Thermal Hydraulic Stability Methodology for Boiling Water Reactors

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CENPD-294-NP-A

Thermal-Hydraulic Stability Methods for Boiling Water Reactors

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CENPD-294-NP-A

Thermal-Hydraulic Stability Methods for Boiling Water Reactors

July 1996

ABB Combustion Engineering Nuclear Operations

CENPD-294-NP-A REPORT CONTENTS

CENPD-294-NP-A REPORT

Part I

NRC Acceptance Letter, Safety Evaluation Report (SER), and Technical Evaluation Report (TER)

ABB Combustion Engineering Nuclear Operations

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

February 22, 1996

Mr. Derek B. Ebeling-Koning, Manager
Licensing and Safety Analysis
BWR Fuel Operations
ABB Combustion Engineering Nuclear Fuel
1000 Prospect Hill Road
Windsor, Connecticut 06095-0500

SUBJECT: ACCEPTANCE FOR REFERENCING OF ABB/CE TOPICAL REPORT CENPD-294-P: THERMAL HYDRAULIC STABILITY METHODS FOR BOILING WATER REACTORS (TAC NO. M92883)

Dear Mr. Ebeling-Koning:

The staff has reviewed the subject report submitted by ABB Combustion Engineering Nuclear Fuel by letter of June 19, 1995. This report provides the description of the validation of the RAMONA code for BWR stability calculations, and is part of the ABB generic BWR reload licensing methodology. The staff has found the subject report to be acceptable for referencing in license applications to the extent specified and under the limitations stated in the enclosed report and U.S. Nuclear Regulatory Commission (NRC) technical evaluation. The evaluation defines the basis for acceptance of the report.

The staff will not repeat its review of the matters described in ABB/CE Topical Report CENPD-294-P and found acceptable when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in ABB/CE Topical Report CENPD-294-P. In accordance with procedures established in NUREG-0390, the NRC requests that ABB/CE publish accepted versions of the report, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract and an -A (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions that the report is acceptable is invalidated, ABB and/or the applicant referencing the topical report will be expected to revise and resubmit its respective

documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Robert C. Jones

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosures:
ABB/CE Topical Report CENPD-294-P Evaluation

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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REGULATION RELATING TO TOPICAL REPORT CENPD-294-P "THERMAL-HYDRAULIC STABILITY METHODS FOR BOILING WATER REACTORS" ABB COMBUSTION ENGINEERING NUCLEAR FUEL

1.0 INTRODUCTION

By letter dated June 19, 1995, Asea Brown Boveri Combustion Engineering, Inc. (ABB/CE) submitted a licensing topical report CENPD-294-P (Ref. 1) for NRC review and acceptance for referencing in future licensing actions. This licensing topical report describes the validation of the RAMONA code for BWR stability calculations, and is part of the ABB generic BWR reload licensing methodology intended to be used in support of SVEA-96 fuel in US reactors. A related ABB/CE report CENPD-295-P (Ref. 2) which was submitted in September 1995, is the subject of a separate safety evaluation report.

Topical report CENPD-294-P describes the validation of the RAMONA code for BWR stability calculations. RAMONA is a best-estimate time-domain code that originates from a development project at the nuclear research institutes of the Scandinavian countries and ABB Atom. The code development has been continued by Scandpower and Brookhaven National Laboratory (BNL). Several versions of the RAMONA code exist; for instance, the current version of BNL's code is RAMONA-4B, while the version used by ABB/CE is Scandpower's version, labeled RAMONA-3B. The review documented in this evaluation is based on information presented about Scandpower's version of RAMONA-3B.

There is significant accumulated experience in the field of BWR stability calculations. This experience indicates that input preparation is often the major source of error, and a stability code review is not complete if it is not accompanied by a set of input preparation procedures. Thus, the conclusions in this evaluation refer to the "RAMONA system", which is composed of the RAMONA-3B code and the associated input preparation procedures described in References 1 and 2.

The NRC staff was supported in this review by its consultant, Oak Ridge National Laboratory (ORNL). The staff has adopted the findings recommended in the consultant's technical evaluation report (TER), ORNL/NRC/LTR-95/33, which is attached as Enclosure 2.

2.0 EVALUATION

The attached TER provides the detailed evaluation.

3.0 CONCLUSIONS

On the basis of the staff's review in conjunction with the consultant's evaluation (Enclosure 2), the staff concludes that

1. The RAMONA-based stability methodology proposed by ABB/CE provides a reasonably accurate estimation of the stability of (1) the channel thermal-hydraulics mode, (2) the fundamental or core-wide coupled neutronics thermal-hydraulics mode, and (3) the out-of-phase or regional coupled neutronics thermal-hydraulics mode.

2. Using the ABB/CE stability methodology, RAMONA decay ratio calculations are accurate to within ±0.2 in a decay ratio range from 0 to 1.1 for all three modes.

3. The RAMONA-based option described in CENPD-294-P and CENPD-295-P is an acceptable methodology for best-estimate stability prediction of operating boiling water reactors.

4. As with all stability codes, input preparation is the major source of error; therefore, to maintain the ±0.2 accuracy, any new calculations must use procedures similar to those used in the qualification report. To insure that input errors do not compromise the accuracy of the calculations, best estimate RAMONA calculations must follow the input-generating procedures described in CENPD-294-P and CENPD-295-P. The RAMONA input must satisfy the following minimum requirements:

   (1) Each thermal-hydraulic region in the core (i.e., channel) model must be divided in a minimum of 24 axial nodes.

   (2) The core model must be divided into a series of radial nodes (i.e., thermal-hydraulic regions or channels) in such a manner that

      (a) No single region can be associated with more than 20% of the total core power generation. This requirement guarantees a good description of the radial power shape, especially for the high power channels.

      (b) The core model must include a minimum of three regions for every bundle type that accounts for significant power generation.

      (c) The model must include a hot-channel for each significant bundle type with the actual conditions of the hot channel.

   (3) Each of the thermal-hydraulic regions must have its own axial power shape to account for 3-D power distributions. For example, high power channels are likely to have bottom peaked shapes.

   (4) For out-of-phase calculations, a full-core representation is recommended. The minimum configuration, however, is two basic "symmetry units" (e.g., in a core with quarter core symmetry, RAMONA must model at least half the core).

   (5) Care must be taken in the selection of the perturbation used to excite each instability mode. A review must be performed to confirm that the perturbation actually excites each mode of oscillation (e.g., a perturbation along a symmetry line will not excite an out-

of-phase oscillation).

5. In addition to best-estimate calculations, the RAMONA-based ABB/CE stability methodology represents an adequate methodology to estimate *Exclusion Region* boundaries to be used with the so-called *BWR Stability Long Term Solutions*. Note that *Exclusion Region* calculations are not best-estimate and they require a well-defined input preparation procedure that has been specified by the Boiling Water Reactor Owners' Group (BWROG) and reviewed by the Nuclear Regulatory Commission. The so-called BWROG procedures are defined in NEDO-31960 (Refs. 3 & 4) "BWR Owner's Group Long Term Stability Solutions Licensing Methodology." In *Exclusion Region* applications using the RAMONA code, care must be taken to ensure that the axial and radial power shapes resulting from RAMONA's 3-D calculation represent as accurately as possible the power shapes prescribed in References 3 & 4. Any departure from the established BWROG procedures to calculate *Exclusion Regions* must be justified.

4.0 <u>REFERENCES</u>

1. CENPD-294-P, *Thermal-Hydraulic Stability Methods for Boiling Water Reactors*, ABB Combustion Engineering Nuclear Operations Report, May 1995.

2. CENPD-295-P, *Thermal-Hydraulic Stability Methodology for Boiling Water Reactors*, ABB Combustion Engineering Nuclear Operations Report, September 1995.

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

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

ENCLOSURE 2

ORNL/NRC/LTR-95/33

Contract Program:	Technical Support for the Reactor Systems Branch (L1697/P2)
Subject of Document:	**Review of ABB Combustion Engineering Thermal-Hydraulic Stability Methods for Boiling Water Reactors: the RAMONA Code**
Type of Document:	**Technical Evaluation Report**
Author:	**José March-Leuba**
Date of Document:	**October, 1995**
NRC Monitor:	**T. L. Huang, Office of Nuclear Reactor Regulation**

Prepared for
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under
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Prepared by
Instrumentation and Controls Division
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MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under Contract No. DE-AC05-84OR21400

SUMMARY

This report documents our review of the thermal-hydraulic stability methods proposed by ABB Combustion Engineering (ABB-CE) Nuclear Operations to analyze the stability of boiling water reactors (BWRs) reload cores. Our review is based primarily on the information presented in topical report number CENPD-294-P,[1] which describes the validation of the RAMONA code for stability calculations. An accompanying report, CENPD-295-P[2] describes ABB-CE's overall methodology for BWR stability evaluations and is the subject of a separate technical evaluation report (TER).[3]

RAMONA is a best-estimate time-domain code that originates from a development project at the nuclear research institutes of the Scandinavian countries and ABB Atom. The code development has been continued by Scandpower and Brookhaven National Laboratory (BNL), with significant funding for the U.S. NRC for a number of years. Several versions of the RAMONA code exist; for instance, the current version of BNL's code is RAMONA-4B, while the version used by ABB-CE is Scandpower's version, labeled RAMONA-3B. BNL's RAMONA-4B has undergone significant changes recently to allow modeling of low-pressure transients in the simplified boiling water reactor (SBWR) and the new version may be released as RAMONA-5. The review documented in this TER is based on information presented about Scandpower's version of RAMONA-3B.

There is significant accumulated experience in the field of BWR stability calculations. This experience indicates that input preparation is often the major source of error, and a stability code review is not complete if it is not accompanied by a set of input preparation procedures. Thus, the conclusions from the present review refer to the "RAMONA system", which is composed of the RAMONA-3B code and the associated input preparation procedures described in CENPD-294-P[1] and CENPD-295-P.[2]

The main conclusion from our review is that the RAMONA system, when using input generated by the procedures described in CENPD-294-P[1] and CENPD-295-P[2], can estimate the decay ratio of a BWR operating under normal conditions for both: (a) the fundamental (core-wide) and (b) the first azimuthal (out-of-phase) modes. We also conclude that the RAMONA system can estimate the channel thermal-hydraulic stability. We estimate the accuracy of RAMONA-calculated decay ratios to be of the order of ± 0.2. The range of decay ratios where we estimate that this accuracy level applies is between 0.0 and 1.1, which covers all the expected operating domain.

As with all stability codes, input preparation is the major source of error; therefore, to maintain the above ± 0.2 accuracy, any new calculations must use procedures similar to those used in the qualification report. We specify minimum requirements to maintain this accuracy for new calculations in our Technical Recommendations Section.

DESCRIPTION OF RAMONA FEATURES

RAMONA is a best-estimate time-domain code that simulates the dynamics of a BWR. RAMONA's main output is a time series that describes the power level and thermal-hydraulic conditions at each node. By establishing an appropriate perturbation, the RAMONA output can be analyzed to determine whether the reactor is unstable (the perturbation grows in magnitude) or stable. If stable, the decay ratio can be estimated by direct measurement in the power response. The instability mode (i.e, core-wide or out-of-phase) can also be determined by examining the time response of symmetrical core locations. Some of the RAMONA-3B features that are relevant to BWR stability calculations can be summarized as follows:

(1) RAMONA models the neutron kinetics based on the three-dimensional time-dependent diffusion equation. The energy dependence is modeled using the 1½ energy-groups approximation, and boundary conditions are approximated by fast-flux extrapolation lengths and thermal-flux albedos.

