ABB Combustion Engineering Nuclear Operations

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CENPD-295-NP-A

Thermal-Hydraulic Stability Methodology for Boiling Water Reactors

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July 1996

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CENPD-295-NP-A REPORT

Part I

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

ABB Combustion Engineering Nuclear Operations

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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-295-P: THERMAL HYDRAULIC STABILITY METHODOLOGY FOR BOILING WATER REACTORS (TAC NO. M93648)

Dear Mr. Ebeling-Koning:

The staff has reviewed the subject report submitted by ABB Combustion Engineering Nuclear Fuel by letter of September 12, 1995. This report provides the description of the general methodology for performing the stability analysis evaluation for reload fuel applications using NRC approved stability analysis codes for boiling water reactors. The staff has found that 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-295-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-295-P. In accordance with procedures established in NUREG-0390, the NRC requests that ABB/CE publish accepted versions of this 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 their 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-295-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-295-P
"THERMAL-HYDRAULIC STABILITY METHODOLOGY FOR BOILING WATER REACTORS"
ABB COMBUSTION ENGINEERING NUCLEAR FUEL

1.0 INTRODUCTION

By letter dated September 12, 1995, as supplemented by letter dated November 2, 1995, Asea Brown Boveri Combustion Engineering, Inc. (ABB/CE) submitted a licensing topical report CENPD-295-P (Ref. 1) and its Revision 1 (Ref. 3) for NRC review and acceptance for referencing in future licensing actions. This licensing topical report describes the general methodology for performing stability analysis evaluations for reload fuel applications. This licensing report 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-294-P (Ref.2), which was submitted in May 1995, is the subject of a separate safety evaluation report.

Topical report CENPD-295-P describes the proposed stability analysis methodology. The topical report also establishes a process to identify the limiting plant conditions to be evaluated and to prepare the input parameters to perform the analysis. It also establishes a process to relate the calculated limits of acceptable stability performance to a domain of acceptable plant operation.

The staff was supported in its 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-95/34, Revision 1, 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 conclusions stated in the staff's Safety Evaluation Report (SER) for the ABB/CE topical report CENPD-294-P are applicable to this review.

2. Any departure from the established Boiling Water Reactor Owners' Group (BWROG) procedures (Refs. 4 & 5) to calculate Exclusion Regions must be justified.

3. The stability methodology described in CENPD-295-P is based primarily on RAMONA stability calculations, but it also includes options to use NUFREQ-

NPW (Ref 6), experimental-loop instability limit measurements, in-plant channel flow measurements, and in-plant core stability measurements. Since CENPD-295-P only develops and documents in detail the RAMONA option, the use of any other option in CENPD-295-P (with the exception of NUFREQ-NPW, which is already licensed for limited purposes) will require a separate review.

4. The acceptance criteria for RAMONA-3 code stated in Reference 3 are acceptable. They are:

   (1) Core-wide decay ratio calculations are set to a calculated decay ratio of 0.8 (i.e., expected error including input preparation is ±0.2).

   (2) Channel thermal-hydraulic decay ratio calculations are set to a calculated decay ratio of 0.8 (i.e., expected error including input preparation is ±0.2).

   (3) Out-of-phase instability-threshold power calculations are set to either:

      (a) The actual threshold power for out-of-phase instabilities calculated by RAMONA minus an uncertainty margin that is calculated as the power required to reduce by 0.2 the core-wide decay ratio under those operating conditions, or

      (b) the power at which the core-wide decay ratio is 1.0 (i.e., 20% higher than the core-wide acceptance criteria) if out-of-phase instabilities are not observed following an appropriate out-of-phase perturbation.

4.0 REFERENCES

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

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

3. CENPD-295-P, Revision 1, *Thermal-Hydraulic Stability Methodology for Boiling Water Reactors*, November 2, 1995.

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

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

6. RPA-90-91-P-A (proprietary) , RPA-90-91-NP-A (nonproprietary) *NUFREQ-NPW: An ABB Atom Computer Code for Core Stability Analysis of Boiling Water Reactors*, ABB Report, December 1991.

