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# FOCUS ON SAFETY

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In the initial NRR Technical Newsletter in May 1988, I noted that I would contribute articles from time to time. This is my last note, and it will be brief.

NRR has a central role within NRC in assuring the continued safe operation of nuclear power plants in the United States. The staff has high technical qualifications, and is motivated to meet its responsibilities. It is the challenge of managers to maintain the strong safety culture in NRC and especially to prevent complacency among the NRC staff and among licensees. A further challenge to all is to maintain the focus on operational safety and not let the staff's intellectual energy be diverted completely by the daily distractions that are a fact of life in NRC.

I leave with pride in the NRR organization and with confidence that these challenges will be met.

# IN THIS ISSUE




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## Stress Corrosion Cracking of BWR Core Shrouds


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### Background

The internal components of many boiling-water reactor (BWR) vessels are made of materials susceptible to intergranular stress corrosion cracking (IGSCC), including stainless steel, alloy 600 and alloy 182 weld metal. IGSCC is a time-dependent material degradation process, known to be accelerated by the presence of crevices, residual stresses, material sensitization, irradiation (irradiation-assisted SCC, IASCC), cold work, and corrosive environments.

Cracking of BWR core shrouds, reported in 1993, has been a significant material degradation event in BWR internals. As operating BWRs age, the probability of cracking is expected to increase. The General Electric Company (GE) recommended that utilities inspect shrouds, and the Boiling Water Reactor Owners Group (BWROG) developed a plan to address the issue.

In addition to core shrouds, other BWR internals, such as access hole covers, jet pump hold-down beams, and top guides, have experienced IGSCC. Cracking of access hole covers has been detected at 14 domestic BWRs, necessitating remedial actions by the licensees. A jet pump hold-down beam failed at Grand Gulf. Cracking was also found in the top guide of one of the older BWRs. NRR continues to evaluate BWR internal cracking on a case-by-case basis.

### Recent Core Shroud Cracking Incidents and Industry Actions

Cracking of the core shroud was observed visually at the Swiss KKM reactor in 1990. The crack was discovered in the heat-affected zone (HAZ) of a circumferential weld in the lower shroud. GE reported this cracking in its Rapid Information Communication Services Information Letter (RICSIL) 054.

In the U.S., a number of BWR licensees have recently examined their core shrouds visually in accordance with the recommendations of RICSIL 054 or in GE's Services Information Letter (SIL) 572 and SIL 572, Rev. 1, issued in 1993 to incorporate Brunswick's experiences and observations of cracking in mid-1993. The scope of SIL 572 is more comprehensive because it recommends licensees review the fabrication and operational histories of their plants. It also provides a more detailed set of criteria for the inspections.

In addition to Brunswick Unit 1, Peach Bottom Unit 3 also reported core shroud cracking. Both plants have experienced axial and circumferential cracking of shroud welds located at the core midplane level, and circumferential cracking at the horizontal weld which joins the lower shroud to the top guide support ring. The cracking at the top guide support ring weld (H-3) at Brunswick is significant, extending nearly 360 degrees around the shroud at a maximum depth of 1.7 inches. Some cracking at welds associated with the upper shroud was also found at Brunswick. Carolina Power and Light Company (CP&L) proposed a modification of the Brunswick shroud which involves installing 12 mechanical clamps around the degraded H-3 weld. Philadelphia Electric Company (PECo) performed a flaw evaluation of the cracking in the core shroud at Peach Bottom and determined that the cracking is not significant enough to threaten the structural integrity of the shroud during the next plant operating cycle. NRR staff agreed with CP&L and PECo conclusions.

In SIL 572, Rev. 1, GE recommends that shrouds fabricated from low-carbon ($<$ 0.03% C) austenitic stainless steels be visually or ultrasonically inspected after 8 full-power years (FPYs) of operation, and that shrouds fabricated from high-carbon-content austenitic stainless steels be visually or ultrasonically inspected after 6 FPY of operation. GE also recommends that shrouds be reinspected at every subsequent refueling outage or every two refueling outages, depending on whether or not cracking is observed in the shroud.

GE has developed a set of screening criteria for evaluating structural integrity of the core shrouds based on both limit load and linear elastic fracture mechanics methods. GE has also developed a generic safety assessment (GE white paper) of core shroud cracking which evaluates the effects of normal operating, design-basis accident, and seismic event conditions on postulated 360 degrees circumferential shroud cracks.

### BWROG Activity

The BWROG has developed a Core Shroud Cracking Action Plan and worked in conjunction with GE to develop the generic safety assessment (GE white paper). In addition, the action plan includes plans for compiling and evaluating the data submitted by BWR licensees who have performed shroud inspections during 1993 Fall/Winter refueling outages, and for developing generic core shroud inspection guidance and acceptance criteria. The NRC staff is closely following this industry initiative and will review the generic core shroud inspection guidance and acceptance criteria when they are ready for review.

### Regulatory Activities

The NRC issued Information Notice 93-17, "Core Shroud Cracking at Beltline Region Welds in Boiling Water Reactors," on September 30, 1993, to inform the industry of the cracking discovered at Brunswick. The NRC met with the BWROG in December to discuss the status of the Owners Group's Action Plan. The need for further generic communications will be determined as additional core shroud inspection data becomes available.