(2) RAMONA has the capability to model each fuel bundle as an individual thermal-hydraulic and neutronic radial node. Often, calculations take advantage of core symmetries to reduce computation time and memory requirements.

(3) The main distinguishing feature in the RAMONA thermal-hydraulic models is the use of the integral momentum equation, which allows for a significant increase in performance without decreasing its ability to model the physical phenomena that are relevant to BWR stability.

(4) Except for the heat conduction equation and the fast-neutron-flux solution, RAMONA uses explicit numerical integration methods, which tend to reduce numerical damping.

(5) RAMONA has the capability of modeling different fuel rod performance parameter for each type of fuel present; thus allowing for its use in mixed-fuel cores.

(6) RAMONA includes a model of the vessel thermal-hydraulics and balance of plant. This model explicitly accounts for the upper plenum, steam separators, vessel dome, down comer, recirculation system, and lower plenum.

ESTIMATE OF RAMONA ACCURACY

ABB has performed a series of benchmarks of the RAMONA system against out-of-pile thermal-hydraulic tests as well as reactor tests. The reactor-test benchmarks have been performed for both core-wide and out-of-phase modes, and it includes data from jet-pump and internal-pump reactors. Some of the benchmarks were "blind," and some were even calculated before the tests were actually performed.

The RAMONA channel stability qualification was performed by benchmarks against ABB's FRIGG loop tests. The qualification basis includes benchmarks for different fuel types, including: (a) a 36-rod boiling heavy water reactor design, (b) 6×6 and 8×8 open lattice assemblies, and (c) SVEA-64 water cross design.

Topical report CENPD-294-P[1] presents results from 36 channel-stability benchmarks, and overall RAMONA predicts conservative critical or instability-threshold powers. The error in predicted threshold powers is always less than 20% and usually within less than 10%. A notable exception is the SVEA-64 cases, where the accuracy is lower than with the open-lattice bundles. For the SVEA-64 benchmarks, RAMONA conservatively under-predicted the critical powers by as much as 55% at the lower pressures (1.7 MPa); however, at higher pressures the accuracy of RAMONA for SVEA-64 fuels is within 20%.

RAMONA was not benchmarked for SVEA-96 or SVEA-100 fuel designs because the FRIGG tests for these designs were performed under conditions were the instability threshold was not reached. CENPD-294-P[1] reports that analysis of the available data from those tests indicated that the SVEA-96 and 100 designs appear to be thermal-hydraulically more stable than SVEA-64.

RAMONA has not been benchmarked with direct channel-decay-ratio measurements but, as described above, ABB-CE chose to use the instability-threshold powers. This an acceptable benchmark option and, given those results, we judge that the RAMONA accuracy for channel decay ratio calculations should be within ±0.2 under realistic BWR operating conditions.

The reactivity instability capabilities of RAMONA were benchmarked against measurements in WNP-2, Leibstadt, Oskarsham-3, Ringhals-1, and Forsmark-3. Report CENPD-294-P[1] contains benchmarks for a total of 69 core-wide stability test results and 3 out-of-phase conditions. Data for core-wide stability tests is more readily available than for out-of-phase conditions because tests in reactors typically measure the core-wide decay ratio, and out-of-phase data is only obtained when a limit cycle is developed. The results of these benchmark analysis indicate that RAMONA can predict core-wide and out-of-phase decay ratios within ±0.2 under realistic BWR operating conditions (the average error in the benchmarks is 0.08).

We conclude from this benchmark exercise that, with the proper input, RAMONA can consistently estimate the channel, core-wide, and out-of-phase decay ratios to within ±0.2 under realistic operating conditions.

RAMONA LIMITATIONS

RAMONA qualification has been limited to expected conditions in operating reactors. Thus, RAMONA can only be used reliably to estimate decay ratios under those conditions. In particular, RAMONA has not been qualified for extremely abnormal conditions, such as LOCAs or very-low-water-level conditions that may result during anticipated transients without scram.

RAMONA has not been qualified by ABB-CE for stability analyses in new passive reactors such as SBWR, where components like the extended upper plenum riser may affect the reactor stability. To use RAMONA under conditions other than its qualification base, ABB-CE would have to justify its applicability.

CONCLUSIONS AND TECHNICAL RECOMMENDATIONS

Based on the present review, we conclude that the RAMONA system as proposed by ABB-CE provides a reasonably accurate estimation of the stability of (1) the channel thermal-hydraulics mode, (2) the fundamental or core-wide coupled neutronics thermal-hydraulics mode, and (3) the out-of-phase or regional coupled neutronics thermal-hydraulics mode. We also conclude that RAMONA decay ratio calculations are accurate to within ±0.2 in a decay ratio range from 0 to 1.1 for all three modes.

Based on its technical merits, we recommend that the RAMONA system be an acceptable methodology for best-estimate stability prediction of operating boiling water reactors. Note that for the purposes of this review, the RAMONA system is the combination of a computer code (Scandpower's version of RAMONA-3B) and a set of input generating procedures described in ABB-CE documents CENPD-294-P[1] and CENPD-295-P.[2]

As with all stability codes, input preparation is the major source of error; therefore, to maintain the ±0.2 accuracy, any new calculations must use procedures similar to those used in the qualification report. To insure that input errors do not compromise the accuracy of the calculations, we recommend that best estimate RAMONA calculations follow the input-generating procedures described in CENPD-294-P[1] and CENPD-295-P.[2] The RAMONA input must then be reviewed to guarantee that the following minimum requirements are satisfied:

(1) Each thermal-hydraulic region in the core (i.e., channel) model must be divided in a minimum of 24 axial nodes.

(2) The core model must be divided into a series of radial nodes (i.e., thermal-hydraulic regions or channels) in such a manner that

  (a) No single region can be associated with more than 20% of the total core power generation. This requirement guarantees a good description of the radial power shape, especially for the high power channels.

  (b) The core model must include a minimum of three regions for every bundle type that accounts for significant power generation.

(c) The model must include a hot-channel for each significant bundle type with the actual conditions of the hot channel.

(3) Each of the thermal-hydraulic regions must have its own axial power shape to account for 3-D power distributions. For example, high power channels are likely to have bottom peaked shapes.

(4) For out-of-phase calculations, a full-core representation is recommended. The minimum configuration, however, is two basic "symmetry units" (e.g., in a core with quarter core symmetry, RAMONA must model at least half the core).

(5) Care must be taken in the selection of the perturbation used to excite each instability mode. A review must be performed to confirm that the perturbation actually excites each mode of oscillation (e.g., a perturbation along a symmetry line will not excite an out-of-phase oscillation).

In addition to best-estimate calculations, our technical review indicates that the RAMONA system represents an adequate methodology to estimate *Exclusion Region* boundaries to be used with the so-called *BWR Stability Long Term Solutions*. Note that *Exclusion Region* calculations are not best-estimate and they require a well-defined input preparation procedure that has been specified by the Boiling Water Reactor Owners' Group (BWROG) and reviewed by the Nuclear Regulatory Commission. The so-called BWROG procedures are defined in NEDO-31960[4,5] "BWR Owner's Group Long Term Stability Solutions Licensing Methodology." In *Exclusion Region* applications using the RAMONA code, care must be taken to ensure that the axial and radial power shapes resulting from RAMONA's 3-D calculation represent as accurately as possible the power shapes prescribed in NEDO-31960.[4,5] Any departure from the established BWROG procedures to calculate *Exclusion Regions* must be justified.

REFERENCES

1. CENPD-294-P, *Thermal-Hydraulic Stability Methods for Boiling Water Reactors*, ABB Combustion Engineering Nuclear Operations Report, May 1995.

2. CENPD-295-P, *Thermal-Hydraulic Stability Methodology for Boiling Water Reactors*, ABB Combustion Engineering Nuclear Operations Report, September 1995.

3. ORNL/NRC/LTR-95/34, *Review of ABB Combustion Engineering Thermal-Hydraulic Stability Methodology for Boiling Water Reactors*, Jose March-Leuba, Oak Ridge National Laboratory Technical Evaluation Report, October 1995.

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

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

CENPD-294-NP-A REPORT

Part II

Body of Report

ABB Combustion Engineering Nuclear Operations

TABLE OF CONT NTS

1 INTRODUCTION

This report describes the RAMONA-3 code used for thermal hydraulic stability analysis and demonstrates the code qualification for performing stability analyses in support of plant specific applications such as reload licensing evaluations.

The ABB methodologies for performing reload stability analysis using the RAMONA-3 code described in this document is presented in Reference 1.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

1 INTRODUCTION

This report describes the RAMONA-3 code used for thermal hydraulic stability analysis and demonstrates the code qualification for performing stability analyses in support of plant specific applications such as reload licensing evaluations.

The ABB methodologies for performing reload stability analysis using the RAMONA-3 code described in this document is presented in Reference 1.

2 SUMMARY AND CONCLUSION

This report demonstrates the ability of the RAMONA-3 code to predict the margin to the plant instability limit.

The features of the RAMONA-3 code that make it an excellent tool for stability calculations include:

(1) The ability to can represent the entire core configuration. This allows explicit modeling of the thermal-hydraulic, neutronic, and fuel thermo-dynamic behavior for each fuel assembly .

(2) The ability to perform a fully coupled three dimensional transient thermal-hydraulics and neutronics calculation.

(3) Accurate convergence of the RAMONA-3 time domain calculations, as shown by time step sensitivity studies.

Based in the information contained in this report, specific conclusions which can be made regarding the use of the RAMONA-3 stability methods are:

(1) RAMONA-3 accurately or conservatively predicts test loop measured thresholds of density wave oscillations for various fuel assembly configurations.

(2) RAMONA-3 can accurately predict the onset of global (core-wide) stability and azimuthal (regional) oscillations.

(3) RAMONA-3 can accurately predict core average power decay ratios for damped oscillations.

(4) RAMONA-3 can accurately predict core average power oscillations frequency for damped oscillations.

(5) Mixed core configurations can be explicitly simulated, eliminating modeling uncertainties due to lumping and grouping of fuel types and averaging of the three dimensional power distribution.

(6) RAMONA-3 comparisons with 70 plant data measurements show an overall prediction uncertainty of ±0.075 for relatively high decay ratios. The uncertainty increases to about ±0.15 for relatively low decay ratios, due to inaccuracies in determining decay ratios of well damped systems.

Therefore, the overall conclusion is that RAMONA-3 as described in this report, can be used to reliably predict the margin to the plant instability limit. Furthermore, the thorough benchmarking of the code provides good definition of the code uncertainties. The methodology for

applying the code and the application of the code uncertainties, for reload applications is discussed in Reference 1.

3 RAMONA-3 CODE DESCRIPTION

The RAMONA-3 computer code used in the ABB stability analysis is described in this section. Several versions of the RAMONA code are widely used in the nuclear industry and some are well documented in the open literature. This section briefly reviews the history and documentation of the RAMONA code and then describes the RAMONA-3 version used by ABB for stability evaluations. The ABB version of RAMONA-3 is described by discussing changes relative to a publicly released and documented version (Reference 2).