ENCLOS 'RE 2

ORNL/NRC/LTR-95/34
Revision 1

Contract Program:	**Technical Support for the Reactor Systems Branch (L1697/P2)**
Subject of Document:	**Review of ABB Combustion Engineering Thermal-Hydraulic Stability Methodology for Boiling Water Reactors**
Type of Document:	**Technical Evaluation Report**
Auth r:	**José March-Leuba**
Date of Document:	**January, 1996**
NRC Monitor:	**T. L. Huang, Office of Nuclear Reactor Regulation**

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

INTRODUCTION

This report documents our review of the overall thermal-hydraulic stability methodology 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-295-P,[1] which describes ABB-CE's methodology for BWR stability evaluations, and the revised pages[2] of CENPD-295-P (Revision 1) that were submitted on November 2, 1995. This methodology is based primarily on RAMONA[3] stability calculations. An accompanying report, CENPD-294-P[3] describes the validation of the RAMONA code for stability calculations, and it is the subject of a separate technical evaluation report[4] (TER).

In topical report number CENPD-295-P,[1] ABB-CE describes their methodology to evaluate the stability of reload cores. This methodology is based primarily on RAMONA stability calculations, but it also includes options to use NUFREQ-NPW,[5] experimental-loop instability limit measurements, in-plant channel flow measurements, and in-plant core stability measurements. Since CENPD-295-P[1] only develops and documents in detail the RAMONA option, the use of any other option in CENPD-295-P[1] (with the exception of NUFREQ-NPW, which is already licensed for limited purposes) will require a separate review.

A related TER[4] documenting the review of CENPD-294-P[3] (ABB-CE's stability methods) concludes that the RAMONA system, when using input generated by the procedures described in CENPD-295-P[1] and CENPD-294-P[3], can estimate the decay ratio of a BWR operating under normal conditions for: (a) the fundamental (core-wide) and (b) the first azimuthal (out-of-phase, or regional) modes, and (c) the channel thermal-hydraulic stability. The methods review TER[4] also concludes that the accuracy of RAMONA-calculated decay ratios is 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.

Following the October 5, 1995, meeting between ABB, WNP-2 and the U.S. NRC, ABB submitted under reference 2 a number of revisions to the original topical report CENPD-295-P.[1] These revisions addressed the concerns raised during the meeting, which are documented in this report. The revised topical report is technically acceptable and satisfies the technical requirements documented in our TER.

The main conclusion from the present review is that RAMONA-based stability methodology is technically acceptable for best estimate stability calculations when a ± 0.2 margin is applied. We have some technical problems with the overall methodology as proposed in topical report CENPD-295-P.[1] The main source of concern is the number of possible options that ABB-CE is proposing but not developing yet. Other discrepancy is related to the stated accuracy of the RAMONA code; ABB-CE claims that the error in decay ratio calculations in ± 0.08. This number is based on the accuracy of their extensive benchmark calculations; however experience indicates that, when codes are used in "production mode", input preparation is not as careful as during benchmarks. Thus, we recommend that an error of ± 0.2 be used. A summary of our proposed technical recommendations and suggested exceptions to the topical report is contained in a recommendations and exceptions sections later on in this report.

SUMMARY DESCRIPTION OF THE METHODOLOGY

Topical report CENPD-295-P[1] describes the proposed stability analysis methodology. The main purpose of this report is to define acceptable stability analyses methods and their respective acceptance limits. The topical report also establishes a process to identify the limiting plant conditions to be evaluated and to prepare the input parameters to perform the analysis. CENPD-295-P[1] also establishes a process to relate the calculated limits of acceptable stability performance to a domain of acceptable plant operation.

The proposed stability analysis tools or methods are:

(1) RAMONA-3, a three-dimensional neutronic-thermal-hydraulics time domain code,

(2) NUFREQ-NPW, a frequency domain code that has been reviewed and approved with significant restrictions.

(3) Experimental-Loop Instability-Limit Measurements, which can demonstrate relative thermal-hydraulic stability performance of specific fuel designs.

(4) In-plant Channel Flow Measurements, which can be analyzed to determine the flow stability of a fuel element under the particular plant operating conditions.

