREKA, is used by the Japan Institute of Nuclear Safety.   RELAP5 calculates the thermal-hydraulic part of the solution, while the nodal power is estimated by EUREKA (which is a 3D neutron kinetics code).

4.3.6 References

1. NRC Bulletin 88-07 Supplement 1, Power Oscillations in Boiling Water Reactors, December 30, 1988.

2. NUREG/CR-6003 ORNL/TM-12130, Density Wave Instabilities in   Boiling Water Reactors, September 1992.

3. General Electric, BWR Core Thermal-hydraulic Stability, "SIL No. 380, Revision 1," February 1984.

4. Bernero, Robert M., Technical Resolution of Generic Issue B-19 - Thermal-hydraulic Stability, "NRC Generic Letter 86-02", 1986.

5. Briefing Report, "LaSalle Unit 2 Oscillation Event," March 1988.

6. May, M.J., "Stability Operating Procedures for Kernkraftwerk Leibstadt plant," October 1984.

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

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

9. Wulff, W., et al., BWR Stability Analysis with the BNL Engineering Plant Analyzer, "NUREG/CR 5816," November 1991.

10. ORNL, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO 31960 Supplement 1, July 1993.

11. W.P. Ang, et al., Special Inspection, NRC Inspection "Report No. 50-397/92-37," November 1992.

Table 4.3-1 SIL-380

General Electric recommends that BWR operators using GE BWR fuel monitor the inherent neutron flux signals   and avoid or control abnormal neutron flux oscillations (with particular attention to the region of sensitivity in where the probability of sustained neutron flux oscillations increase) as follows:

1. Become familiar and aware of your plants normal average power range monitor (APRM) and local power range monitor (LPRM) peak-to-peak neutron flux for all operating regions of the power/flow map and for all operating modes (e.g., two loop and single loop operation).   In particular, establish an expected APRM and LPRM peak-to-peak signal for your plant at various operating states and also for special operating modes (single loop operation) if these modes will be used.   The expected APRM noise amplitude can be easily determined from past steady state strip chart recordings or can be established based on current operating conditions.

2. Whenever making APRM or LPRM readings, verify that the neutron flux noise level is normal.   If there is any abnormal increase in the neutron flux response, follow the recommendation in Section 6d to suppress the abnormal noise signal.

3. The LPRM gains should be properly calibrated per current plant procedures.   This will permit the LPRM upscale alarm trip setpoints to be set as high as full scale while providing appropriate indication against unacceptable reduction in thermal margin because of power oscillations.   The LPRM upscale alarm indicators should be regularly monitored and all upscale alarms should be investigated to determine the cause and to assure that local limits are not being exceeded.

4. Whenever changes are made or happen that cause reactor power to change, monitor the power change on the APRMs and locally on the LPRMs to become familiar with the expected neutron flux signal characteristics.

5. If a recirculation pump(s) trip event results in operation in the crosshatched region of Figure 4.3-6:

   a. Immediately reduce power by inserting control rods to or below the 80% rod line using the plant's prescribed control rod shutdown insertion sequence.

   b. After inserting control rods, frequently monitor the APRMs and monitor the local regions of the core by using the control rod select switch to display the various LPRM strings which surround the selected control rod. A minimum of nine control rods should be selected to adequately display LPRMs representing each octant of the core and the core center Figure 4.3-7.   If there is any abnormal increase in the expected signals, insert additional control rods to suppress the oscillations using the plant's prescribed control rod shutdown insertion sequence.

   c. After inserting control rods, monitor the LPRM upscale alarm indicators and verify (using recommendation 5b) that any LPRM upscale alarms which are received are not the result of neutron flux limit cycle oscillations.

   d. When restarting recirculation pumps (or switching from low to high frequency speed for flow control valve plants), the operation should be performed below the 80% rod line.

Table 4.3-1 SIL-380

  e. Once pumps have been restarted and recovery to power is to commence, follow recommendations in Section 6.

6. When withdrawing control rods during startup in dotted region of Figure 4.3-6:

   a. Monitor the APRMs and the LPRMs surrounding control rod movement continually as power is being increased or flow is being reduced for any abnormal increase in the normal neutron flux response.

   b. Monitor the LPRM upscale alarm indicators and verify (using recommendation 5b) that any LPRM upscale alarms which are received are not the result of neutron flux limit cycle oscillations.

   c. Operate the core in as symmetric a mode as possible to avoid asymmetric power distributions.   When possible, control rods should be moved in octant (sequence A) and, quadrant mirror (sequence B) symmetric patterns.   Control rod movement should be restricted to no more than 2 feet at a time and control rods within a symmetric rod group should be within 2 feet of each other at all times.   For BWR/6 plants with ganged rod withdrawal, control rods should be moved in gangs as much as possible to maintain symmetric rod patterns.

   d. If there is any abnormal increase in the normally expected neutron flux response, the variations should be suppressed.   It is suggested that the operation which caused the increase in neutron flux response be reversed, if practical, to accomplish this suppression.   Control rod insertion or core flow increase (PCIOMRs should be followed during flow increases) will result in moving toward a region of increased stability.

   e. An alternative to recommendations 6 a-d is to increase core flow such that operating in region 2 is avoided.   PCIOMR guidelines should still be followed.

7. When performing control rod sequence exchanges:

   a. Follow recommendations 6 a-d, or

   b. Perform control rod sequence exchanges outside of both regions of Figure 4.3-6.

8. When inserting control rods during shutdown, insert control rods to or below the 80% rod line prior to reducing flow into dotted region of Figure 4.3-6 (i.e., avoid crosshatched region during shutdown).

9. Should any abnormal flux oscillations be encountered data should be recorded on the highest speed equipment available and all available power, flow, power shape, feedwater, pressure and rod pattern information documented for subsequent evaluation and operational guidance.