NRR has performed a Core Shroud Cracking Preliminary Safety Assessment. The staff evaluated the potential consequences of a shroud failure during normal, transient, and accident conditions. The staff's preliminary assessment concluded that for a 360 degrees through-weld IGSCC crack at the H3 weld location and under certain low-probability accident conditions: (1) the core configuration may not be maintained, (2) control rods may not be inserted, (3) some emergency core cooling system (ECCS) equipment may be damaged and unavailable for core cooling, and (4) severe core damage may occur. For these scenarios, the staff has preliminarily estimated that the probability of core damage is on the order of 10-4 to 10-5/reactor-year. Although not quantified, the staff also believes that the significant cracking required to result in a loss of structural integrity of the H3 weld would likely be observable during normal plant operation and would result in a safe plant shutdown.

### Region I Core Shrouds

A survey was sent to all BWRs in Region I to determine the current status of the core shroud, future inspection plans, and technical parameters which indicate core shroud susceptibility to cracking. The responses were tabulated and sent to NRR/DE/EMCB for consideration. Region I DRS/EB/MS is continuing to monitor the utilities' activities on this issue. Inspection personnel will review the results of the plant examinations.

## Canadian Integral Systems Test Facility – RD-14M

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### Background

Today most industries use some form of model and/or prototype (or pilot) plant testing in the design and construction process. Each, the model and the prototype, has advantages and disadvantages. Model testing is usually lower in cost, easier to modify, and better instrumented, but may have phenomenological distortions due to scaling. Prototype testing can produce realistic responses and behavior characteristics, but, in the case of a nuclear power plant, can be prohibitively expensive. Many of the countries involved in nuclear power plant design and construction build and operate scale model test facilities in support of their nuclear programs. As part of the preapplication review being performed on the CANadian DeUterium 3 (CANDU 3) design developed by Atomic Energy of Canada Limited (AECL), the staff has begun to look at results from the RD-14M scaled integral system test facility.

### Objectives

The AECL philosophy behind the entire RD test program has been to provide integral systems test data on thermal-hydraulic behavior on a multiple-channel test facility, and to facilitate computer code assessment. The AECL staff has been emphatic in pointing out that the simulation of CANDU accident and transient response is not attempted with the test facility. Numerous tests have been performed, however, to evaluate the effect of specific hardware components, such as header interconnect lines.

### History

The RD-14M test facility is the latest in a 20-year series of integral systems test facilities at the Whiteshell Laboratory (Ref. 1). Each new facility has represented an improvement to an existing facility. The AECL test facilities are:

- 1974-76: RD-4 Small scale
- 1976-83: RD-12 Half height scale
- 1983-87: RD-14 Full height, one channel/pass
- 1987- : RD-14M Full height, multiple-channel (5)/pass

### Description

The RD-14M test facility, Figure 1, is designed to be a full-height, figure-of-eight loop similar in layout to the CANDU 6 reactor design (Ref. 2). The facility includes 2 full-height steam generators, full-length fuel channel feeder pipes that are connected to volume-scaled header pipes at prototypical angles. The fuel is simulated by electrical heaters configured in horizontal pressure tubes, inside calandria tubes similar to the CANDU core. The core is simulated by five fuel channels per pass, one pass from each steam generator, arranged to span the full 6-meter elevation change of the CANDU core. Four of the ten fuel channels operate at the same power and flow as the hot channels in the CANDU 6 reactor.

The RD-14M facility is a 1:100 volume scale of the CANDU 6 reactor, operating at prototypical pressure and temperature. While the fuel channels are full length, the fuel is simulated with seven heaters instead of the 37 fuel rods in a CANDU fuel bundle to maintain the proper flow area.

The data acquisition system consists of 800 channels, including pressures, differential pressures, temperatures, power, flow rates, coolant density, and liquid levels.

One component, the pressurizer, is not properly scaled, nor is it used in the tests performed to date. The AECL staff have been careful to only test areas in which the pressurizer is not important to the progress of the event, and to only use the test results to evaluate computer code performance, not actual CANDU simulation.

## Scaling

All scaled facilities have incorporated scaling compromises in specific hardware components or phenomena. Today, designers know that facilities must be scaled to preserve tne phenomena of importance to the testing being performed. Reconfiguration may be required when different types of tests are performed.

The scaling approach taken for RD-14M was to capture the behavior that occurs in a CANDU reactor during thermoshiphoning (natural circulation), coolant system blowdow, and emergency coor cooling. To achieve this, full linear dimensions and elevations typical of CANDU reactors were maintained. Since the areas of interest involve system depressurization, the scaling was aimed at simulating the reactor void distribution that would occur.

The scaling rationale employed in these facilities is representative of the growth in understanding of phenomenological modeling. LOFT, by its very nature as a nuclear-powered facility, incorporated numerous compromises in scaling. These included fuel assembly length and the use of filler blocks surrounding the core. Semiscale, an electrically heated facility, went through many changes and configurations as the testing requirements evolved. The early design was to provide large-break LOCA support for LOFT. By the end of its life, the facility had tested small-break LOCAs, transients, and anticipated transients without scram.

Numerous scaling approaches have been developed. A recent work prepared by Larson (Ref. 3) on behalf of the NRC discusses several of those approaches and the phenomena to which they apply. AECL has followed the scaling rationale of Ishii and Kataoka (Ref. 4), since both single-phase as well as two-phase flow and natural circulation are important in the accidents and transients investigated.