(5) In-plant Core Stability Measurements, which can be analyzed to demonstrate relative stability of a specific reload core.

ABB-CE has only developed and documented in detail option number (1), the RAMONA option. The use of any of the other options will require a case-specific review.

A reload stability methodology must necessarily fall within one of the following five cases:

(1) Plant implementation of the BWR Owners' Group (BWROG) Interim Corrective Actions,

(2) BWROG Long Term Solution Option I-A,

(3) BWROG Long Term Solution Option I-D,

(4) BWROG Long Term Solution Option II, or

(5) BWROG Long Term Solution Option III.

Option Number (1) is an interim solution that will not be in effect when the other four options are fully implemented. The four Long Term Solutions have been reviewed and they all prescribe specific procedures for reload confirmation analysis. These Long Term Solution reload methodologies have been developed so that they are not vendor-specific; the only vendor requirement is the use of a qualified best-estimate stability code. Thus, a ABB-CE-specific review of these reload methodologies is not required.

One of ABB-CE options is to use RAMONA-3B as the best estimate code to implement the required reload confirmation calculations in the above BWROG Long Term Solutions. For this purpose, topical reports CENPD-295-P[1] and CENPD-294-P[3] describe the typical RAMONA input preparation procedure and the selection of the most limiting time in the cycle. The calculational procedure is as follows:

(1) Set up a steady-state operating condition and a RAMONA input deck following the procedures established in CENPD-295-P[1] and CENPD-294-P.[3]

(2) Establish an appropriate perturbation (e.g., a control rod movement in a non-symmetrical location) and observe the perturbation decay. The decay ratio and mode of oscillation can be measured directly from the resulting time traces.

(3) The channel thermal-hydraulic stability is computed by performing a thermal-hydraulics only calculation (i.e., without neutronics) and fixing a constant core pressure drop by increasing the resistance of the recirculation loop.

One limitation of time-domain codes such as RAMONA is that they can only measure the decay ratio of the dominant oscillation mode (i.e., core-wide or out-of-phase, but not both). In most stable cases, the dominant mode is the core-wide mode; thus, RAMONA cannot easily compute the out-of-phase decay ratio unless it is greater than 1.0 (i.e., an unstable condition). For this reason, the RAMONA acceptance criterion for out-of-phase stability is stated in terms of a critical instability-threshold power rather than in the more conventional decay ratio < 0.8 acceptance limit. This is an acceptable approach.

CONCLUSIONS AND TECHNICAL RECOMMENDATIONS

Based on the present review, we conclude that 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. 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-based option described in CENPD-295-P[1] be an acceptable methodology for best-estimate stability prediction of operating boiling water reactors.

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-295-P[1] and CENPD-294-P.[3] 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-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[5,6] "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.[5,6] Any departure from the established BWROG procedures to calculate *Exclusion Regions* must be justified.

EXCEPTIONS TO THE METHODOLOGY PROPOSED IN CENPD-295-P

Our technical review indicates that the following exceptions to the methodology proposed in CENPD-295-P[1] should be considered:

(1) The stability methodology described in CENPD-295-P[1] is based primarily on RAMONA stability calculations, but it also includes options to use NUFREQ-NPW,[4] experimental-loop instability limit measurements, in-plant channel flow measurements, and in-plant core stability measurements. Since CENPD-295-P[1] only develops and documents in detail the RAMONA option, the use of any other option in CENPD-295-P[1] (with the exception of NUFREQ-NPW, which is already licensed for limited purposes) will require a separate review.

(2) The acceptance criteria for RAMONA core-wide decay ratio calculations should be set to a calculated decay ratio of 0.8 (i.e., expected error including input preparation is ± 0.2). Note, this exception has been recognized by ABB and it has been corrected in Revision 1[2] of CENPD-295-P.

(3) The acceptance criteria for RAMONA channel thermal-hydraulic decay ratio calculations should be set to a calculated decay ratio of 0.8 (i.e., expected error including input preparation is ± 0.2). Note, this exception has been recognized by ABB and it has been corrected in Revision 1[2] of CENPD-295-P.