Table 4.3-2 NRCB 88-07 Supplement 1

1. Intentional operation shall not be allowed in Region A or Region B of Figure 4.3-8.

2. Group 1 plant operators shall take immediate actions to exit the region. Immediate action consists of either:

   Insertion of predefined set of control rods which will most effectively reduce core thermal power.

   or

   Increasing recirculation pump speed if one or more pumps are in operation. Starting a recirculation pump to exit this region is NOT an appropriate action.

   Group 2 plant operators shall manually scram the reactor to exit the region.

3. If Region B is unintentionally entered:

   Group 1 and Group 2 operators shall take immediate action to exit the region. Immediate action consists of:

   Insertion of a predefined set of control rods which will most effectively reduce core thermal power.

   or

   Increasing recirculation pump speed or recirculation flow (FCV plants) if one or more pumps are in operation. Starting a recirculation pump or shifting from low to high speed (FCV plants)to exit this region is not an appropriate action.

4. Intentional operation in Region C shall be allowed only for control rod withdrawals during startup requiring PCIOMR. This region should be avoided for control rod sequence exchanges, Surveillance testing and reactor shutdowns.

   During control rod withdrawal, flux monitoring should be conducted in accordance with SIL 380, Revision 1.

5. If at any time during operation in Region A, B, or C, core thermal hydraulic instability occurs, the plant operator shall manually scram the reactor.

   Evidence of thermal hydraulic instability consists of APRM peak to peak oscillations of greater than 10% or periodic LPRM upscale or Downscale alarms in addition to the guidance provided in SIL 380, Revision 1.

Table 4.3-2 NRCB 88-07 Supplement 1

Group 1

    Oyster Creek
    Nine Mile 1
    Dresden 2,3
    Millstone
    Quad Cities 1,2
    Pilgram
    Montecello
    Duane Arnold
    Cooper
    Vermont Yankee
    Peach Bottom 2,3
    Limerick