Even the best scaling approach has limitations. Some of the limits for RD-14M are that the scaling is applicable only if the flow is well mixed and the void/quality relationship for homogeneous flow applies. Also, if the flow is horizontally stratified, or if horizontal or vertical annular flow occur, the similarity between CANDU/RD-14M behavior diverges. Specifically, stratified flow occurs for a mass flux as much as 30 percentlower than what would be typical of a CANDU reactor. Finally, one aspect of Ishii scaling is that while single and two-phase flows are preserved, time is not always preserved. With RD-14M, channel refilling may take longer than is typical of a CANDU reactor.

The question often comes up regarding the scale of some recent U.S. integral facilities. In looking at these facilities, one must keep in mind that in the early design periods, scaling was not as well understood as in the later stages of the facilities. Distortions existed in the earlier facilities which were not fully explained until computer code analyses of test results began to uncover the effects of the scaling. Other distortions were known, such as those due to use of external vessel downcomers, but were necessary hardware compromises. A few of the important PWR integral system test programs in the U.S. follow:

| Facility | Volume Scale | Purpose |
|---|---|---|
| LOFT | 1:50 | Nuclear-LB, SBLOCA, Trans |
| Semiscale | 1:1,706 | LB, SBLOCA, Trans |
| MIST (2x4) | 1:817 | Post SBLOCA, Trans |
| UMCP (2x4) | 1:522 | NC after SBLOCA |
| SRI (2x4) | 1:1,367 | Scale comparison for UMCP |

The efforts to better understand the Three Mile Island Unit 2 accident led to the Multiloop Integral System Test (MIST), University of Maryland at College Park (UMCP), and Stanford Research Institute (SRI) facilities. Each of the B&W-oriented facilities was scaled to preserve specific phenomena of interest for specific accident and transient periods. The SRI facility included as part of its mission, investigation of some of the scaling compromises inherent in the UMCP facility.

## Tests

Testing in the RD-14M facility has been limited to three specific areas: natural circulation (thermosiphoning), loss-of-coolant accidents, and header interconnect flow stability. The philosophy of the testing program has been to test hardware concepts and to acquire data for computer code assessment rather than actual plant transient simulation. The early stages of the test program acquired large quantities of test data for natural circulation. Only a small number of the test cases have been used for code assessment purposes.

| Test Program | Number of Tests |
|---|---|
| – Natural circulation | |
| Partial inventory | 49 |
| Transition to thermosiphoning | 17 |
| – LOCA | |
| Small break | 9 |
| Critical break | 14 |
| Large break | 15 |
| – Flow stability | |
| Header interconnect size and geometry | 9 |

A major concern in all experimental programs is test reproducibility. Test reproducibility in the RD-14M program was checked by repeating several of the natural circulation tests, and one test in each LOCA series. Statistical tests (T-tests for averages and F-tests for variances) were applied in assessing repeatability of flows, periods of oscillations, temperatures, and pressures.

## Summary

AECL has developed and implemented a large test program to support the CANDU reactor design. The integral systems test facility, RD-14M, which is the hallmark of the research effort, has been designed in keeping with state-of-the-art scaling rationale and data acquisition. The test program makes a strong effort to check for test reproducibility.

Although the program may have shortcomings, such as not including a scaled pressurizer to date, the AECL staff understands those limitations and restricts testing accordingly.

## References


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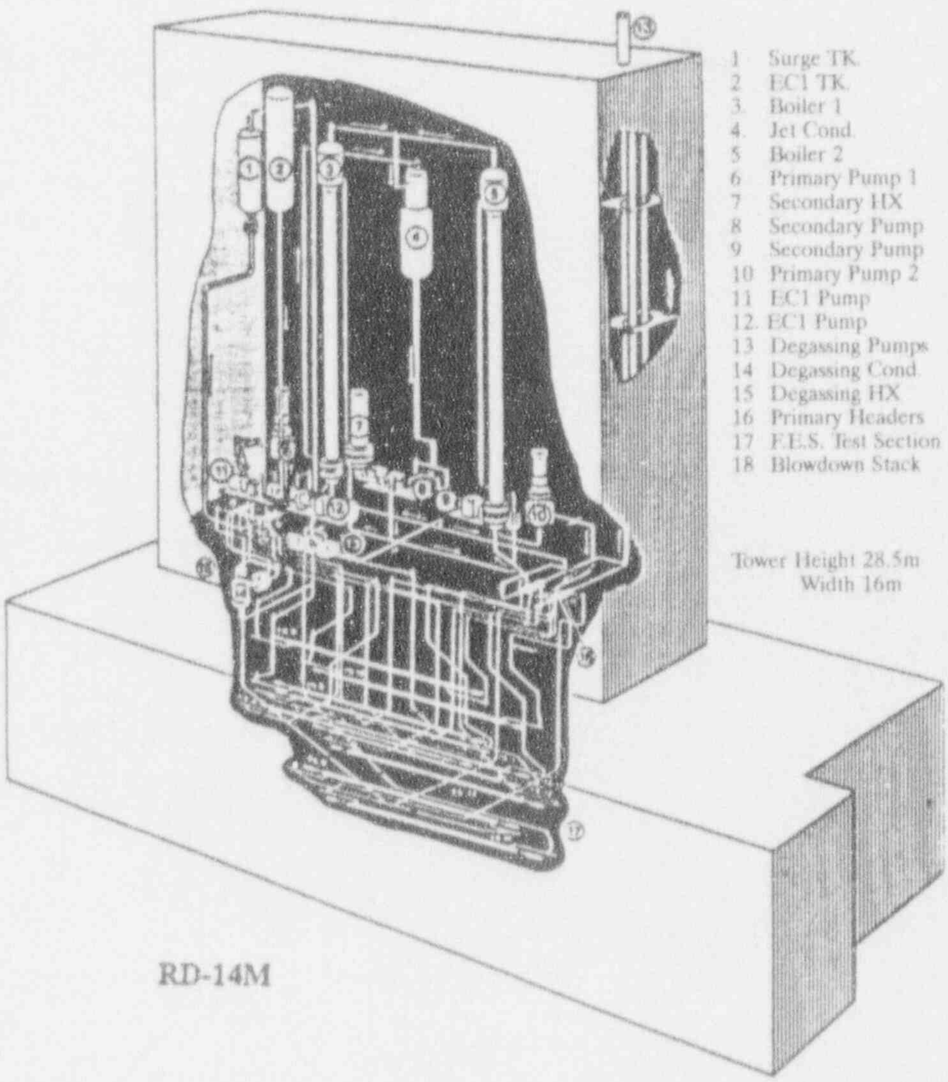