(4) The acceptance criteria for RAMONA out-of-phase instability-threshold power calculations should be set to either:

(a) The actual threshold power for out-of-phase instabilities calculated by RAMONA minus an uncertainty margin that is calculated as the power required to reduce by 0.2 the core-wide decay ratio under those operating conditions, or

(b) the power at which the core-wide decay ratio is 1.0 (i.e., 20% higher than the core-wide acceptance criteria) if out-of-phase instabilities are not observed following an appropriate out-of-phase perturbation.

Note, this exception has been recognized by ABB and it has been corrected in Revision 1[2] of CENPD-295-P.

REFERENCES

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

2. NFBWR-95-160, *Transmittal of Revised Pages to CENPD-295-P (Revision 1) "Thermal-Hydraulic Stability Methodology for Boiling Water Reactors"*, Combustion Engineering Nuclear Operations Report, November 2, 1995.

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

4. ORNL/NRC/LTR-95/33, *Review of ABB Combustion Engineering Thermal-Hydraulic Stability Methods for Boiling Water Reactors: the RAMONA Code*. Jose March-Leuba, Oak Ridge National Laboratory Technical Evaluation Report, October 1995.

5. RPA-90-91-P-A (proprietary) , RPA-90-91-NP-A (nonproprietary) *NUFREQ-NPW: An ABB Atom Computer Code for Core Stability Analysis of Boiling Water Reactors*, ABB Report, December 1991.

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

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

CENPD-295-NP-A REPORT

Part II

Body of Report

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1 INTRODUCTION

The ABB reload safety analysis process for a Boiling Water Reactor (BWR) is described in the "Reference Safety Analysis Report for BWR Reload Fuel" (Reference 1). One component of the reload safety analysis process is evaluation of the plant thermal-hydraulic stability performance. ABB has stability analysis methods for evaluating core and fuel stability performance (Reference 2 and 3). This document describes the ABB methodology for evaluating plant thermal-hydraulic stability performance for licensing safety evaluations of reload fuel applications and plant operation modifications.

<u>Stability Overview</u>

An integral element of Boiling Water Reactor design is safe, controlled operation of the reactor core. Sufficiently large enough power oscillations can lead to violation of the specified acceptable fuel design limits. The fact that boiling occurs in the reactor coolant creates the potential for thermal-hydraulically induced power oscillations. Power oscillations can result from thermal-hydraulic density-wave flow oscillations being created in individual fuel channels. The density-wave oscillations cause power oscillations due to:

- Void reactivity feedback on the nuclear power generation, and
- Changes in heat removal characteristics on the fuel rods.

Fuel channel density-wave oscillations are generally referred to as "channel stability". Coupled density-wave and nuclear power oscillations are referred to as "core stability". Core stability potentially can be:

- Core-wide oscillations – where the power throughout the core is oscillating in-phase for all the channels, or
- Regional oscillations – where the power in two opposing regions of the core oscillate out-of-phase.

Regional oscillation patterns observed in plants are associated with the first azimuthal harmonic of the three-dimensional steady-state core neutron flux. The mode is characterized by two halves of the core oscillating with a phase shift of 180 degrees from one another. The total core remains essentially constant, while the two halves of the core alternate increasing and decreasing inlet flows to maintain a constant pressure drop across the core.

The margin to an instability limit is usually quantified in terms of a "decay ratio", which is defined as the ratio of two consecutive peaks. A decay ratio less than one is associated with a damped, stable system, and a decay ratio greater than one implies growing oscillations, thus an unstable system.

Strictly speaking, channel stability performance is a not safety concern. However, it is a precursor to intermittent, reduced fuel cladding heat transfer and nuclear power oscillations in the fuel. Also, regional oscillations have been observed to occur only under combinations of conditions of relatively small channel and core stability margins. Hence, BWR stability performance margin historically has been evaluated using only the core-wide and channel decay ratios as measures.

Historical Background

In the U.S. BWR design acceptance criteria to preclude power oscillations induced by thermal-hydraulic stability are typically based on analytical calculations of core and channel stability decay ratios. Analytical methods are verified, as warranted, by test loop and in-plant measurements.