Group 2

    Brunswick 1,2
    Hatch 1,2
    Browns Ferry 1,2,3
    Fermi 2
    Fitzpatrick
    Hope Creek
    Susquehanna 1,2
    LaSalle 1,2
    Hanford 2
    Shoreham
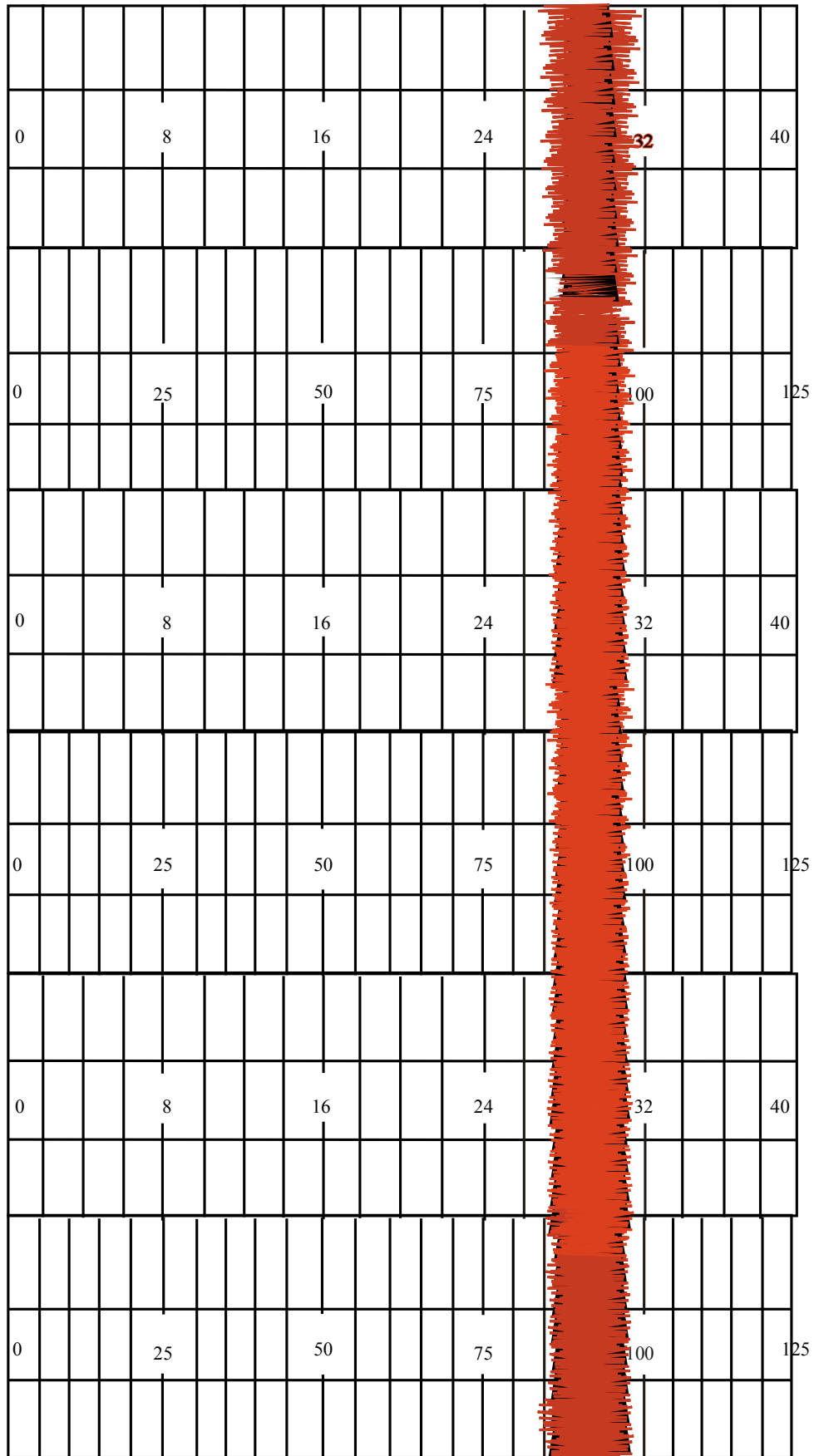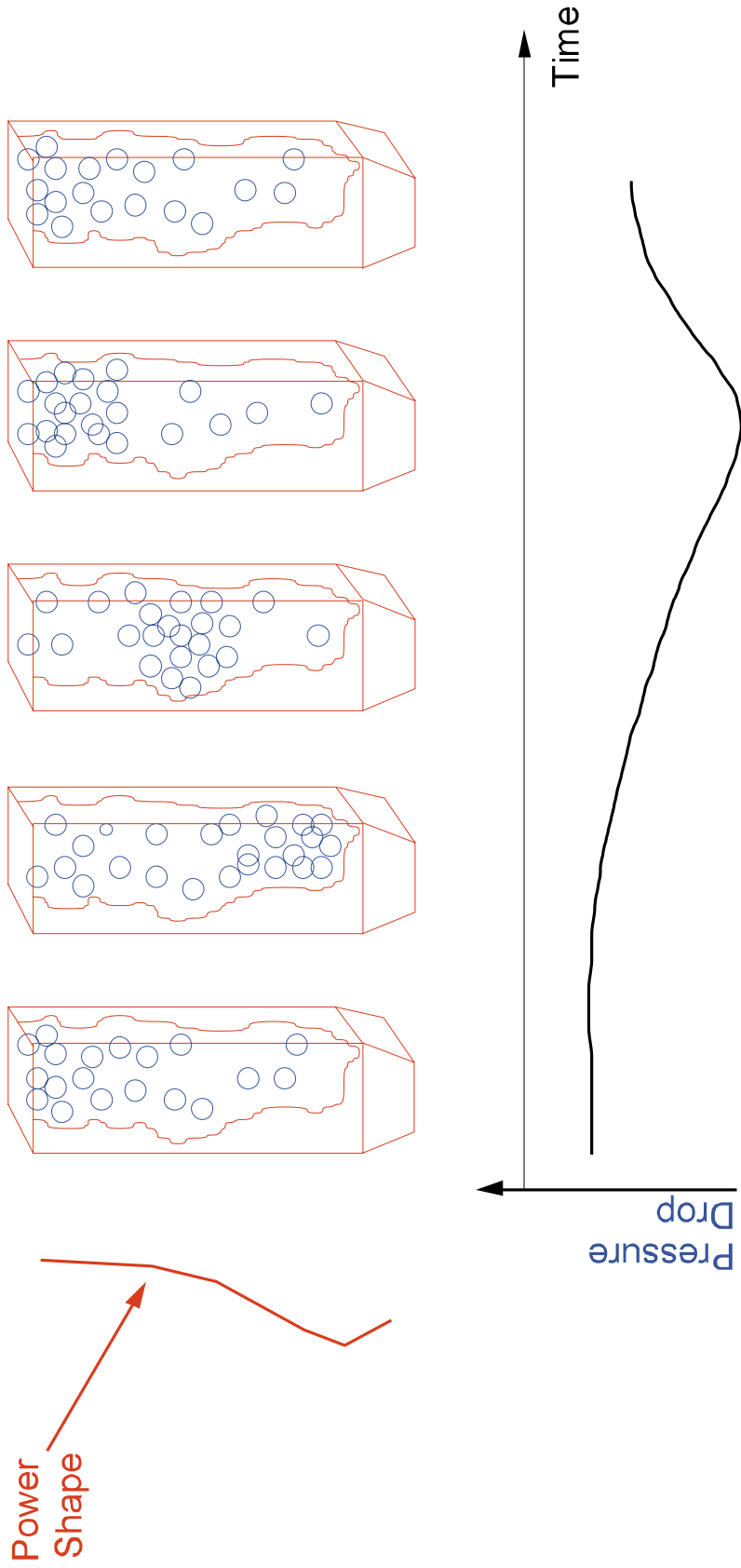   Nine Mile PT 2
    Clinton
    Perry
    Riverbend
    Grand Gulf 1