Figure 1 CANDU Test Facility

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# Technical Issue Training Bulletin
# BWR Thermal-Hydraulic Stability

October 20, 1993
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**Contents**

## Introduction

The concept of the Technical Issue Training Bulletin (TITB) was identified as a method for expedited development and distribution of training on emerging technical issues. This TITB, "BWR Thermal-Hydraulic Stability," is the second in a series of information documents. The contents of this TITB have been reviewed by the management and subject matter experts in the regions, AEOD, and NRR. Targeted NRC employees for this TITB include BWR resident inspectors, engineering support inspectors, operator licensing examiners, headquarters operation officers, BWR instructors, project managers, technical reviewers in NRR, and supervisors throughout the NRC. Any questions concerning current NRC policy on this issue should be directed to Laurence Phillips (NRR/SRXB). Any additional questions concerning this TITB should be directed to Steven Arndt (AEOD/TTC).

The TITB has been developed to convey the safety issues, relevant engineering considerations, and phenomena to NRC employees to help them with their understanding in this area. Thermal-hydraulic instability has been a regulatory concern for some time, particularly after the LaSalle event of 1988 and the WNP-2 event of 1992. The NRC is currently working with the Boiling Water Reactor Owners Group (BWROG) to finalize the methodologies to be used for long-term licensing solutions. A historical perspective of this issue appears in Appendix 1.

TITBs are not of themselves an appropriate basis for regulatory action.

## Definition of the Issue

Boiling-water reactors (BWRs) have complex dynamic responses that can result in the initiation of power oscillations. Since the terminology associated with this phenomenon is not universal, terms used in this TITB are defined in Appendix 2. 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 percent). Other control systems that affect 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 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 15 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 provides 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. 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 the section 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 3, Analysis Methods Used for BWR Stability Calculations, is, therefore, provided for additional information.

**Discussion of Technical Issues**

The basic mechanism causing flow and power instabilities in BWRs is the density wave. The effect of a density wave is illustrated in Figure 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 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 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 reenforce themselves (positive feedback), and oscillations to 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 af fect 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 negative void feedback tends to stabilize the reactor. However, 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 reenforce 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 WNP-2 event, because the most unstable bundles dominate the response.

When conditions within a reactor are such that it is unstable (e.g.,: 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 3 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 affect 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.

**Mitigation of Power Instability**

GDC 10, 12, and 20 of 10 CFR Part 50, Appendix. A, require that protection systems be designed to ensure 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 NEDO-31960, "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 setpoints and demonstrate the adequacy of the protection systems to prevent violation of MCPR limits in compliance with GDC 10 and 12 in Appendix A of Part 50.

Because of the variety of plant types, and 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 (Option II is not discussed in this TITB). However, for most of the BWRs, new or modified plant systems may be necessary. A summary of the three most promising BWROG long-term solutions is provided below.

**Solution Description Option I-A**

Regional Exclusion, Option I-A, ensures 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 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, (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 ensure that plant conditions that are sensitive to stability are bounded by the assumptions of the *exclusion region* boundary analysis.

**Solution Description Option I-D**

Regional Exclusion with Flow-Biased APRM Neutron Flux Scram, Option I-D, ensures 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 setpoint that varies as a function of core flow. When the unfiltered APRM neutron flux signal exceeds the flow-biased setpoint, a scram signal is generated. An example of the administratively controlled region and the instability region is shown in Figure 5.

**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 consists of a microprocessor unit that takes amplified LPRM signals from available locations in the APRM cabinets. These LPRM signals are grouped together so 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 exceeded. The OPRM would be installed and maintained as a reactor protection system (RPS) parameter input in the standard 1-out-of-2-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 percent 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.