3.1 Background

RAMONA-3 originates from a development project at the nuclear research institutes of the Scandinavian countries (Kjeller, Riso, and Studsvik) and ABB Atom. The code development has been continued by Scandpower and Brookhaven National Laboratory (BNL), with significant funding from the U.S. NRC. The code has been selected by NRC as a reference 3D BWR transient code, and has been applied in special studies and in the verification of other transient methods. As consultants to the nuclear industry, Scandpower has applied and provided licensing rights to use the code to numerous BWR plants in Europe and the United States. ABB has been, and is, an active contributor to numerous development and application efforts with RAMONA-3.

Detailed documentation of basic methods, code features, and limitations can be found in Reference 2, which can be considered as a complete documentation of the code version released by BNL in 1983 (RAMONA-3B Mod 0 Cycle 4). Since publication of Reference 2, Scandpower has upgraded the code and added new features. This effort, new features, and code capabilities, as well as limitations, models used, and solution methods, are discussed below.

The version in use at ABB is the Scandpower version of RAMONA-3B. Unless, otherwise indicated, the term "RAMONA-3" in the discussion below refers to the Scandpower version of the code. ABB also uses the RAMONA-3 code described below for control rod drop accident (CRDA) licensing calculations. The ABB CRDA methodology using RAMONA-3 (Reference 3) is in the final stage of approval by the NRC.

3.2 Overview of Code

3.2.1 Code Capabilities

RAMONA-3 is a three-dimensional, transient, coupled neutronic and thermal-hydraulic code that explicitly models each fuel type in the reactor core. The code is designed to simulate local and global reactor

core transient responses without introducing significant spatial or temporal approximations.

RAMONA-3 is used for analysis of BWR operational transients for conditions ranging from cold standby to full power. The code capabilities include simulations of both global and azimuthal instabilities; transients with full, partial, or no scram; rod drop accidents; MSIV closure; turbine trip; etc., and simulation of any combination of plant control actions related to the main steam supply system.

RAMONA-3 simulates processes in the pressure vessel, one representative recirculation loop, and one representative steam line. Plant control and protection functions that are related to the main steam supply system are also modeled. RAMONA-3 does not model the containment building nor the balance of plant.

Table 3-1 provides a summary of the some of the transient applications for which RAMONA is used.

3.2.2 Neutron Kinetics and Power Generation

The core is divided into neutronic cells with a vertical stack of such cells constituting a neutronic channel. Each neutronic channel represents one fuel bundle. Cross sections are assigned to each neutronic cell depending on the fuel cell history prior to the transient (fuel composition, burnup, and void history).

The neutron kinetics model starts from the two-group, three-dimensional, time-dependent diffusion equations and reduces to the 1.5-group coarse mesh model by approximating the thermal leakage term. The boundary conditions at the core periphery are based on extrapolation lengths for the fast flux and albedos for the thermal flux. Six delayed neutron groups are used and the effective delayed neutron fractions are treated as nodal variables.

The power distribution is the sum of the prompt and delayed energy generation rates. The prompt generation rate is proportional to the fission rate, whereas the delayed generation rate accounts for the decay heat from the fission products and is calculated using the 1979 ANS Standard 5.1. The cross section are represented as functions of:

- exposure and void history
- coolant density
- fuel temperature
- control fraction

- xenon poisoning

An important capability of RAMONA-3 is in providing systematic procedures for collapsing the 3D core model into larger nodes or a 1D model, thereby reducing core modeling complexity and computing time. Another capability is the editing of the neutronics results to produce continually the various components of the total core reactivity (fuel and moderator temperature, void, control rods). This helps in the analysis of the core response to different perturbations and is valuable in supplying reactivity functions for use in point-kinetics scoping calculations.

3.2.3 Thermal Conduction

Thermal heat storage and conduction in the fuel rods are modeled using a discrete-parameter model for the pellet, gap, and cladding. Axial heat conduction is neglected. The fuel thermal conductivity and heat capacity, as well as gap conductance, are temperature dependent. Clad volumetric heat capacity and conductivity are constant parameters. Maximum fuel pin enthalpy in the core is calculated using node-specific local power peaking factors.

3.2.4 Thermal-Hydraulics

The thermal hydraulics modeling uses a multichannel representation of the core, lower plenum, downcomers, riser, upper plenum, steam separator, steam dome, and a steam line with SRVs, MSIV, and turbine stop and bypass valves. The core consists of parallel heated channels modeling the in-bundle flow region, and a flow channel representing the bypass flow region. The feedwater spargers are typically located in the upper part of the downcomer, and the jet pumps are in the lower part.

The vessel thermal hydraulics is based on a four-equation two-phase flow model, based in conservation of:

- vapor mass
- mixture mass
- mixture momentum
- mixture energy

A slip model is used to calculate the relative velocity between vapor and liquid phases. Nonequilibrium vapor generation and condensation are accounted for with the vapor phase at saturation and the liquid phase being either subcooling, saturated or superheated. Constitutive

equations include correlations for the single- and two-phase wall friction, form losses, and wall heat transfer including post-CHF.

The processes in the steam line are assumed adiabatic, and the code solves mass and momentum equations. Acoustic effects in the steam line produced by valve openings and closures are taken into account.

The transient boron concentration is computed from the boron mass balance equation in parallel with the liquid and vapor mass conservation equations. Boron is assumed to propagate with the liquid velocity. Boron stratification in the lower plenum is neglected.

3.2.5 Plant Control and Protection Systems

RAMONA-3 includes numerous models for system components, plant control systems, and plant protection systems. Major among them are:

- jet pumps
- recirculation pumps
- steam line with safety/relief, main steam isolation, bypass, and turbine stop valves
- safety injection systems (i.e., HPCI and RCIC)
- boron transport and standby liquid control system for boron injection
- reactor pressure controller
- feedwater flow controller
- reactor scram, main steam isolation valve closure, and trips
- IRM, LPRM, and APRM nuclear monitor detectors

3.2.6 Solution Methods

The solution of the conservation equations in RAMONA-3, begins by transforming all partial differential equations into a set of ordinary differential equations. Then RAMONA-3 calculates a steady-state solution. For the subsequent transient calculation, the following integration schemes are used,

- Gauss-Siedel iteration for the fast flux solution,
- explicit integration for the delayed neutron equations,
- an iterative predictor-corrector method for heat conduction,

- explicit first-order method for the vessel thermal-hydraulics, and optionally, higher order explicit and implicit integration of the momentum equations, and

- fourth-order Runge-Kutta method for steam line dynamics

The transient neutron kinetics and heat conduction equations are integrated with a master time step, and within this master integration step, several integration substeps are used for the thermal-hydraulics equations. Separate accuracy and stability criteria are applied to each of these substeps. Several minor substeps for the steam line dynamics are used within the thermal-hydraulic substep. Convergence tests on the fast neutron fluxes are performed, and the entire procedure is either accepted and the integration is advanced, or it is rejected and the entire step is repeated with a smaller master step size.

Two simplifications are introduced in the solution of the thermal-hydraulics equations, which reduce considerably the computational time without significant loss of accuracy. The first simplification consists of combining the mixture mass and energy equations, and integrating the resulting equation over the entire vessel. This gives a time-dependent equation for the average pressure vessel in terms of the vessel boundary conditions (i.e., feedwater and ECCS injection rates, and steam line flow), total vapor generation rate, and compressibility of the phases. This pressure is used to compute steam and water properties in the reactor vessel. An option in steady state allows calculation of the pressure distribution by solving the mixture momentum equation. In this case, steam and water properties are determined at the local pressure.

The second simplification is introduced by transforming the mixture momentum equations into closed-contour integral momentum equations by integration along each parallel channel and the bypass channel. The resulting integral momentum equations are solved, together with the mixture mass equation, for the two-phase flow field in the reactor vessel.

3.3 Description of Models

The RAMONA code utilized by ABB is an extension of the version released and documented by Brookhaven National Laboratory (BNL) in 1983 (Reference 2). Documentation of basic methods, code features and limitations are discussed in detail in Reference 2. Reference 2 provides a complete documentation of that code version and also describes some model qualification and application results.

Starting in 1984, Scandpower has been engaged in modifying and improving RAMONA. An initial code development program was conducted whose objectives were to:

- merge the BNL code version of RAMONA with Scandpower RAMONA-III programs
- introduce basic improvements in both modeling and user-oriented features
- write a new user's manual

Since the conclusion of the initial development program, more improvements have been introduced and new features have been added. At the same time, the code was continually assessed using experimental and plant data and applied to a number of transients; particularly, these past few years, to BWR stability measurements.

Table 3-2 lists the RAMONA-3 models and provides the corresponding reference describing each model. The following sections provide model features important for stability analysis that have not been previously documented in public literature.

3.4 New Code Features

The RAMONA-3 modifications relative to Reference 2 documentation are described in the subsection. The models are grouped by:

- Neutron Kinetics and Power Generation models,
- Thermal Conduction models,
- Thermal-Hydraulics models,
- Plant Control and Protection Systems models, and
- Solution Methods models,

These modifications also have been discussed during the NRC review of Reference 3.

3.4.1 Neutron Kinetics and Power Generation

3.4.1.1 Cross Section Model

In the BNL cross-section model, the cross sections are represented as polynomial functions in the instantaneous parameters (void, fuel temperature, and moderator temperature). No burnup dependency is included, but complete data sets are required for each material region in the core, even though the regions might be of the same fuel design and differ only in burnup. This necessitates a rather coarse core model with large nodes and data set generation explicitly for each core state. The BNL cross-section model is still available as an option in RAMONA-3.

The Scandpower cross section option, on the other hand, is designed to interface with static core simulators, such as, PRESTO or POLCA (Reference 4 and 15), which have been approved by NRC for static BWR core design analysis. Each node is a material region and is characterized by its fuel design type and its burnup value. The burnup distribution file for a specific core state is read from a burnup file generated by the nodal simulator. Data for each nuclear parameter set are separated into functions of burnup (exposure and void history) and instantaneous parameters (coolant density and fuel temperature). The burnup function needs only to be evaluated at the initialization of a RAMONA-3 calculation and its constant value is then carried along throughout the transient. The instantaneous parameter function is evaluated at each power-void iteration in steady state and at each time step in the dynamics.

3.4.1.2 Neutronics

Several modifications have been introduced in the neutronics model. They are summarized as:

(1) The PRESTO thermal flux model (Reference 4) has been introduced in RAMONA-3. This new model improves the accuracy, especially for high flux gradient cases.

(2) Two options existed for the xenon representation: neglecting the xenon effect or assuming the concentration in equilibrium with the steady-state local power. A new option has been added where the initial-state non-equilibrium xenon concentration is read from a distribution file.

(3) In addition to the burnup data required by the cross-section model, first guesses on flux, power, and void distributions can be provided.

(4) The effective delayed neutron fraction can be treated as a nodal variable function of fuel design and burnup state.

(5) Edits of total reactivity, from an inverse kinetics solution, and reactivity components (void, fuel temperature, moderator temperature, control rods) based on a first-order perturbation formulation, are provided.

(6) Models for detector response (i.e., IRM, LPRM, APRM) and associated scram functions have been introduced.

3.4.2 Thermal Conduction

The thermal conduction model now allows fuel rod performance parameters to be input for each fuel type being simulated.

3.4.3 Thermal-Hydraulics

Several modifications have been introduced in the thermal-hydraulics models. They are summarized as:

(1) A general recirculation pump model has been implemented with pump head and torque described by homologous curves in four quadrants.

(2) The code has been improved and made more robust during transients with flow reversal.

(3) For reactivity insertion accidents, the maximum fuel enthalpy is evaluated using both the adiabatic and non-adiabatic approaches.

(4) Optional models for two-phase flow multipliers have been introduced and slip correlations have been extended.