Figure 4.3-1 Normal Observed Power Oscillation on APRMs

Figure 4.3-2 Density Wave Mechanism

Total
Pressure
Drop

Density wave time delay

Local Pressure Drops

Local
Pressure
Drops

Inlet
Flow

Time
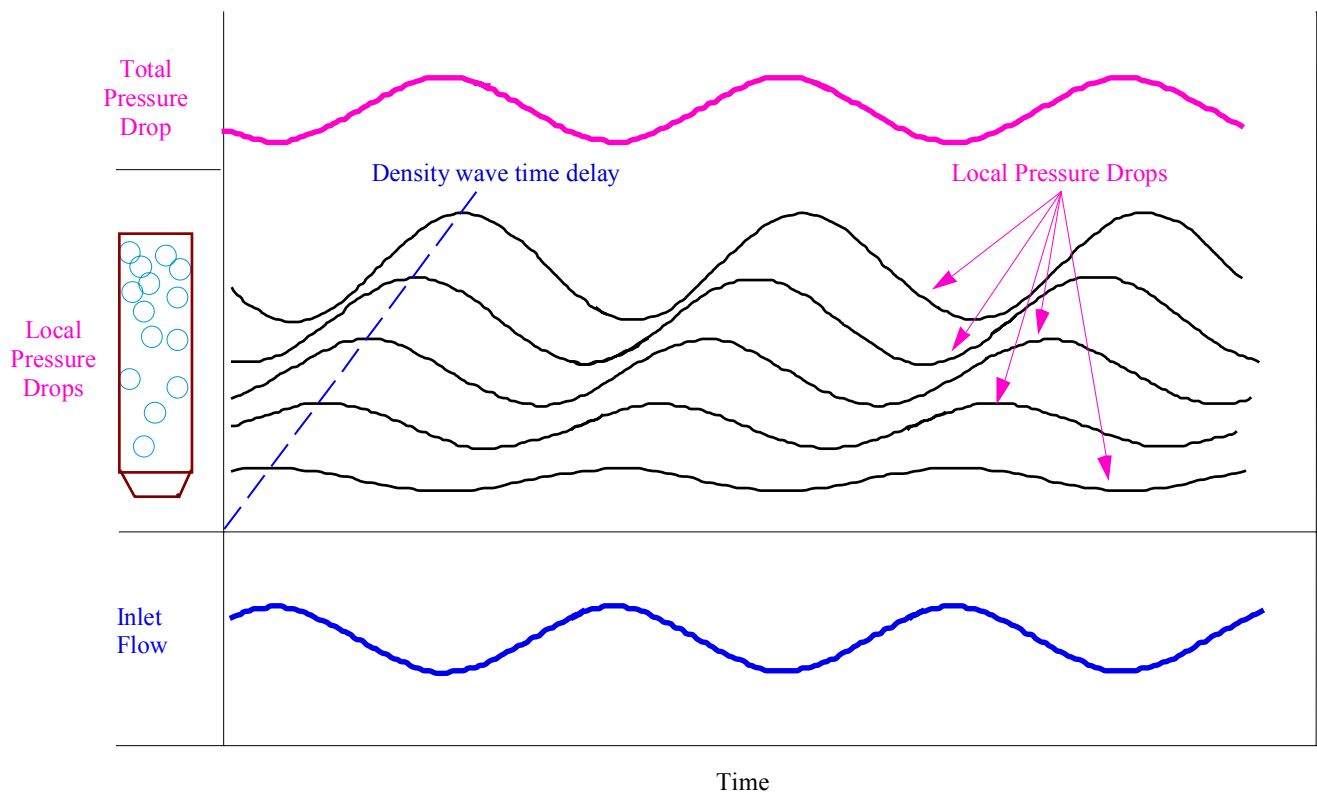
Figure 4.3-3 Sinusoidal Pressure Drop
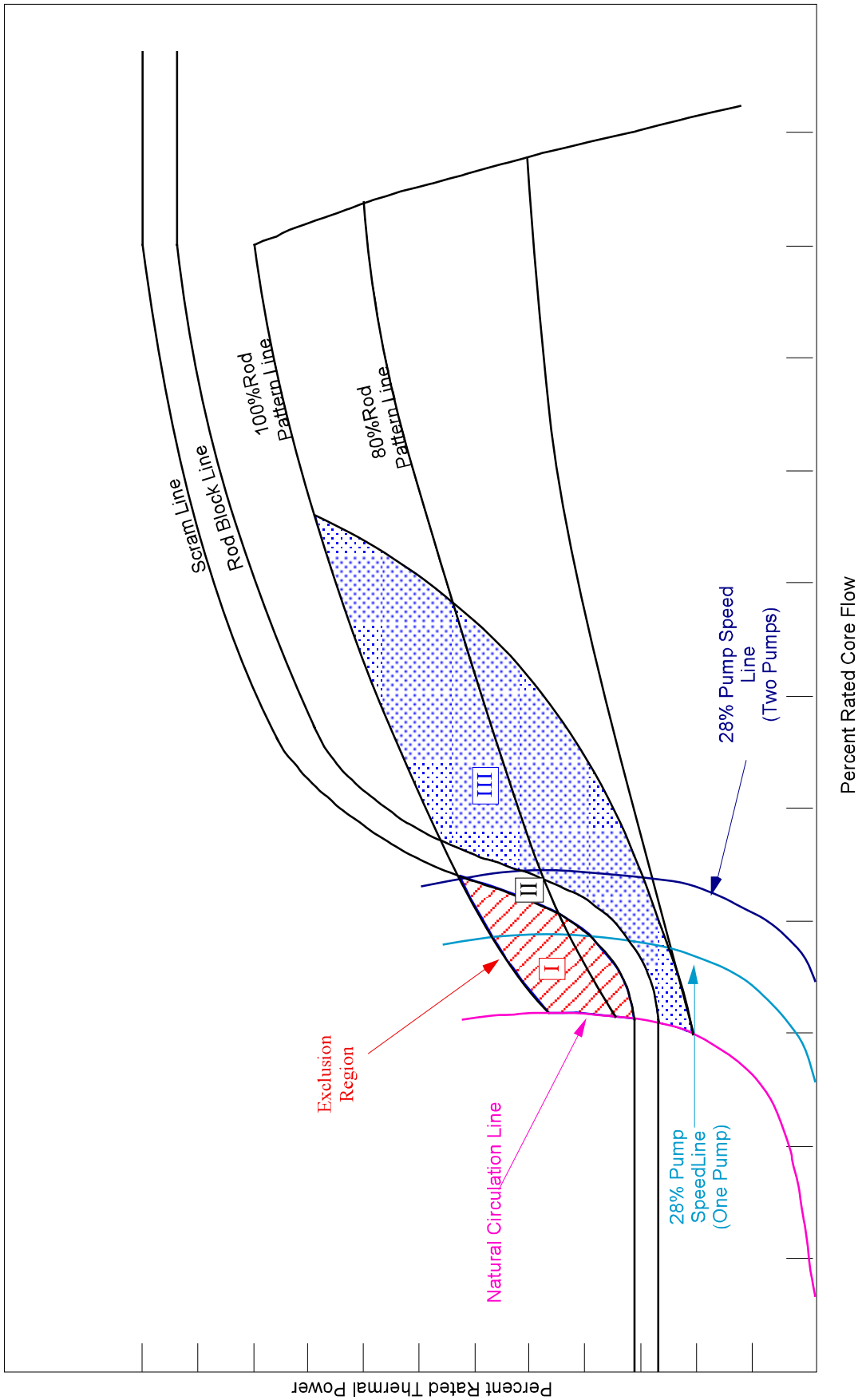
Figure 4.3-4 Reduced Rod Block & Scram Stability Protection for Option I-A