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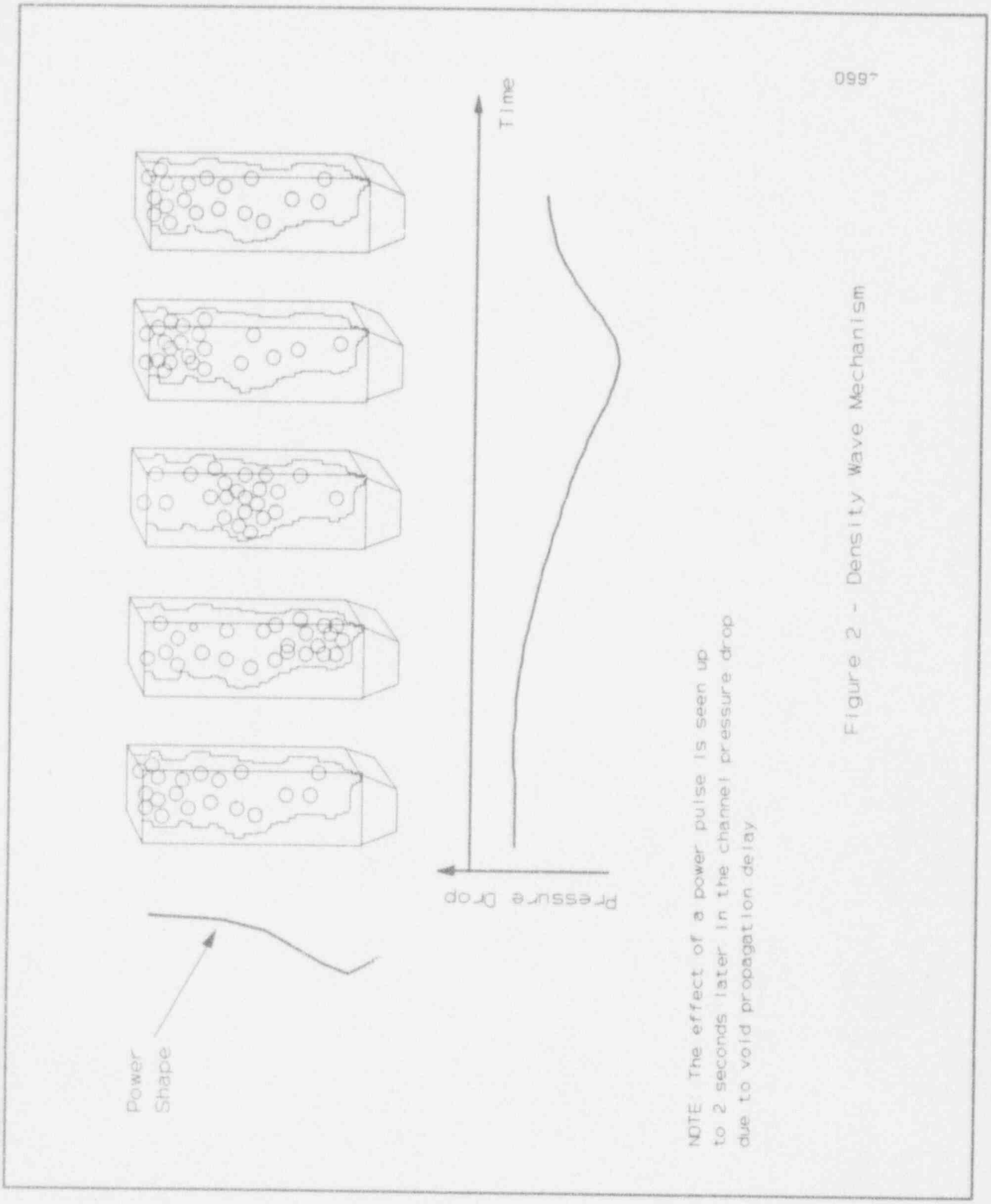