(5) The hydraulic and fuel pin models have been improved to allow for modeling of fuel assemblies with partial length rods. The hydraulic and fuel parameters can now be specified separately at each axial position in the channel.

3.4.4 Plant Control and Protection Systems

Several modifications have been introduced in the plant control and protection systems models. They are summarized as:

(1) A pressure controller model for ABB designed plants and a feedwater controller have been implemented.

(2) The code calculates a collapsed water level which is input to reactor protection systems level trips.

3.4.5 Solution Methods

The time integration of the hydraulics has been improved to optionally allow for higher order explicit methods as well as implicit integration of some of the equations.

There are now four hydraulics integration options: explicit, explicit with corrector method, explicit with higher order corrector method, and implicit integration of momentum equation. This implicit integration method speeds up the code significantly for slow transients, where the hydraulics computation time is limiting. The higher-order explicit method avoids numerical damping, and is used in stability analyses.

3.4.6 General Modifications

Numerous miscellaneous features have also been introduced in RAMONA-3. These new features facilitates effective use of the code. Some of these modifications are:

(1) The input data processing has been completely revised. Each card is identified by a unique 6-digit number and is structured logically into groups. Cards with any character other than the digit 1-9 in column 1 are treated as comment cards. All input data are read in list-directed free format.

(2) The thermal-hydraulics input has been simplified and the change of core geometry from one symmetry to another is streamlined.

(3) RAMONA-3 is able to directly read RETRAN-02 cross section files. When the RETRAN cross-section option is selected, RAMONA will automatically modify the core model to 1D geometry. With this option, a 3D model defined by a complete input deck can be switched to a 1D case.

(4) The code can generate a hot channel data file for use in a thermal-hydraulics code for hot channel calculations.

(5) Almost all internal scratch files have been eliminated, all common blocks are specified with INCLUDE statements, and dimensions are declared with PARAMETER statements.

(6) There is an on-line screen output which allows the user to monitor the calculation while running interactively.

3.5 RAMONA-3 General Code Qualification

This section summarizes some examples from the general qualification data base for the RAMONA-3 code, both for separate effect tests and BWR plant tests. The intent of this section is to indicate the general reliability of the code to predict the response of a broad range of events and to describe the broad experience in applying the code. Qualification of RAMONA-3 for density-wave oscillations is presented in Section 5. Qualification of the code for core stability applications is presented in Section 6.

The information contained in this section is derived from two publications: a report published by Brookhaven National Lab. (Reference 2) and a documentation and qualification report from Scandpower (Reference 5). The assessment matrix includes comparisons of the code results with data from separate effect tests

(e.g., FRIGG experiments), as well as plant transients (e.g., Peach Bottom Turbine Trip Test Series (Reference 6)).

3.5.1 Separate Effects Testing

3.5.1.1 Thermal-Hydraulics Model

RAMONA-3 hydrodynamic model was tested against data from the FRIGG experiments for various operating conditions and fuel bundle types. Void distribution, pressure drop distribution, transfer functions between different variables, dynamic response to power ramps, and stability limits were tested for various flow, pressure, subcooling, and power shape conditions (Reference 7).

For example, void measurements for the electrically heated 6x6 BWR fuel bundles and 8x8 fuel bundles in the FRIGG loop have been compared to the hydrodynamic model implemented in PRESTO and RAMONA-3. The FRIGG stability limit tests for several fuel test bundles are discussed in detail in Section 5.

3.5.1.2 Steam Line Model

Assessment of steam line dynamics has been done by comparing code results with analytical solutions (Reference 8) and with Peach Bottom 2 turbine trip test results (Reference 6). Results of the assessment show close agreement between the solution based on the RAMONA-3 steam line model and an analytical solution. Comparison with measurements taken during Peach Bottom 2 Turbine test show that computed pressures at the location of the MSIVs agree well with the measured mean pressure (discounting the high-frequency oscillations in the measured data which were caused by the sensor connecting lines).

3.5.1.3 Critical Heat Flux Correlation

RAMONA-3 contains a publicly available CHF correlation package based in part on the Condie-Bengston correlation. This correlation has been qualified against experiments, as reported in Reference 12.

In addition, for licensing calculations, an NRC approved CHF or CPR correlation may be implemented for a specific application.

3.5.1.4 Neutronics Model

The RAMONA-3 model is, for steady state calculations, essentially the same as the model in PRESTO. It has been extended to kinetics by including time-dependent terms and the delayed neutron precursor equation. PRESTO is Scandpower's steady-state 3D simulator, and is now in use at a number of power plants in Europe and the US.

PRESTO has been fully qualified against fine mesh diffusion solutions of numerical benchmarks, and against gamma scan measurements and operating plant data. PRESTO has been approved by NRC for use as a three-dimensional steady-state BWR simulator (Reference 4).

3.5.2 BWR Transient Tests

3.5.2.1 Peach Bottom Turbine Tests

Data from the Peach Bottom 2 turbine trip tests have been used extensively for assessment of RAMONA-3. Qualification programs have been conducted independently by both BNL and Scandpower.

BNL reported their results in Reference 2. A relatively coarse mesh neutronics model was used together with boundary conditions imposed on the turbine and bypass valves action.

The Scandpower results are reported in Reference 6. The Scandpower model used the recorded steam dome pressure as a boundary condition, (i.e., no steam line was included) the pressure was treated nonuniformly within the vessel, and the neutronics model was more detailed with a different nuclear data base from that of the BNL model.

3.5.2.2 Scram Tests in Gundremmingen A

An early application of RAMONA-3 was the calculation of scram tests in Gundremmingen A. These tests, sponsored by the Institute of Energy Technology, included a full scram as well as single rod insertions.

3.5.2.3 SPERT Experiments

RAMONA-3 has been used to calculate reactivity-insertion accidents and has been benchmarked against six of the SPERT-III E-core power excursion experiments (Reference 9). A number of tests were conducted at cold startup initial conditions. These conditions are also applicable to the startup conditions of BWRs.

Results from the qualification work are reported in Reference 3.

3.5.3 Predictive Calculations

3.5.3.1 ATWS Calculations

BNL has reported (Reference 13) results from a generic study on ATWS initiated by a closure of all MSIVs. Several scenarios were investigated to provide better understanding of mitigative effects of operator actions and to help in developing adequate emergency procedure guidelines. This application demonstrates the ability of

RAMONA-3 to simulate long transients with extreme and reverse flow conditions, and very low water level.

3.5.3.2 Control Rod Drop Analyses

RAMONA-3 is well-suited for reactivity insertion accidents because of the detailed three-dimensional neutronics modeling. Also, the code accounts for hydraulic feedback effects which, although of second order, may be significant. Numerous studies have performed (see for example Reference 2, 10, and 11).

ABB has submitted topical reports to the NRC documenting the methodology and methods it uses in calculations of rod drop accidents with RAMONA-3 (Reference 3).

3.5.3.3 Other Applications

Many plant transients have been simulated with RAMONA-3. Table 3-3 gives a list of some of the applications of the code to date.

TABLE 3-1

RAMONA-3 TRANSIENTS APPLICATION MATRIX

Classification	Event
Instabilities	• Global (Core-wide) Oscillations • Azimuthal (Regional) Oscillations
Transients with Reactivity Changes	• Control Rod Drop Accident • Full or Partial Scram • Gradual or Sudden Rod Withdrawal or Insertion • Boron Injection • Moderator Temperature and/or Density Change (e.g., Void Collapse Due to Pressure Rise on the Core)
Transients with Pressure Changes	• Turbine Stop Valve Closure (Load Rejection, Turbine Trip, etc.) • Main Steamline Isolation Valve Closure • Failure of Pressure Regulator • Faulty Operation of Safety Relief Valves
Transients with Coolant Mass Inventory and/or Mass Flow Rate Changes	• Changes in Feedwater Mass Flow • Inadvertent Actuation of Safety Injection Systems • Recirculation Pumps Trip
Transients with Coolant Temperature Changes	• Loss of Feedwater Heater • Failure of Feedwater Flow Controller

TABLE 3-2

SUMMARY OF RAMONA-3 CODE DESCRIPTION

Model Description	Reference
<u>Neutron Kinetics and Power Generation</u>	
Diffusion Equations	Reference 2, Section 2.3
Void Feedback	Reference 2, Section 2.4.1 This document, Section 3.4.1
Moderator Temperature Feedback	Reference 2, Section 2.4.2 This document, Section 3.4.1
Doppler Feedback	Reference 2, Section 2.4.3 This document, Section 3.4.1
Boron Reactivity Feedback	Reference 2, Section 2.4.4
Xenon Reactivity Feedback	Reference 2, Section 2.4.5 This document, Section 3.4.1
Transverse Leakage Correction	Reference 2, Section 2.4.6 This document, Section 3.4.1
Linear Extrapolation Lengths	Reference 2, Section 2.5.1
Thermal to Fast Current Ratios	Reference 2, Section 2.5.2
Cref Coefficient	Reference 2, Section 2.5.3
Two Group Reflector Parameters	Reference 2, Section 2.5.4
Average Neutron Velocities	Reference 2, Section 2.6.1
Delayed Neutron Parameters	Reference 2, Section 2.6.2 This document, Section 3.4.1
Prompt Fission Heat	Reference 2, Section 2.7.1
Decay Heat	Reference 2, Section 2.7.2
Heat Deposition	Reference 2, Section 2.7.3

TABLE 3-2 (CONTINUED)

SUMMARY OF RAMONA-3 CODE DESCRIPTION

<u>Thermal Conduction</u>	
Structural Components	Reference 2, Section 3.3.1
Fuel Elements	Reference 2, Section 3.3.2
Thermal Conduction Equations	Reference 2, Section 3.4.1.1 This document, Section 3.4.2
Boundary Conditions for Fuel Elements	Reference 2, Section 3.4.1.2
Thermal Conduction Constitutive Relations	Reference 2, Section 3.4.1.3
<u>Thermal-Hydraulics</u>	
Field Equations of Coolant	Reference 2, Section 4.4.1
Heat Transfer from Cladding Surface to Coolant	Reference 2, Section 4.4.2.1
Wall Shear and Form Losses	Reference 2, Section 4.4.2.2 This document, Section 3.4.3
Vapor Generation Rate	Reference 2, Section 4.4.2.3
Slip Correlations	Reference 2, Section 4.4.2.4 This document, Section 3.4.3
Thermophysical Properties of Coolant	Reference 2, Section 4.4.2.5
Transport Properties of Coolant	Reference 2, Section 4.4.2.6
System Pressure	Reference 2, Section 4.4.3
Coolant Circulation	Reference 2, Section 4.4.4.1
Recirculation Flow	Reference 2, Section 4.4.4.2
Jet Pump Model	Reference 2, Section 4.4.4.3
Recirculation Pump Model	Reference 2, Section 4.4.4.4 This document, Section 3.4.3
Reactor Components Model	Reference 2, Section 4.4.5 This document, Section 3.4.3
Boron Transport	Reference 2, Section 4.4.6

TABLE 3-2 (CONTINUED)