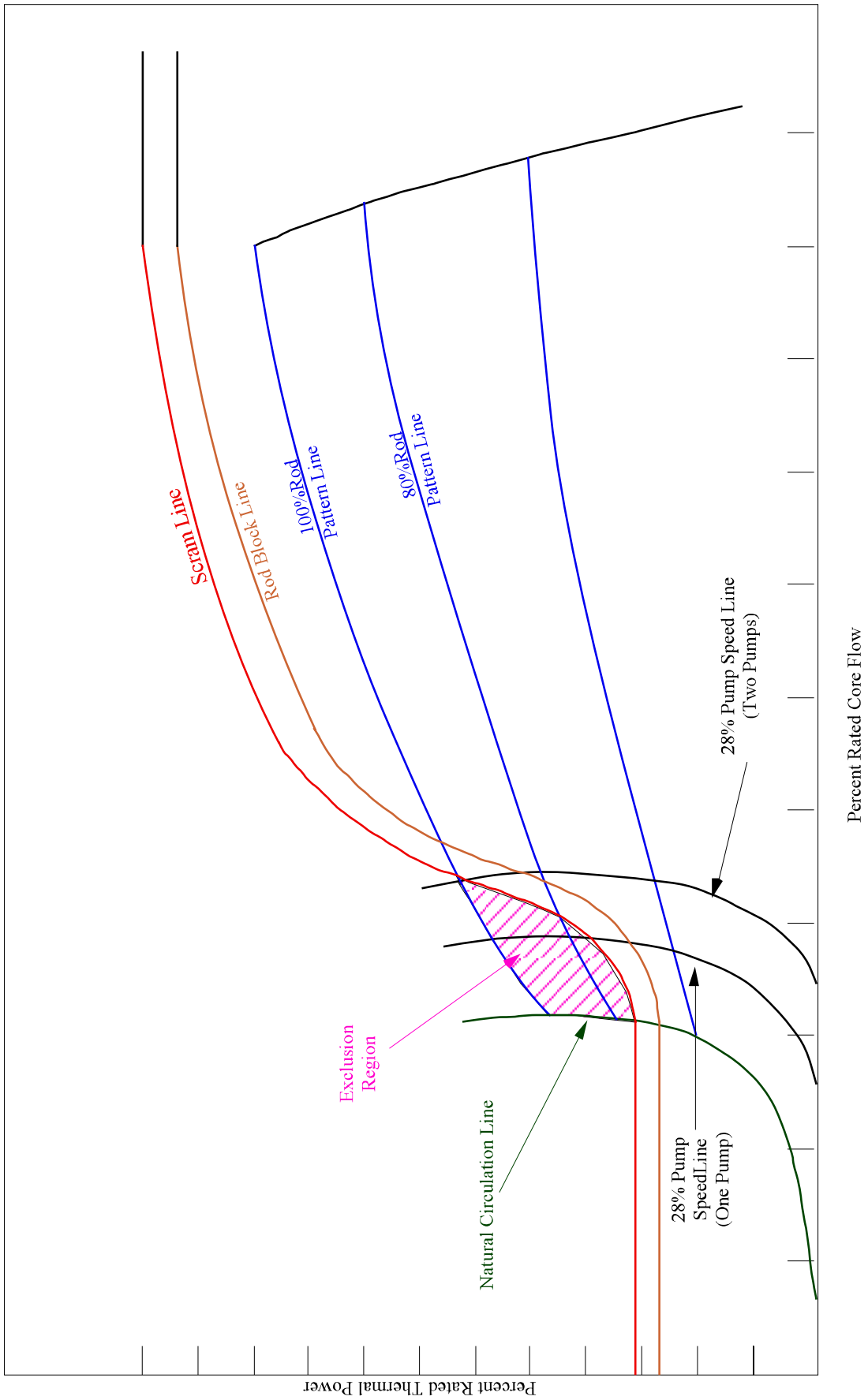
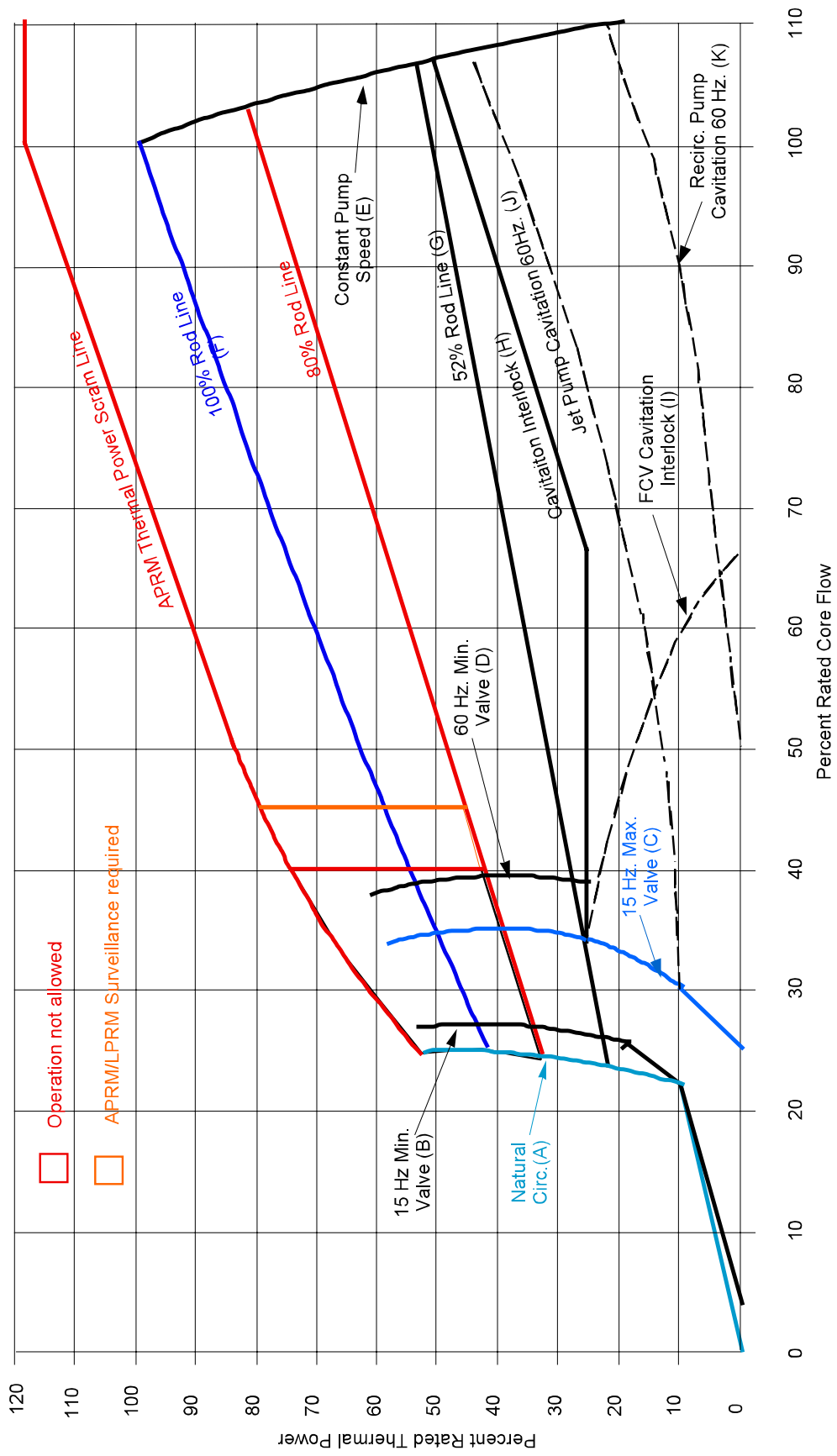**Figure 4.3-5 Administratively Controlled & Instability Region proposed for Option I-D**