Evolving fuel designs and power oscillation events in several operating reactors world-wide, prompted the nuclear industry to improve the analysis and understanding of BWR stability performance. In 1982 and 1984, General Electric issued Service Information Letters pertaining to stability (Reference 4) and the NRC issued Generic Letter 86-02 (Reference 5). The GE Service Information Letters addressed specific plant conditions that strongly influence the thermal-hydraulic stability characteristics of a BWR. Following the power oscillation event at in LaSalle Unit 2 in March 1988, the BWR utilities and vendors in conjunction with the U.S. NRC embarked on an effort to develop generic long term solutions to the stability issue. Shortly after the LaSalle event, the BWR Owners Group recommended short term Interim Corrective Actions (ICAs), the NRC issued two Information Bulletins (Reference 6), and the BWR licensees implemented the recommended ICAs. Work supporting development of a long term solution continued with several updates to the original recommended ICAs (Reference 7 and 8) reflecting results of the ongoing research efforts and observations from the power oscillation event at Washington Nuclear Plant 2 in August 1992. Starting in 1992, the BWROG and participating utilities submitted several Licensing Topical Reports supporting the implementation of long term solutions in the operating BWRs (e.g., Reference 9 through 15). To date, the NRC has issued several Safety Evaluation Reports (e.g., Reference 15 through 19) on the submitted Reports and issued Generic Letter 94-02 (Reference 20) supporting final resolution of the outstanding stability issue.

In Europe, some BWR utilities, have augmented the approaches being adopted in the U.S. with frequent plant specific measurements. In some cases, the regulatory authorities' required plant acceptance criteria is based on measurements for the specific plant.

The extensive in-plant measurements performed by some European BWR utilities has supported the ability for ABB to:

- Develop well qualified stability analysis code methods, and
- Develop a reload analysis methodology for application in the U.S., that incorporates in-plant experience of the relative importance of the key parameters that affect thermal-hydraulic stability.

Report Structure

This document describes the ABB stability analysis methodology. A summary of the methodology is given in Section 2. Section 3 presents the stability evaluation design bases and acceptance criteria, and Section 4 summarizes the stability analysis tools. Then Section 5 describes the ABB reload analysis methodology including the general process of evaluating the plant stability boundaries for a reload, and the more specific process of evaluating a reload using the specific stability licensing basis adopted by the plant. Appendix A shows examples of reload stability evaluations using the ABB stability analysis methodology. Appendix B summarizes the specific acceptance limits for the RAMONA-3 code. Appendix C provides a detailed description of two RAMONA-3 code-wide stability calculations.

2 SUMMARY AND CONCLUSION

In the ABB reload safety analysis process (described in Reference 1), thermal-hydraulic stability safety evaluations are performed as required by the plant specific stability licensing bases. In general, stability performance is evaluated for each reload fuel applications or plant modification with the potential to significantly change the core nuclear or thermal-hydraulic performance characteristics. This report presents the ABB thermal-hydraulic stability methodology used in performing plant specific safety evaluations, as required by the specific plant licensing bases.

Plant specific stability licensing bases may be characterized as analysis methodology approved generically by the NRC, a generic methodology with additional plant specific licensing commitments, or a unique methodology for the specific plant.

In summary, the ABB thermal-hydraulic stability methodology:

(1) Establishes acceptance limits for demonstrating acceptable stability performance,

(2) Identifies the stability analysis methods that are used to demonstrate compliance with the acceptance limits,

(3) Establishes the process for identifying the limiting plant conditions to be evaluated, and

(4) Identifies the process of relating the calculated limits of acceptable stability performance to a domain of acceptable plant operation.

In support of the licensing bases adopted by a specific plant, the ABB stability methodology is performed, as required, in concert with other NRC-approved stability methodologies (e.g., BWR Owners Group Long Term Solutions). The ABB reload safety evaluation process follows all applicable requirements prescribed in the specific NRC-approved stability methodology adopted by the plant.