SUMMARY OF RAMONA-3 CODE DESCRIPTION

<u>Plant Control and Protection Systems</u>	
System Pressure Regulation	Reference 2, Section 5.2.1 This document, Section 3.4.4
Safety and Relief Valves	Reference 2, Section 5.2.2
Main Steam Isolation Valves	Reference 2, Section 5.2.3
Plant Protection System	Reference 2, Section 5.2.4
High Pressure Coolant Injection System	Reference 2, Section 5.2.5
Reactor Core Isolation Cooling System	Reference 2, Section 5.2.6
Feedwater Control System	This document, Section 3.4.4
<u>Solution Methods</u>	
Neutron Kinetics and Power Generation	Reference 2, Section 6.2
Thermal Conduction	Reference 2, Section 6.3
Coolant Thermal-Hydraulics	Reference 2, Section 6.4
State Equations	Reference 2, Section 6.5
Steady-State Conditions	Reference 2, Section 6.6
Transient Conditions	Reference 2, Section 6.7 This document, Section 3.4.5

TABLE 3-3

RAMONA-3 APPLICATIONS

Plant	Application
Browns Ferry	ATWS
Brunsbüttel	Control rod drop
	Continuous rod withdrawal
Brunswick-1	Overpressurization
	Control rod drop
Caorso	Stability analysis
Dodewaard	Stability analysis
	Control rod drop
	Inadvertent control rod withdrawal
Forsmark-1	Stability analysis
Forsmark-3	Stability analysis
Fukushima-III	Overpressurization
Gundremmingen	Control rod drop
	Startup transient
	Cold water injection
Laguna Verde Unit 1	Load rejection without bypass
	Turbine trip without bypass
	Feedwater controller failure
	MSIV closure at 100% power
	Load rejection at 100% power
	Loss of feedwater heating at 50% power
	Stability analysis
LaSalle	Stability analysis
Leibstadt	Reactivity insertion accidents
	Stability analysis
	MSIV closure ATWS
Mühleberg	Reactivity insertion accidents
	Loss of feedwater flow transient
Oskarshamn 3	Stability analysis
Oyster Creek	ATWS
Peach Bottom	Turbine trip tests
Ringhals-1	Stability analysis
TVO-I	Stability analysis
WNP-2	Stability analysis

4 STABILITY ANALYSIS PROCESS

This section describes the general ABB process for performing a stability calculation with RAMONA-3. The general process is used in the plant measurement qualification calculations presented in Section 6 and is provided primarily to clarify the discussion in Section 6. As discussed in Reference 1, a similar process is used for reload design and licensing analyses.

The general steps for performing a stability calculation with RAMONA-3 can be summarized as follows:

<u>Set up a plant vessel model</u>

[Proprietary Information Deleted]

<u>Set up the core configuration</u>

[Proprietary Information Deleted]

<u>Set up the core statepoint</u>

[Proprietary Information Deleted]

<u>Establish the initial conditions for the statepoint of interest</u>

[Proprietary Information Deleted]

<u>Perform the core stability evaluation</u>

[Proprietary Information Deleted]

The flow-loop model is very similar to those of standard one-dimensional system analysis models (e.g., RETRAN, BISON). Channel geometry, core layout, and control rod pattern are generally based on three-dimensional steady state simulator models (e.g., SIMULATE, POLCA, PRESTO). Steam lines, pressure controller or feedwater flow controller models are generally not included because of their small impact on the stability margin evaluation.

The nuclear data library is based on data generated with a two-dimensional lattice physics code (e.g., RECORD, PHOENIX, CASMO). These data are converted to the appropriate format in order to generate polynomial fits appropriate for use in RAMONA-3. The code POLGEN is used for the generation of the polynomial fits as functions of exposure, void history, coolant density, and fuel temperature. Control rod and xenon effects are treated as correction terms to the cross sections.

Generally, burnup, void history, and xenon distribution files are read directly from the three-dimensional steady-state core simulator distribution file.

As discussed above, steady state results are used to confirm that the RAMONA-3 model is consistent with three-dimensional steady state simulator, thermal-hydraulic system analysis, and plant core supervision results.

The dynamic behavior of the reactor is simulated by RAMONA-3 starting from the appropriate initial conditions. The response of the system is analyzed by imposing a suitable perturbation (typically control rod movement) and observing the evolution in time, after the effect of the initial perturbation has died out. This evolution determines the rate at which oscillations will decay or grow, thereby determining the decay ratio. The perturbation is typically imposed for a period of about one second

The perturbation has no influence on the asymptotic behavior of the system. That is, after the effects of the perturbation have disappeared, the system will oscillate at its proper frequency and will exhibit decaying oscillations if the system is damped or growing oscillations if the system in undamped. For cases which are unstable (i.e. a decay ratio greater than one), the effect of the initial perturbation is to decrease the computation time required for the oscillations to develop. In this case oscillations will grow even in the absence of an initial control rod perturbation.

5 RAMONA-3 DENSITY-WAVE OSCILLATION QUALIFICATION

The capability of the RAMONA-3 code to predict the margin to the occurrence of plant oscillations is assessed in this report by comparisons of RAMONA-3 predictions to reactor power oscillation events observed in operating BWRs as well as with experimental loop test data. Qualification against plant measurements provides a measure of the code capability to capture the effects of density wave oscillations combined with the effects of power feedback and fuel thermodynamics. Qualification against plant measurements is discussed in Section 6. Qualification against loop test data measures the capability to predict onset of density-wave oscillations for a single channel (i.e. fuel assembly) operating at constant surface heat flux. The loop test comparisons, therefore, represent a separate effects test in the sense that the effects of power feedback and fuel thermodynamics are not included. This section presents a comparison of the RAMONA-3 with loop test data.

ABB FRIGG loop measurements of density-wave oscillations have been performed on a variety of assembly designs. Measurements have been conducted on early 36-rod Boiling Heavy Water Reactor (BHWR) fuel designs, open-lattice (as opposed to water cross design) 6x6 and 8x8 BWR designs, and on the SVEA, or water cross, design. Specifically, this section provides a discussion of comparisons between loop measurements and RAMONA-3 predictions for the following assembly types:

(1) Test sections simulating a Marviken BHWR fuel assembly.

(2) Thirty-six rod test assemblies simulating an ABB open-lattice 8x8 assembly.

(3) Full-scale test assembly simulating an ABB open-lattice 8x8 assembly.

(4) Test assemblies simulating the ABB 8x8 (SVEA-64) and 10x10 (SVEA-96/100) water cross design fuel.

Table 5-1 summarizes the lattice types for which FRIGG loop density-wave oscillation measurements are available for comparison with RAMONA-3.

[Proprietary Information Deleted]

5.1 Test Sections Simulating a Marviken BHWR Fuel Assembly

The Marviken reactor was a small Swedish BHWR that operated in the 1960's and early 1970's. Data obtained for two Marviken reactor test assemblies, designated FT36B and FT36C, were compared with

RAMONA-3 predictions. The FT36B and FT36C test sections are full scale simulations of a Marviken BHWR fuel assembly. The test assemblies have 36 heater rods in a circular geometry. The radial power distribution is nonuniform in both assemblies. The FT36B axial power distribution was uniform, and the FT36C axial power distribution was a 1.19-peaked chopped cosine. The only other difference between the two test sections was a somewhat higher exit pressure drop in the FT36C assembly due to additional instrumentation and cables. The FT36 data are publicly available and have been used extensively for thermal-hydraulic code verification.

The test loop and assemblies were instrumented to measure axial and radial void distributions and pressure drops through the different parts of the loop as well as coolant temperature and flow rates. The loop was operated under natural circulation flow conditions. Measurements were performed at pressures of 3.0, 5.0 and 7.0 MPa for various subcoolings, inlet throttling, and bundle power levels.

The calculated flow rates for various assembly powers reflect the accuracy of the RAMONA-3 hydraulic model. [Proprietary Information Deleted]

5.2 Thirty-Six Rod Simulation of ABB Open-Lattice 8x8 Assembly

The ABB FRIGG loop OF36 test assembly simulated the central part of an ABB open-lattice 8x8 BWR fuel assembly. "Open-lattice" refers to arrays of fuel and water rods without a water cross such as that in ABB SVEA design. The test assembly consisted of 36 rods in a 6x6 square lattice. The radial and axial power distributions in this test assembly were uniform. Tests were conducted for both natural and forced circulation.

The test loop and assemblies were instrumented to measure axial and radial void distributions, coolant temperature, flow rates, and pressure drops. Both forced and natural-circulation mass flow rate measurements were performed. Power thresholds for oscillations were performed for forced-circulation conditions at pressures ranging from 31 to 69 bar and inlet subcoolings ranging from 9 to 29 °C.

The calculated flow rates for various assembly powers reflect the accuracy of the RAMONA-3 hydraulic model. [Proprietary Information Deleted]

5.3 Full Scale Simulation of ABB Open-Lattice 8x8 Assembly

The OF64 test section is a full-scale simulation of an ABB open lattice 8x8 fuel assembly. The radial power distribution is nonuniform, and the axial power distribution has a 1.55-peaked top-skewed shape. The

test loop was instrumented to measure voids and pressure drops along the test section as well as flow rates and temperatures.

Both forced and natural-circulation flow rate measurements were performed. In addition, void fraction and pressure drop measurements were performed in the test section. Power threshold measurements for oscillations were performed for forced-circulation conditions at pressures ranging from 29 to 68 bar and inlet subcoolings ranging from 9 to 28 °C.

The calculated flow rates for various assembly powers reflect the accuracy of the RAMONA-3 hydraulic model. [Proprietary Information Deleted]

5.4 Simulated ABB 8x8 (SVEA-64) and 10x10 (SVEA-96/100) SVEA Fuel

Descriptions of the ABB SVEA (i.e. water cross) fuel designs are provided in Section 2 of Reference 19. As discussed in Reference 19, the SVEA-64 fuel assembly consists of four 4x4 subbundles. The SVEA-96 and SVEA-100 fuel assemblies represent an evolution of the SVEA-64 assembly containing 5x5-1 or 5x5 subbundles.

[Proprietary Information Deleted]

5.5 Conclusions

The following conclusions regarding the capability of RAMONA-3 to predict density-wave oscillations are based on the comparisons between FRIGG loop measurements and RAMONA-3 predictions in Sections 5.1 through 5.4:

[Proprietary Information Deleted]

TABLE 5-1

FRIGG LOOP DENSITY WAVE STABILITY EXPERIMENTS

Test Assembly	Description
FT36	BHWR (Marviken) 36-rod geometry
OF36	BWR 36-rod geometry
OF64	ABB Atom 8x8 fuel geometry
SVEA-64	Complete assembly simulated with and without communication slots
SVEA-100	One subassembly
SVEA-96	One subassembly

Figures 5.1-1 through 5.1-6 Proprietary Information Deleted

Figure 5.2-1 through 5.2-4 Proprietary Information Deleted

Figures 5.3-1 through 5.3-4 Proprietary Information Deleted

Figure 5.4-1 Proprietary Information Deleted

6 RAMONA-3 CORE STABILITY QUALIFICATION

To date, a large database has been assembled for qualification of RAMONA-3 for stability evaluations. The core stability measurement database includes numerous tests conducted in European plants and stability events world wide. The data cover a large range of flow and power conditions, and a variety of fuel designs and power distributions.

The RAMONA-3 qualification effort against core stability measurements is an ongoing process as new data become available. This section provides an assessment of the reliability of RAMONA-3 to simulate BWR instabilities based on comparisons of code predictions against data from stability tests and actual plant events using the RAMONA-USA1S ABB stability analysis process described in Section 4. The qualification cases presented here are:

- WNP-2 Cycle 8 Stability Event,
- Leibstadt Cycle 7 and 10 Stability Test,
- Oskarshamn 3 Cycle 7, 9 and 10 Stability Tests,
- Ringhals 1 Cycle 14, 15, 16, and 17 Stability Tests, and
- Forsmark 3 Cycle 8, 9, and 10 Stability Tests.