Scram Line

Rod Block Line

100%Rod Pattern Line

80%Rod Pattern Line

28% Pump Speed Line (Two Pumps)

Exclusion Region

Natural Circulation Line

28% Pump SpeedLine (One Pump)

Percent Rated Thermal Power

Percent Rated Core Flow

Figure 4.3-6 SIL - 380 Power/Flow Map
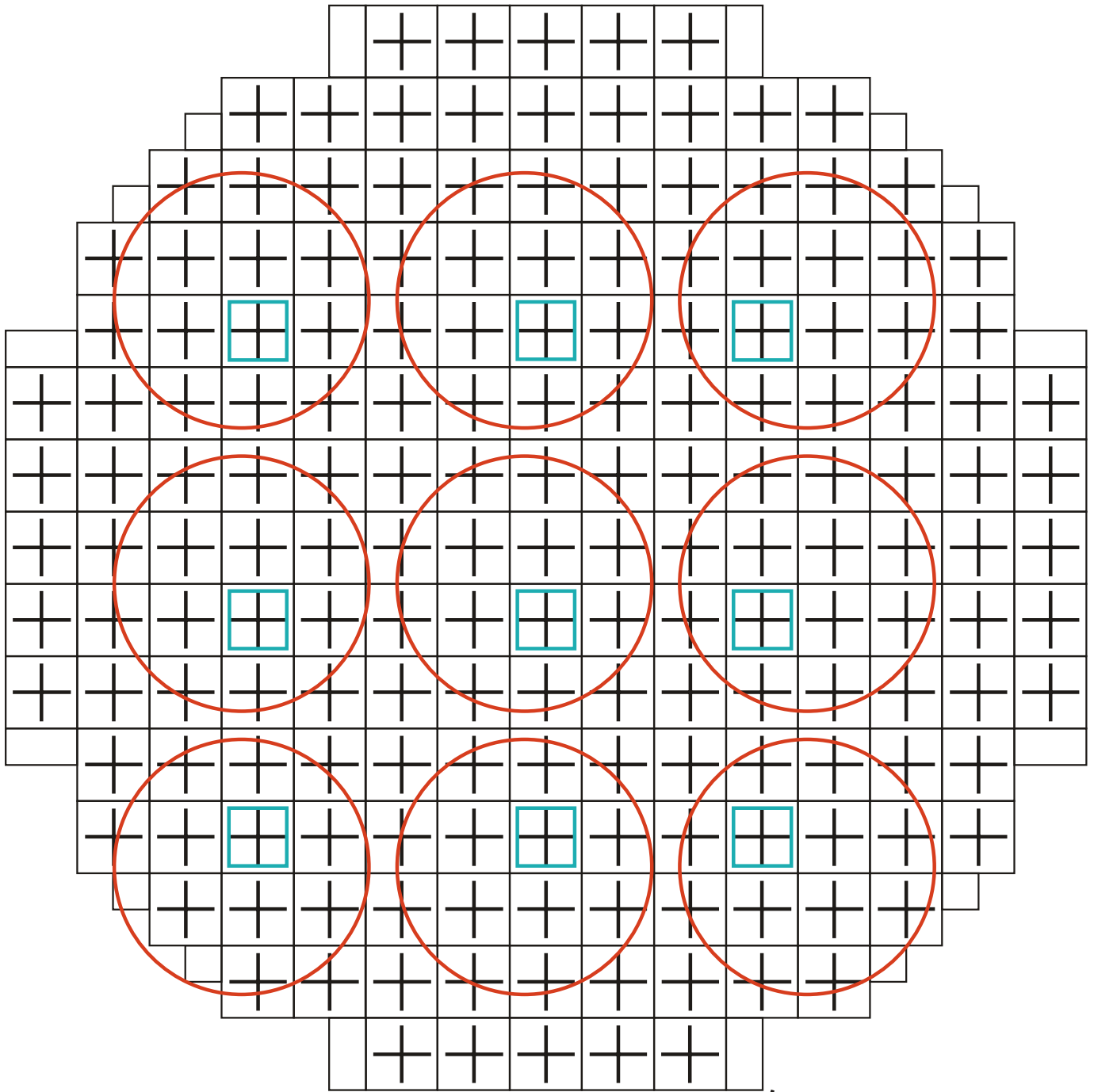
Figure 4.3-7 Typical Local Region Monitoring Scheme
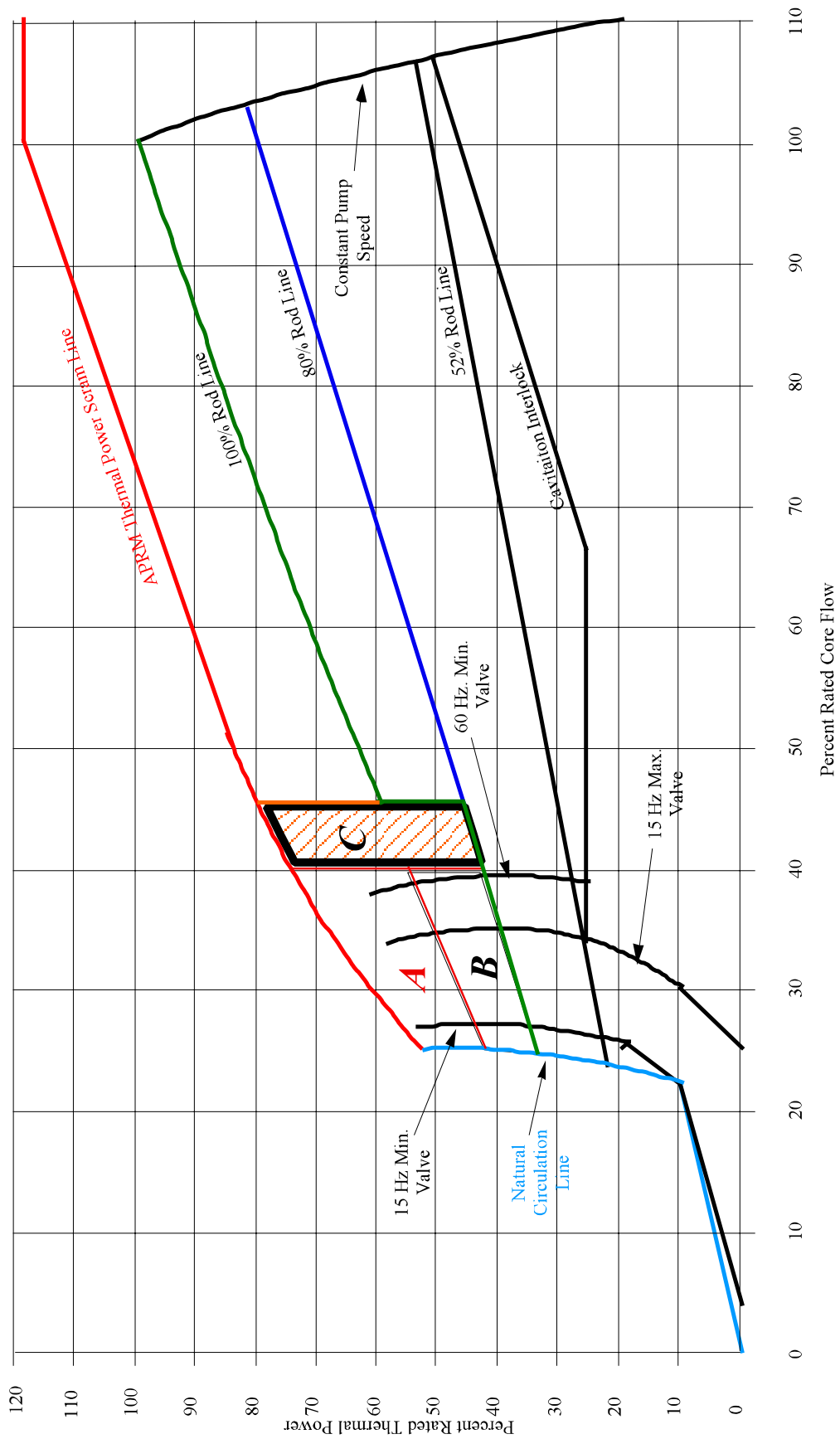
Figure 4.3-8 NRCB Power/Flow Map