Figure 2 - Density Wave Mechanism

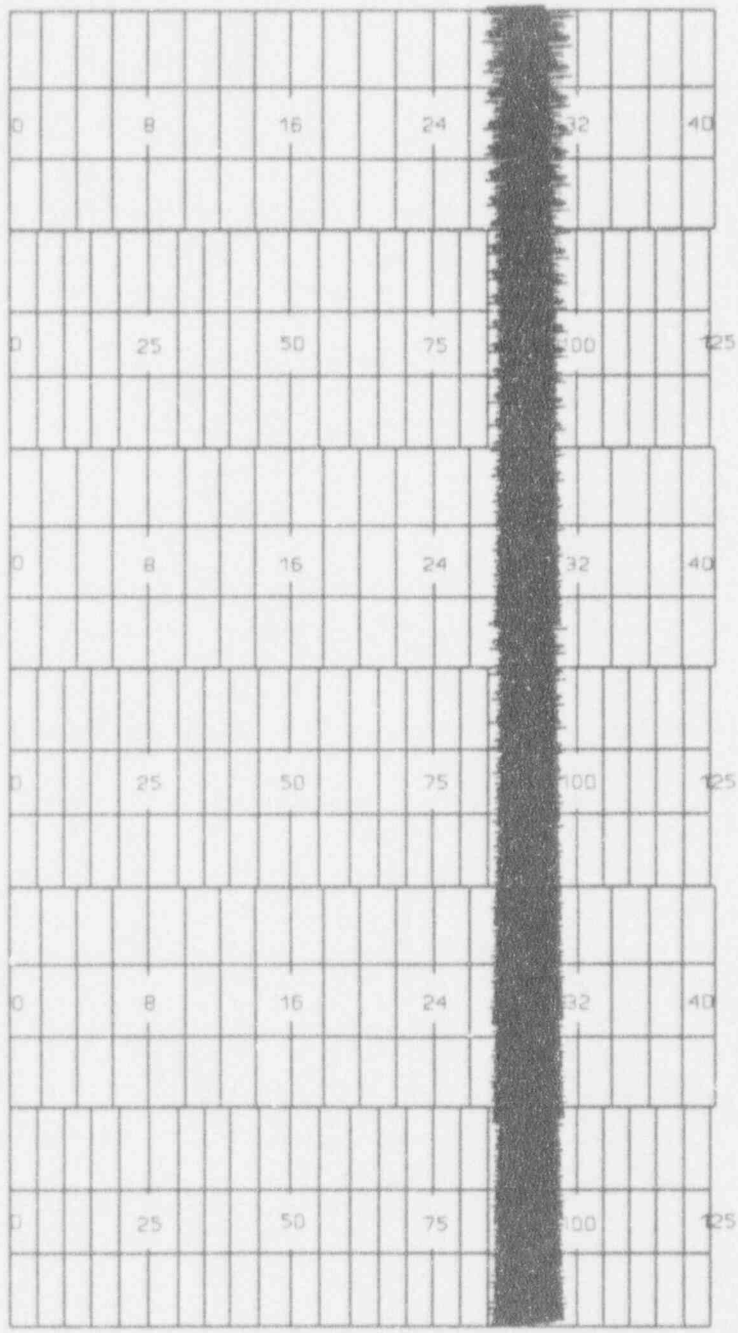


Figure 1 Normal Observed Power Oscillation on APRMs

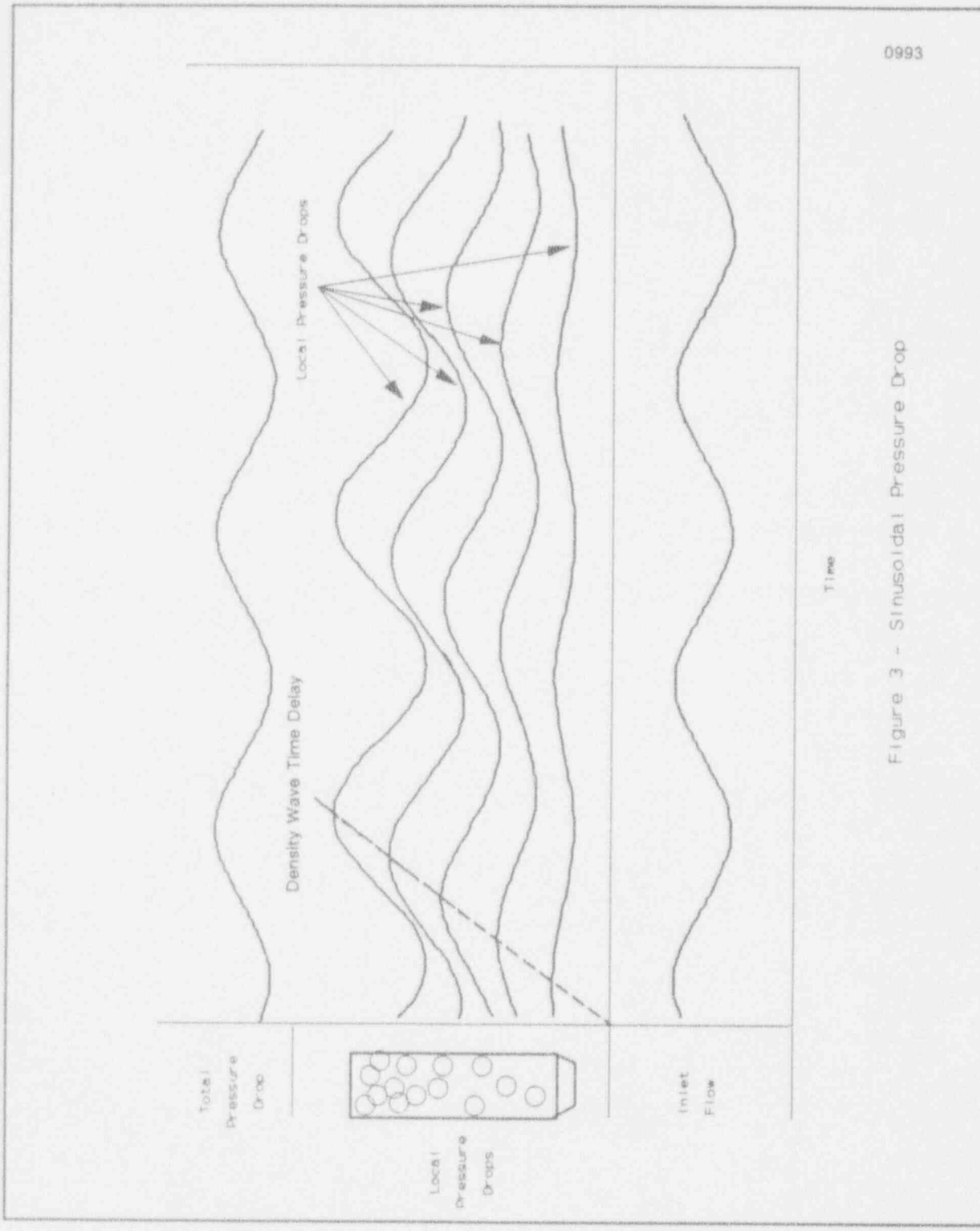


Figure 3 - Sinusoidal Pressure Drop

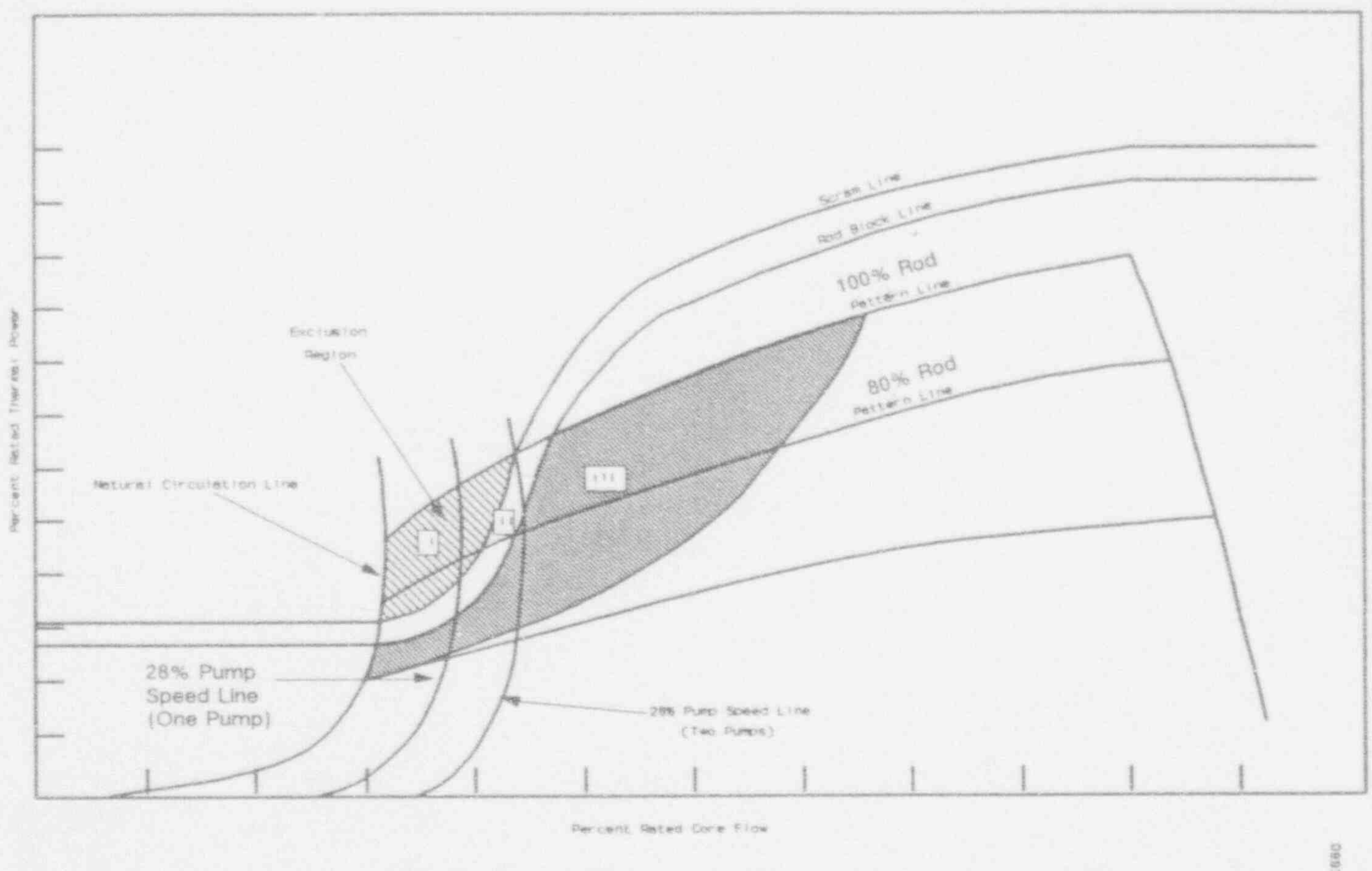


Figure 4 - Reduced Rod block and Scram Stability Protection for Option I-A

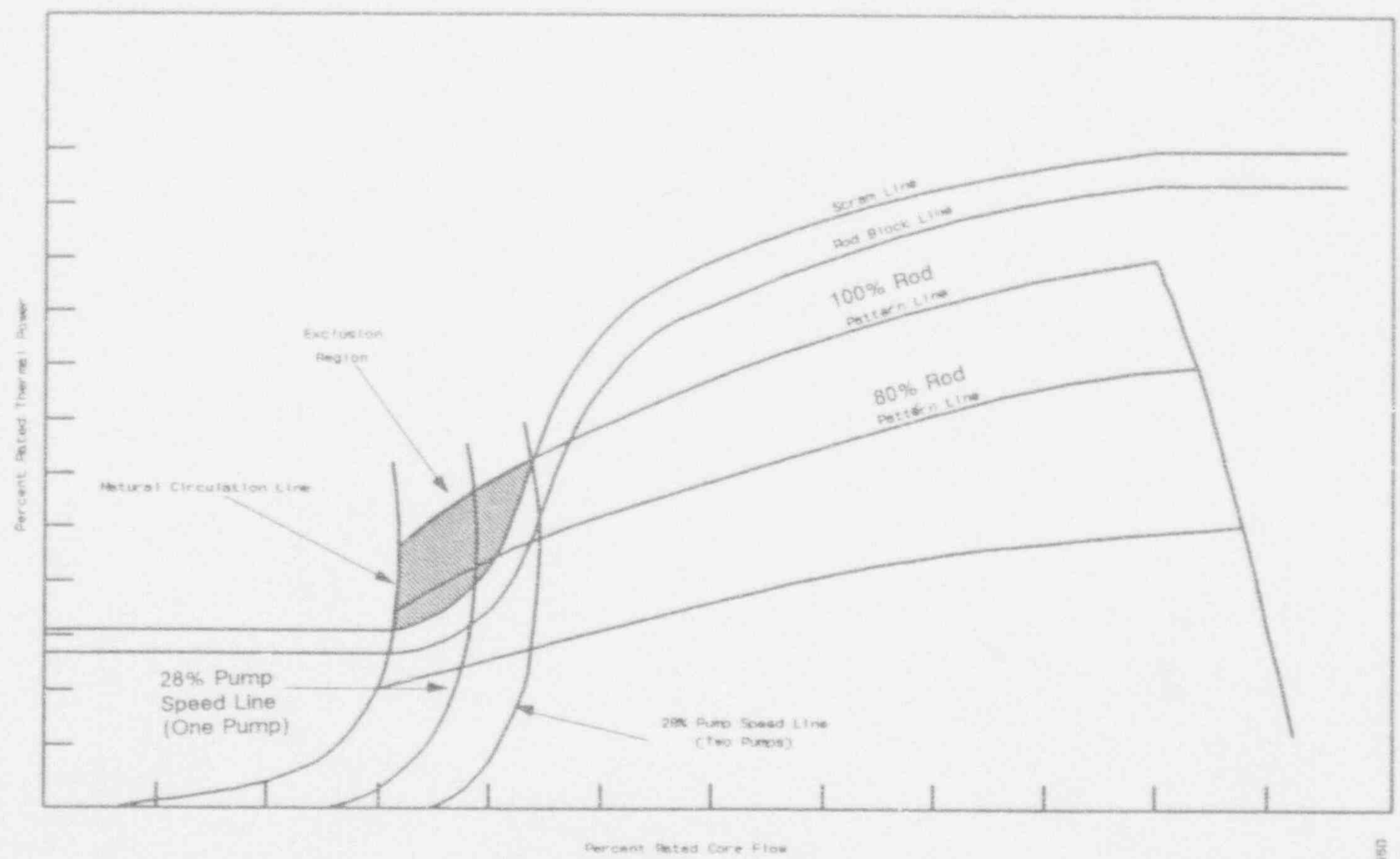


Figure 5 - Administratively Controlled and Instability Region Proposed for Option I-D

## Appendix 1 - 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 percent 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 degrees 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 percent of peak-to-peak power, APRMs indicated oscillations of only 12 percent.

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 that 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 in 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 percent steady state power and 76 percent 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 electrohydraulic (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 degrees 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 osciilating between 25 and 50 percent power every 2 to 3 seconds accompanied by oscillating LPRM upscale and downscale alarms. The unit automatically scrammed at the 7-minute mark from a fixed APRM scram signal of 118 percent.

On December 30, 1988, the staff issued NRC Bulletin 88-07, Supplement 1, dealing with pov er oscillations in BWRs. This supplement provided additional information concerning power oscillations in BWRs and to asked licensees to 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, at the Ringhals-l 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 runback from 79 percent power and 4,120 kg/s flow to 68 percent power and 3485 kg/s flow. 