These qualification cases comprise 70 individual RAMONA-3 simulations of measured stability data.

6.1 WNP-2 Cycle 8 Stability Event

6.1.1 Plant Description

Washington Nuclear Plant Unit 2 (WNP-2) is a General Electric designed BWR located on the Columbia River near Richland, Washington, USA. The principal plant characteristics are listed in Table 6.1-1. The plant original design power was uprated five percent in 1995, at the beginning of Cycle 11.

A stability event occurred at WNP-2 in 1992, during operation of Cycle 8. This event was simulated with RAMONA-3. The core fuel composition for Cycle 8 is shown in Table 6.1-2. The core consisted primarily of open lattice ANF 8x8-2 and water box ANF 9x9-9X fuel designs, with the 8x8 fuel being the dominant design.

6.1.2 Event Description

On August 15th, 1992, during Cycle 8 operation in WNP-2 a stability event occurred. The oscillations occurred during a startup after repairs to a valve in the drywell were complete. The event resulted in

global limit cycle oscillations which were promptly terminated by shutdown of the plant. Details of the event are provided in Reference 14.

Event Sequence

The oscillations began after closure of the flow control valve (FCV) in recirculation loop "A". Closure of the FCV is required before shifting the pump speed from low to high. Before closing the valve, a series of control rod adjustments were conducted from time 18:38 on August 14 to time 02:45 the following day.

Reactor power oscillations started as the FCV was closing at about 02:58:18 while power decreased from 36.4% to 33.5% and flow from 30.5% to 26%. All LPRM signals were in-phase, and thus the oscillations were of the global type. The oscillations grew until they reached a limit cycle. APRM peak-to-peak oscillation was about 77% of actual average power during the event, or about 26% of rated core power, before a scram was initiated. From Reference 14, the estimated decay ratio during the oscillation growth period was 1.06 and the frequency 0.5 Hz (see Table 6.1-4).

6.1.3 Model Description

Two core models were used in the analysis of the WNP-2 Cycle 8 stability event: a full core and a quarter core model.

The full core has a total of 764 bundles. In both the neutronic and hydraulic descriptions, all channels are treated explicitly. [Proprietary Information Deleted]

6.1.4 Event Simulation

During the reactor startup, computer calculations (POWERPLEX MON runs) were used by the operators to monitor reactor power distribution and control rod positions. A series of edits from the POWERPLEX MON data were provided by the utility at a few time points before the event and the scram.

RAMONA-3 stability calculations of WNP-2 were performed for the conditions given by the POWERPLEX MON data at 03:00:14. These conditions are summarized in Table 6.1-3. The radial core power distribution at these initial conditions is shown in Figure 6.1-1.

[Proprietary Information Deleted]

TABLE 6.1-1

WNP-2 PLANT CHARACTERISTICS

Parameter	Value (Original)	(Uprate)
Plant Manufacturer	General Electric	
Product Line	BWR/5	
Commercial Operation Date	1984	1995
Rated Thermal Power	3323 MWt	3486 MWt
Rated Core Flow	13,667 kg/sec (108.5 Mlb/hr)	
Number Fuel Assemblies	764	
Recirculation System	20 Jet Pumps	
Core Power Density	49.2 kW/liter	51.6 kW/liter

TABLE 6.1-2

PROPRIETARY INFORMATION DELETED

TABLE 6.1-3

PROPRIETARY INFORMATION DELETED

TABLE 6.1-4

WNP-2 CYCLE 8 MEASUREMENT RESULTS

Case	Decay Ratio	Frequency (Hz)	Mode	Comments
Event	1.00	0.50	global	Growth rate 1.06 to limit Cycle Oscillations

TABLE 6.1-5

WNP-2 CYCLE 8 SIMULATION RESULTS

Case	Decay Ratio		Frequency (Hz)		Mode	
	Measured	Calculated	Measured	Calculated	Measured	Calculated
base	1.00	[Proprietary Information Deleted]	0.50	[Proprietary Information Deleted]	global	[Proprietary Information Deleted]

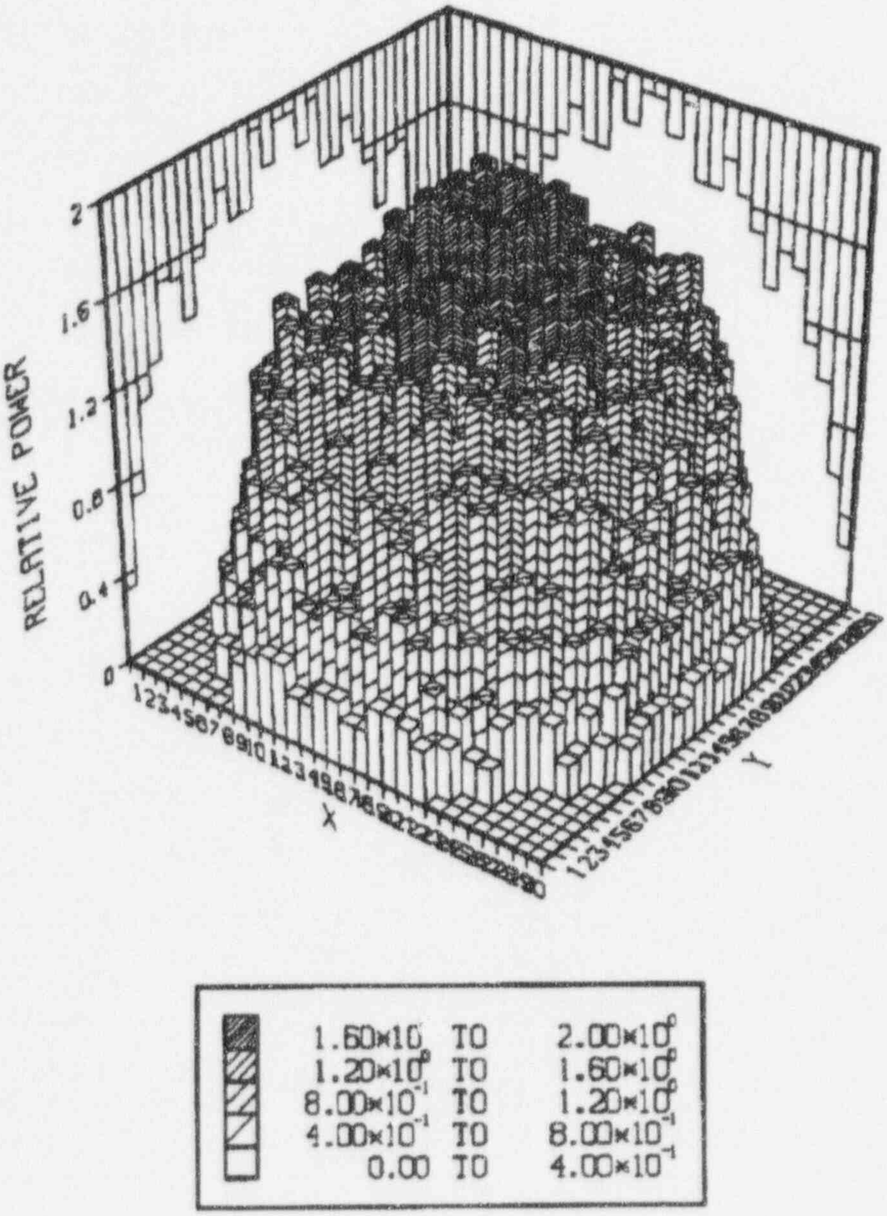

Figure 6.1-1 WNP-2 RAMONA-3 Calculated Core Power Distribution at Initiation of Instability Event

Figure 6.1-2 through 6.1-7 Proprietary Information Deleted

6.2 Leibstadt Cycle 7 and 10 Stability Tests

6.2.1 Plant Description

The Leibstadt (KKL) nuclear power plant is a General Electric designed BWR located in the northern part of Switzerland near the town of Leibstadt. The principal plant characteristics are listed in Table 6.2-1.

Core stability tests were conducted in the KKL reactor at the beginning of the seventh and tenth cycles of operation. The core composition for Cycle 7 is shown in Table 6.2-2. The core consisted primarily of GE 8x8 open lattice fuel. The core composition for Cycle 10 is shown in Table 6.2-3. The core dominant fuel was ABB SVEA-96 (10x10) watercross fuel with the remainder consisted of GE 8x8 fuel.

6.2.2 Leibstadt Cycle 7 Tests

6.2.2.1 Cycle 7 Test Description

A core stability test was conducted in the KKL reactor in the beginning of the seventh operating season during the power ascension phase (Reference 18). [Proprietary Information Deleted]

Testing Sequence

[Proprietary Information Deleted]

Test Data

[Proprietary Information Deleted]

6.2.2.2 Cycle 7 Model Description

A RAMONA-3 model was developed for the Leibstadt plant. Both plant specific data and Cycle 7 specific data are described in this section.

[Proprietary Information Deleted]

6.2.2.3 Cycle 7 Test Simulations

[Proprietary Information Deleted]

6.2.3 Leibstadt Cycle 10 Tests

6.2.3.1 Cycle 10 Test Description

A core stability test was conducted in the Leibstadt BWR at the beginning of its tenth operating cycle. When the test was performed, the reactor had been operating at full power for a few days, after a restart upon completion of its annual refueling and maintenance shutdown period. The core composition for Cycle 10 is shown in Table 6.2-3. [Proprietary Information Deleted]

Testing Sequence

[Proprietary Information Deleted]

Test Data

[Proprietary Information Deleted]

6.2.3.2 Cycle 10 Model Description

A RAMONA-3 cycle specific model was developed for KKL Cycle 10. The plant specific Leibstadt model is identical to that used for Cycle 7 (see Section 6.2.2.2). The core model was specifically developed for the Cycle 10 composition (Table 6.2-3) using a similar procedure as described previously.

6.3.3.3 Cycle 10 Test Simulation

[Proprietary Information Deleted]

TABLE 6.2-1

LEIBSTADT PLANT CHARACTERISTICS

Parameter	Value (Original)	Value (Uprate)
Plant Manufacturer	General Electric	
Product Line	BWR/6	
Commercial Operation Date	1984	1996
Rated Thermal Power	3138 MWt	3515 MWt
Rated Core Flow	11151 kg/sec (88.5 Mlb/hr)	
Number Fuel Assemblies	648	
Recirculation System	20 Jet Pumps	
Core Power Density	54.7 kW/liter	61.3 kW/liter

TABLES 6.2-2 THROUGH 6.2-9

PROPRIETARY INFORMATION DELETED

Figures 6.2-1 through 6.2-5 Proprietary Information Deleted

6.3 Oskarshamn 3 Cycle 7, 9, and 10 Stability Tests

6.3.1 Plant Description

Oskarshamn 3 is an ABB Atom designed BWR located on the east coast of Sweden. Oskarshamn 3 went into commercial operation in 1985. The original rated core thermal power was 3020 MW_{th}. In 1990, it was uprated to operate at 109.3 per cent of rated power (3300 MW_{th}) while maintaining 3020 MW_{th} as 100% of core power. Plant characteristics for Oskarshamn 3 are summarized in Table 6.3-1.

Stability tests were performed at Oskarshamn 3 during Cycles 7, 9, and 10. The core composition for Cycle 7, 9, and 10 are shown in Table 6.3-2, 6.3-3, and 6.3-4 respectively.