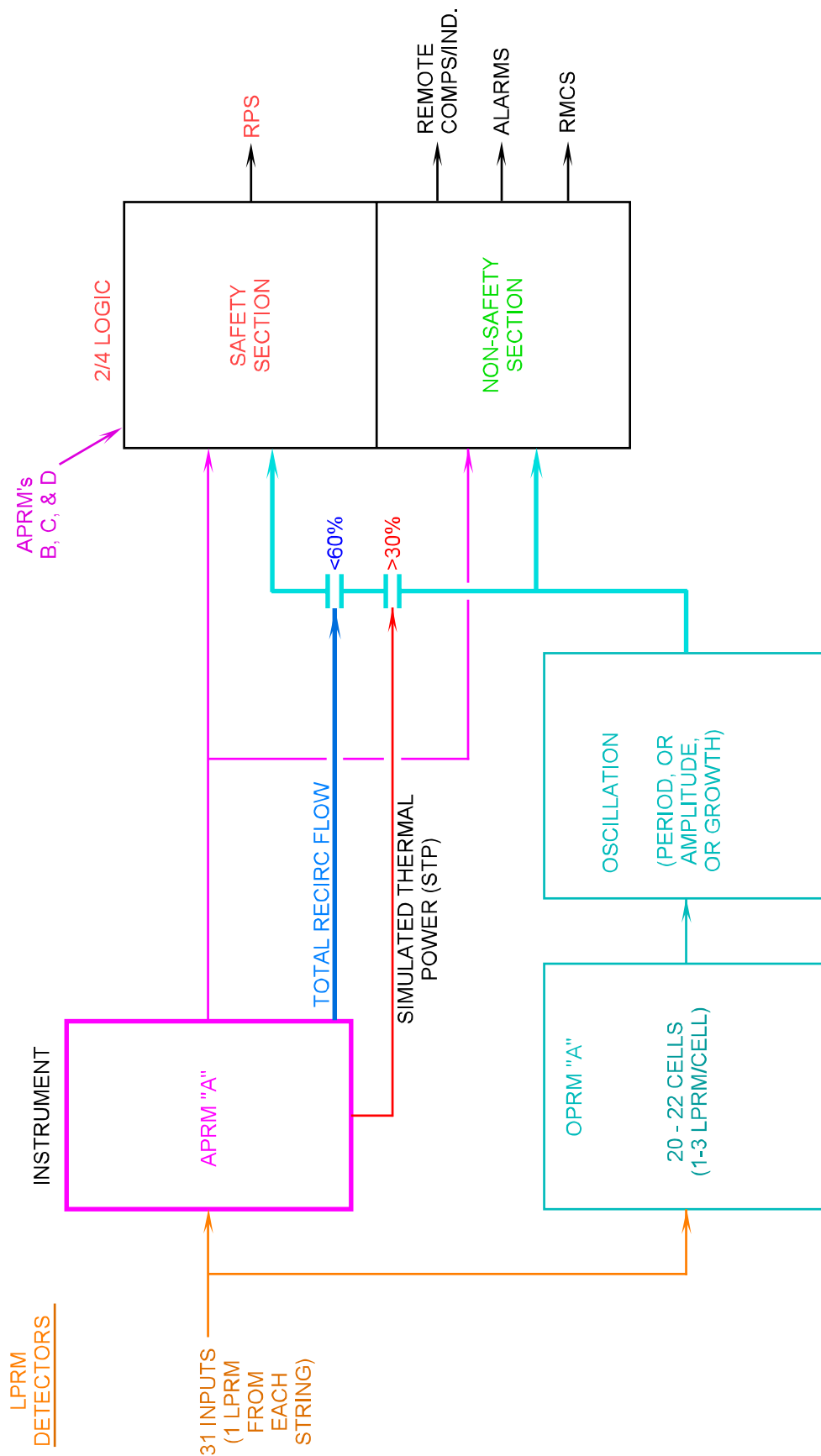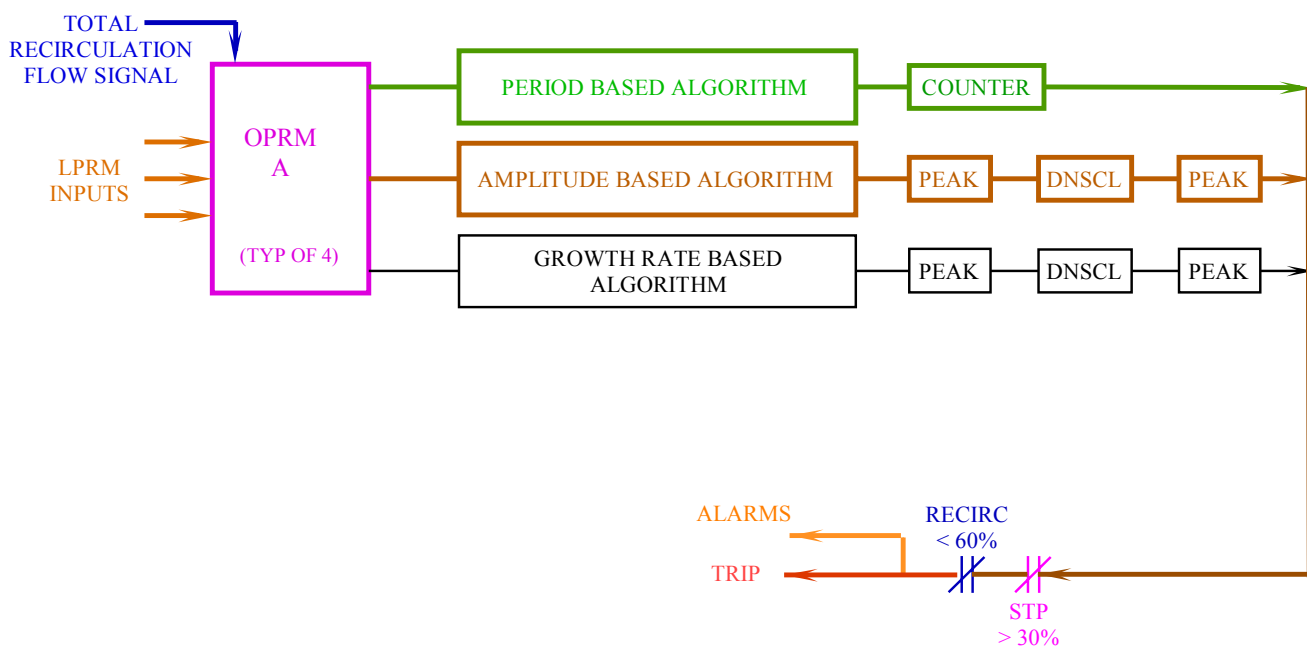**Figure 4.3-9 APRM, OPRM, 2/4 Logic Relationship**

Figure 4.3-10 OPRM Trip Channel