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 150 degrees F and 56 BTU/lb of subcooling. An interesting effect was observed 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.

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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 percent 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.

## Appendix 2 - Terminology

**Oscillation** - As applied to reactor power and control systems, oscillation refers to the increase and decrease of that parameter in a steady uninterrupted rhythm.

**Instability** - This is the condition of a system in which excessive positive feedback causes persistent, unwanted oscillations. A property of the steady state of a system such that certain disturbances or perturbations introduced into the steady state will always remain larger than the initial amplitude.

**Minimum Critical Power Ratio** - This is the minimum critical power ratio that exists in the core (where critical power ratio is the ratio of that power in a fuel assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power). Typical value is 1.07 (safety limit) with a factor of 0.24 added to establish the operational limit.

**Decay Ratio** - This is the measure of stability of a system given an external stimulus 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. A typical plant value is 0.6.

**Boiling Noise** - This is the random fluctuation in neutron flux as a result of the stochastic nature of the boiling process.

**Selected Rod Insert (SRI)** - This is an option provided for BWRs with 100 percent steam bypass capability that allows the automatic insertion of selected control rods to reduce reactor power to preclude the need for a full scram in the event of a turbine trip at 100 percent reactor power.

**Limit Cycle** - A decay ratio greater than 1.0 causes the amplitude of oscillations to diverge and become unstable. The oscillation amplitude continues to grow until it reaches a saturation value. At this point the system is said to be in a Limit Cycle.

**Limit Cycle Amplitude** - This is the peak-to-peak oscillation amplitude when the reactor is in a limit cycle. Amplitude is typically measured in percent of rated power units, but sometimes it is reported as percent of the actual reactor power at the moment of the instability.

**Transverse Wave** - Any wave in which the vibration direction is perpendicular to the direction of propagation.

**Period** - The time taken to move through one complete cycle.

**Frequency** - The number of complete cycles executed each second.

**Rod Pattern Lines** - These are lines of constant control rod configuration and xenon concentration that are traversed by changes in recirculation (core) flow on a BWR power/flow map.

**Plant Group** - This is a group of 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.

## Appendix 3 - 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 thermal-hydraulic 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 Brookhaven National Laboratory (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 Finnish 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).

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