6.3.2 Oskarshamn 3 Cycle 7 Tests

6.3.2.1 Cycle 7 Test Description

[Proprietary Information Deleted]

6.3.2.2 Cycle 7 Model Description

[Proprietary Information Deleted]

6.3.2.3 Cycle 7 Test Simulation

[Proprietary Information Deleted]

6.3.3 Oskarshamn 3 Cycle 9 Tests

6.3.3.1 Cycle 9 Test Description

[Proprietary Information Deleted]

6.3.3.2 Cycle 9 Model Description

[Proprietary Information Deleted]

6.3.3.3 Cycle 9 Test Simulation

[Proprietary Information Deleted]

6.3.4 Oskarshamn 3 Cycle 10 Tests

6.3.4.1 Tests Description

[Proprietary Information Deleted]

6.3.4.2 Model Description

[Proprietary Information Deleted]

6.3.4.3 Tests Simulation

[Proprietary Information Deleted]

TABLE 6.3-1

OSKARSHAMN 3 PLANT CHARACTERISTICS

Parameter	Value (Original)	(Uprate)
Plant Manufacturer	ABB Atom	
Product Line	BWR 75	
Commercial Operation Date	1985	1990
Rated Thermal Power	3020 MWt	3300 MWt
Rated Core Flow	13100 kg/sec (104.0 Mlb/hr)	
Number Fuel Assemblies	700	
Recirculation System	8 Internal Pumps	
Core Power Density	49.0 kW/liter	53.5 kW/liter

TABLES 6.3-2 THROUGH 6.3-13

PROPRIETARY INFORMATION DELETED

Figure 6.3-1 Proprietary Information Deleted

6.4 Ringhals 1 Cycle 14, 15, 16, and 17 Stability Tests

6.4.1 Plant Description

Ringhals 1 is an ABB Atom designed BWR located on the west coast of Sweden. Ringhals 1 is a external recirculation loop design BWR that went into commercial operation in 1977. In 1989, the plant was uprated from its original rated power of 2270 MW_{th} to 2500 MW_{th}. Plant characteristics for Ringhals 1 are summarized in Table 6.4-1.

Vattenfall AB, the Ringhals 1 plant utility, performed extensive stability measurements during Cycles 14, 15, 16, and 17. For all these cycles the core was comprised almost exclusively of the ABB SVEA-64 water cross fuel. The core compositions for these four cycles are shown in Tables 6.4-2 through 6.4-5.

Ringhals 1 tests for these cycles have been selected by the nuclear committee of the OECD/NEA to provide a benchmark problem for stability calculations. The purpose of the benchmark is to validate the predictive capability of various codes in the OECD member countries. Test measurements from Cycles 14 and 15 were used to benchmark the stability analysis model for Ringhals 1. Blind calculations were then performed for the specified test conditions of Cycles 16 and 17. Results using RAMONA-3 are summarized below.

6.4.2 Ringhals 1 Cycle 14 Tests

6.4.2.1 Cycle 14 Test Description

Noise measurements were performed by Vattenfall's plant personnel at the beginning of Cycle 14 during power ascension after refueling in September 1990 (Reference 20). The recordings were made at points arranged in a grid layout in the high power/low flow region of the operating range.

The most important process parameters (APRM, LPRM on levels 2 and 4 from the top, core flow, steam flow, feedwater flow and temperature, reactor pressure, reactor water level) were measured with a data scanner at ten operating states, as described in Table 6.4-6.

The recording order follows the alphabetic identifiers shown in Table 6.4-6. No oscillations were observed in the readings for the operational points 1 to 6. After measurement 6, the core flow was reduced to minimum flow and control rods were withdrawn. At about 72% power (Rec. 9) the LPRM instruments started to oscillate out-of-phase. The control rod pattern chosen for this measurement point created a lower power region dividing the core in to two high power regions favoring regional oscillations. After about 15 seconds the oscillations reached a limit cycle with an amplitude of 30% peak-to-peak. Oscillations were

suppressed by gradual insertion of control rods. The oscillations stopped after two steps in the control rod insertion sequence. The flow was then slightly increased from the flow limit line, and the last two points (Rec. 8 and 10) were measured under stable conditions.

The recorded data were analyzed several times by Vattenfall and ABB to evaluate the quality of the measured signals, resonance frequencies, decay ratios and phase shift between LPRM detector signals (see Section 7.1). The result of the latest data analyses are shown in Table 6.4-7.

6.4.2.2 Cycle 14 Model Description

[Proprietary Information Deleted]

6.4.2.3 Cycle 14 Test Simulation

The BWR Stability Benchmark calculations specified by the Nuclear Committee of the OECD/NEA were jointly performed by ABB and Scandpower. [Proprietary Information Deleted]

6.4.3 Ringhals 1 Cycle 15 Tests

6.4.3.1 Cycle 15 Test Description

Stability measurements were made a the beginning of Cycle 15 (September 10-11, 1991) following a procedure similar to that performed for Cycle 14 (see Section (6.4.2.1). The test state points used for stability evaluation are shown in Table 6.4-9. The deduced stability parameters for each test point are given in Table 6.4-10. In all measurements the in-phase (global) oscillation mode dominated.

6.4.3.2 Cycle 15 Model Description

A RAMONA-3 cycle specific model was developed for Ringhals 1 Cycle 15. The plant specific Ringhals 1 model is identical to that used for Cycle 14 (see Section 6.4.2.2). The core model was specifically developed for the Cycle 15 composition (Table 6.4-3) using the procedure described previously.

6.4.3.3 Cycle 15 Test Simulation

[Proprietary Information Deleted]

6.4.4 Ringhals 1 Cycle 16 Tests

6.4.4.1 Cycle 16 Test Description

Stability measurements again were made at the beginning of Cycle 16 (February 11-27, 1993) and also about six months later (July 24, 1993) during cycle operation. The measurements followed the same procedure as that performed for Cycle 14 and 15 (see Section (6.4.2.1). The test state points used for stability evaluation are shown in Table 6.4-12. The deduced stability parameters for each test point are given in Table 6.4-13. In all measurements the in-phase (global) oscillation mode dominated.

6.4.4.2 Cycle 16 Model Description

A RAMONA-3 cycle specific model was developed for Ringhals 1 Cycle 16. The plant specific Ringhals 1 model is identical to that used for Cycle 14 and 15 (see Section 6.4.2.2). The core model was specifically developed for the Cycle 16 composition (Table 6.4-4) using the procedure as described previously.

6.4.4.3 Cycle 16 Test Simulations

[Proprietary Information Deleted]

6.4.5 Ringhals 1 Cycle 17 Tests

6.4.5.1 Cycle 17 Tests Description

Stability measurements again were made at the beginning of Cycle 17 (November 17, 1993). The measurements followed the same procedure as that performed for Cycle 14, 15, and 16 (see Section (6.4.2.1). The test state points used for stability evaluation are shown in Table 6.4-15. The deduced stability parameters for each test point are given in Table 6.4-16. In all measurements the in-phase (global) oscillation mode dominated.

6.4.5.2 Cycle 17 Model Description

A RAMONA-3 cycle specific model was developed for Ringhals 1 Cycle 17. The plant specific Ringhals 1 model is identical to that used for Cycle 14, 15, and 16 (see Section 6.4.2.2). The core model was specifically developed for the Cycle 17 composition (Table 6.4-5) using the procedure as described previously.

6.4.5.3 Cycle 17 Test Simulation

[Proprietary Information Deleted]

TABLE 6.4-1

RINGHALS 1 PLANT CHARACTERISTICS

Parameter	**Value (Original)**	**(Uprate)**
Plant Manufacturer	ABB Atom	
Product Line	External Pump Design	
Commercial Operation Date	1977	1989
Rated Thermal Power	2270 MWt	2500 MWt
Rated Core Flow	9400 kg/sec (74.6 Mlb/hr)	
Number Fuel Assemblies	648	
Recirculation System	6 External Pumps	
Core Power Density	40.8 kW/liter	44.9 kW/liter

TABLES 6.4-2 THROUGH 6.4-5

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TABLE 6.4-6

RINGHALS 1 CYCLE 14 TEST CONDITIONS

Case	Power (%)	Core flow (kg/sec)	Lower Plenum Temp (C)
1 (A)	65.0	4105	266
3 (C)	65.0	3666	263
4 (D)	70.0	3657	261
5 (E)	70.0	3868	263
6 (F)	70.2	4126	264
8 (H)	75.1	3884	261
9 (G)	72.6	3694	260
10 (I)	77.7	4104	262

TABLE 6.4-7 THROUGH 6.4-17

PROPRIETARY INFORMATION DELETED

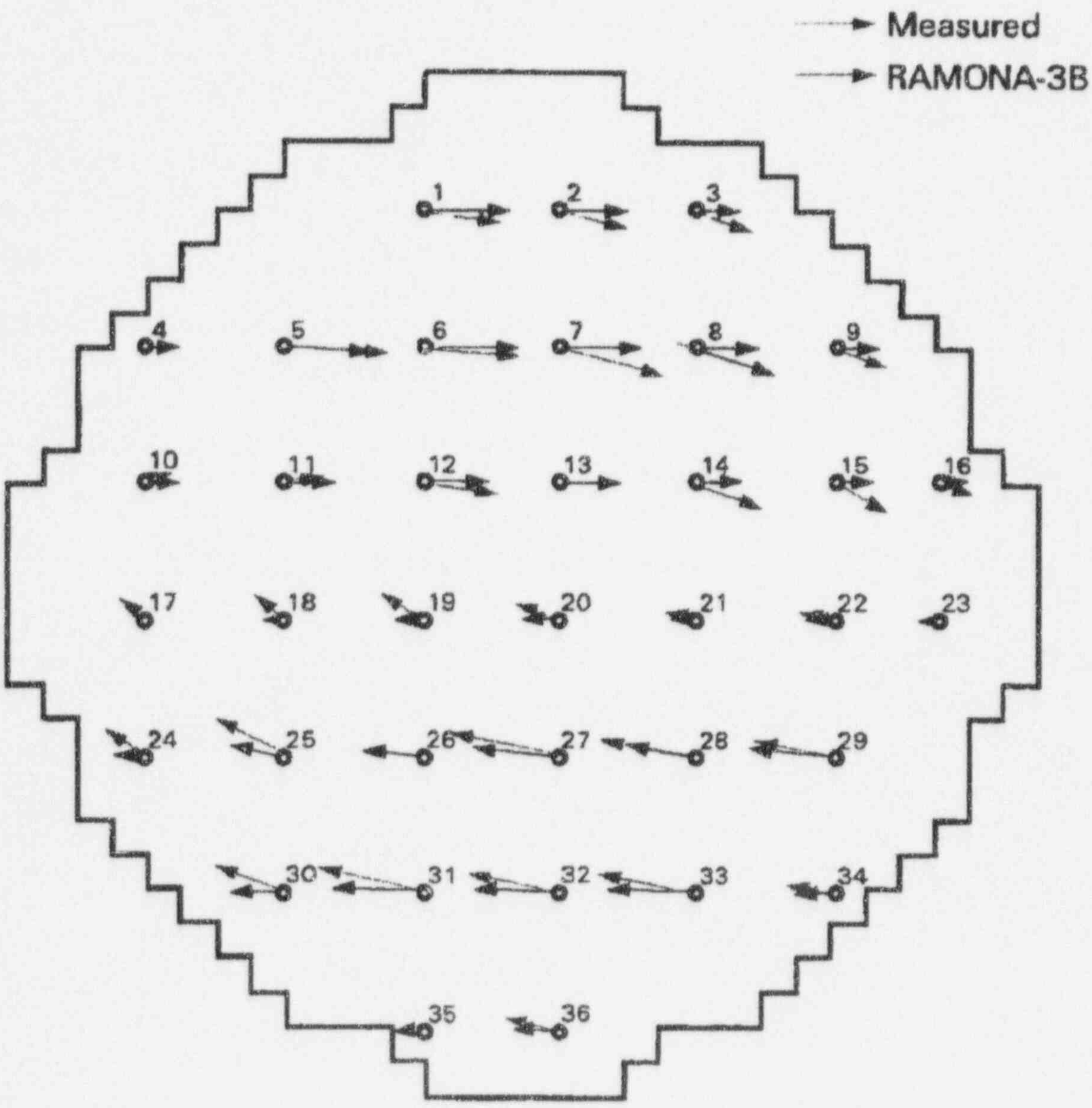

Figure 6.4-1 Comparison of Ringhals 1, Cycle 14, Recording 9 RAMONA-3 LPRM Readings Amplitude and Phases with Measurements

Figure 6.4-2 through 6.4-3 Proprietary Information Deleted

6.5 Forsmark 3 Cycles 8, 9, and 10 Stability Tests

6.5.1 Plant Description

Forsmark 3 is an ABB Atom designed BWR located on the east coast of Sweden north of Stockholm. Forsmark 3 went into commercial operation in 1985. The rated core thermal power is 3020 MW_{th}. Forsmark 3 has basically the same plant design as Oskarshamn 3. Similar to Oskarshamn 3, in 1990, it was uprated to operate at 109.3 percent of rated power (3300 MW_{th}). The original rating of 3020 MW_{th} maintained as 100 percent power following the uprate. Plant characteristics for Forsmark 3 are summarized in Table 6.5-1.

Stability tests were performed during Cycles 8, 9 and 10. The core composition for Cycle 8, 9, and 10 are shown in Table 6.5-2, 6.5-3, and 6.5-4 respectively. The core content has gradually changed from a SVEA-64 into a SVEA-100 dominated core. A small number of 8x8 bundles from previous cycles are still present.

6.5.2 Forsmark 3 Cycle 8 Tests

6.5.2.1 Cycle 8 Test Description

[Proprietary Information Deleted]

6.5.2.2 Cycle 8 Model Description

[Proprietary Information Deleted]

6.5.2.3 Cycle 8 Test Simulation

[Proprietary Information Deleted]

6.5.3 Forsmark 3 Cycle 9 Tests

6.5.3.1 Cycle 9 Test Description

[Proprietary Information Deleted]

6.5.3.2 Cycle 9 Model Description

[Proprietary Information Deleted]

6.5.3.3 Cycle 9 Test Simulation

[Proprietary Information Deleted]

6.5.4 Forsmark 3 Cycle 10 Tests

6.5.4.1 Cycle 10 Test Description

[Proprietary Information Deleted]

6.5.4.2 Cycle 10 Model Description

[Proprietary Information Deleted]

6.5.4.3 Cycle 10 Test Simulation

[Proprietary Information Deleted]

TABLE 6.5-1

FORSMARK 3 PLANT CHARACTERISTICS

Parameter	Value	
	(Original)	(Uprate)
Plant Manufacturer	ABB Atom	
Product Line	BWR 75	
Commercial Operation Date	1985	1990
Rated Thermal Power	3020 MWt	3300 MWt
Rated Core Flow	13100 kg/sec (104.0 Mlb/hr)	
Number Fuel Assemblies	700	
Recirculation System	8 Internal Pumps	
Core Power Density	49.0 kW/liter	53.5 kW/liter

TABLE 6.5-2 THROUGH 6.5-13

PROPRIETARY INFORMATION DELETED

Figure 6.5-1 Proprietary Information Deleted

7 RAMONA-3 CORE STABILITY UNCERTAINTY

To estimate the uncertainty associated with determining the accuracy of decay ratio predictions using RAMONA-3, two sources of uncertainty must be considered in examining the benchmark results:

- Measured Decay Ratio Uncertainties, and
- Simulation Decay Ratio Uncertainties.

Measurement uncertainties are associated with the accuracy to which a decay ratio can be deduced from the APRM or LPRM signals. Contributing factors are:

- sample rate and length of the recording,
- relative magnitude of the signal to background noise,
- accuracy of the method used to transform time signals into decay ratios and oscillation frequencies.

The RAMONA-3 simulation uncertainties can be attributed to the following:

- accuracy to which the initial conditions are known,
- degree to which the plant operating conditions are constant,
- accuracy of the RAMONA-3 model in simulating the plant,
- accuracy in deducing a decay ratio from the simulations results.

The above uncertainty elements are discussed in more detail below. Section 7.1 discusses measurement uncertainty. Section 7.2 discusses simulation uncertainties and associated studies performed to understand specific stability predictions.

7.1 Measurement Uncertainty

7.1.1 Data Reduction

The concept of the decay ratio is often used to measure the stability of BWRs. The decay ratio is the ratio between two consecutive maxima of the impulse response. For a second order system, this ratio is constant for any two consecutive maxima. For higher-order systems, the impulse response is formed by contributions of all the poles and the ratio between consecutive peaks is not constant. It converges though to an asymptotic value associated with the least stable pair of poles (Reference 21).

In the data reduction of the plant measurements, two methods have been commonly used:

- the autocovariance function (Reference 22)
- parameter identification methods based on Auto-Regressive (AR) modeling of the neutron noise (Reference 23).

The autocovariance function of a signal from an oscillatory system has similar properties regarding decay ratio and resonance frequency as the impulse response. That is, the "asymptotic" decay ratio from the autocovariance function is used to quantify the stability of the system, and the "apparent" decay ratio is not a good indicator of the system's stability margin.

Parameter identification methods applied to the neutron detector signals has been used almost exclusively for the measurements described in this report. The method consists in approximating the signal by a model of the form:

$$y(t) + a_1 y(t-1) + a_2 y(t-2) + ... + a_{na} y(t-na) =$$

$$b_1 u(t) + b_2 u(t-1) + . . . + b_{nb} u(t-nb+1) +$$

$$c_0 e(t) + c_1 e(t-1) + . . . + c_{nc} e(t-nc) \qquad (7\text{-}1)$$

where

u is the input signal to the system

y is the measured output signal

e is the white noise

t is the discrete time values from the measurement.

The identification process consists in determining the coefficients a, b, and c given a selected model order. Once the coefficients are determined, the stability characteristics of the system can be readily derived, either from the impulse response of the model or preferably, from the model coefficients directly. The model given above is called an auto-regressive moving average (ARMA) model.

7.1.2 Generic Measurement Uncertainty Expression

The uncertainty associated with the measured decay ratio depends on the sampling time, the signal-to-noise ratio, and the stability margin of the system. For a given recording quality, the uncertainty in deducing a decay ratio value from Equation 7.1 contains:

- the uncertainty associated with the choice of model order of the ARMA model
- the uncertainty due to the conversion of the ARMA model into a decay ratio value

[Proprietary Information Deleted]

7.1.3 Evaluation Differences

Ringhals 1 stability measurements for Cycle 14 and 15 (see Sections 6.4.2 and 6.4.3) have been evaluated independently by ABB and by the utility company. Figure 7.1-2 shows results from two different evaluations for the 14 measurement points versus the average of the two evaluations. The figure also shows the generic uncertainty expression, Equation 7-2.

[Proprietary Information Deleted]

7.2 Simulation Uncertainty

Simulation uncertainty includes:

- uncertainty of the RAMONA-3 code and plant model,
- uncertainty in the plant state, and
- uncertainty in the reduction of the simulation results.

[Proprietary Information Deleted]

7.2.1 Qualification Data Base Studies

The stability data base used in qualification of RAMONA-3 spans a wide range of conditions:

- plant designs (GE BWR/5, GE BWR/6, ABB External Pump ABB BWR-75),
- core power (~ 1100 to 2350 MW),
- core flows (~ 3200 to 9850 MW),
- fuel designs (e.g., open lattice 8x8, water rod 8x8, waterbox 9x9, watercross 8x8, watercross 10x10)
- combinations between fuel designs (see fuel composition Tables in Section 6)

- core power distributions (see discussions in Section 6)

Figure 7.2-4, 7.2-5, and 7.2-6 show, respectively, the core flow and core power, and maximum core nodal power peaking dependence of the data base as a function of the measured decay ratio.

Figure 7.2-7 shows the dimensionless parameter:

$$PPF*P/F*\Delta H = \frac{(\text{Max. Nodal Peaking})*(\text{Core Power})}{(\text{Core Flow})*(\text{Inlet Subcooling})}$$

as a function of the measured decay ratio. For a constant power distribution, the PPF*P/F*ΔH parameter is a relative measure of the core stability.

[Proprietary Information Deleted]

7.2.2 WNP-2 Data Studies

Section 6.1.1 described RAMONA-3 simulations of the WNP-2 stability event. A finding by the NRC Augument Inspection Team (AIT) for the event were that a mixed core was the least stable configuration compared to core of all 8x8 geometry fuel or an all 9x9-9X geometry fuel (Reference 14). In the AIT analysis (with LAPUR) the core power distribution was assumed unchanged for each sensitivity case. This observation was tested with RAMONA-3 model of WNP-2. The reference case was rerun replacing all 9x9-9X bundles by 8x8 bundles in one case, and all 8x8 fuel by 9x9-9X fuel in a second case. [Proprietary Information Deleted]

7.2.3 Leibstadt Data Studies

[Proprietary Information Deleted]

7.2.4 Cycle to Cycle Data Studies

The RAMONA-3 qualification data base contains a number of data measurements following similar test procedures, but performed for difference core fuel compositions and configurations. Specifically RAMONA-3 simulations have been made for:

- Oskarshamn 3 – Cycle 7, 9, and 10,
- Ringhals 1 – Cycle 14, 15, 16, and 17, and
- Forsmark 3 – Cycle 8, 9, and 10.

[Proprietary Information Deleted] Included in the results are several examples of pure predictions:

- Ringhals 1 – Cycle 16 and 17, the decay ratio was measured, but not known by ABB at the time of the simulation,
- Oskarshamn 3 – Cycle 9, the decay ratio was measured, but not known by ABB at the time of the simulation, and
- Forsmark 3 – Cycle 9, the test prediction was done before the measurement took place, hence included uncertainties associated with the accuracy with which the predetermined operating point was obtained.

For each plant, the follow on cycle predictions have the same degree of accuracy as the initial cycle results. The observation further support the ability of RAMONA-3 to calculate the stability margin for different fuel and core designs.

TABLE 7.2-1

WNP-2 CYCLE 8 INFLUENCE OF CORE LOADING ON STABILITY

Core Loading	Decay Ratio	Frequency (Hz)	Decay Ratio Change (%)	
	RAMONA	RAMONA	RAMONA	LAPUR (Ref. 14)
Mixed	[*]	[*]	-	-
All 9x9-9X	[*]	[*]	[*]	-10%
All 8x8	[*]	[*]	[*]	-20%

* Proprietary Information Deleted

TABLE 7.2-2

PROPRIETARY INFORMATION DELETED

Figures 7.1-1 and 7.1-2 Proprietary Information Deleted

Figures 7.2-1 through 7.2-8 Proprietary Information Deleted

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