The general ABB thermal-hydraulic stability methodology can be used to evaluate the acceptability of plant stability performance by:

(1) Evaluating core stability performance relative to core-wide, regional, and channel stability performance acceptance limits,

(2) Using NRC-approved stability analysis methods (e.g., RAMONA-3) to demonstrate compliance with the NRC-approved code-specific applicable acceptance limit(s),

(3) Analyzing a set of plant conditions that conservatively represent all expected plant conditions for which there is a potential for thermal-hydraulic instabilities, and

(4) Correlating the calculated limits of acceptable stability performance to a core power and flow domain of acceptable plant operation, based on the plant specific licensing basis.

Core-wide and regional coupled neutronic and thermal-hydraulic stability performance and thermal-hydraulic channel stability performance are generally evaluated with the RAMONA-3 code. The RAMONA-3 code:

(5) Is described and qualified for stability analysis in CENPD-294-P (Reference 2),

(6) Uses a sufficient number of axial nodes (typically, 24 or 25) to capture the transient axial variations important to stability,

(7) For code-wide oscillation evaluations, a symmetric section of core is modeled. Each fuel channel in the sector is specifically modeled hydraulically and neutronically eliminating any power distribution averaging or grouping approximations.

(8) For regional oscillation evaluations, a symmetric section of full core is modeled. Each fuel channel is specifically modeled hydraulically and neutronically eliminating any power distribution averaging or grouping approximations.

(9) Stability performance evaluations utilizes a set of acceptance limits that include code simulation uncertainties.

Thermal-hydraulic channel stability performance can also be evaluated with the NUFREQ-NPW code. The code:

(10) Is described and qualified for stability analysis in RPA-90-91-P-A (Reference 3),

(11) Uses a sufficient number of axial nodes (typically, 24 or 25) to capture the axial shape dependencies important to channel stability,

(12) Models the limiting channel for channel oscillation evaluations,

(13) Uses a relative channel stability performance acceptance limit based on fuel designs of acceptable channel stability performance.

Core-wide and regional coupled neutronic and thermal-hydraulic stability performance and thermal-hydraulic channel stability performance can also be evaluated based on measured data. The evaluations using measured data will be presented to the NRC on a case basis, in conjunction with the specific application for which it is will be used.

3 EVALUATION DESIGN BASES AND ACCEPTANCE CRITERIA

3.1 Design Bases

The design bases for stability is documented in the "Reference Safety Analysis Report for BWR Reload Fuel" (Reference 1). The design bases, as currently stated in Section 9.2.1 of Reference 1, are repeated here.

Basis

The allowable plant operating domain for the reload core shall be defined such that the potential for growing or limit cycle power oscillations are sufficiently minimized throughout the domain. Power oscillations that can occur shall not exceed the specified acceptable fuel design limits (SAFDLs) or will be readily detected and suppressed.

Discussion

The above design basis establishes reactor thermal-hydraulic stability compliance with General Design Criteria 12 of 10CFR50 Appendix A (Reference 21).

3.2 Acceptance Criteria

A set of specific Acceptance Criteria are established to demonstrate that the design bases given in Section 3.1 are satisfied. The acceptance limits are used to define the allowable plant operating domain with respect to safe thermal-hydraulic stability performance.

The three acceptance limits used to quantify the thermal-hydraulic stability operating boundary are the:

- Core-wide Stability Limit,
- Channel Stability Limit, and
- Core Regional Stability Limit.

The instability boundaries are depicted in Figure 3-1.

The actual parameter and value of the acceptance limit used for each criterion depends on the licensed stability analysis tool. For example, the decay ratio in a core average power is typically used for core-wide stability. For channel stability, typically used parameters are the decay ratio in the channel inlet flow, decay ratio in the two-phase to single-phase pressure drop, or the relative power level of the instability threshold. For regional stability, typically used parameters are the phase shift in local powers, local power decay ratios, relative power level of the instability threshold, or combination of core-wide and channel decay ratios. For each acceptance criterion and associated stability analysis tool, an acceptance limit parameter and limiting value is established. The limiting allowable value is set such that uncertainties associated with the analysis tool are included. Figure 3-2 depicts the instability boundaries for a specific set of calculational tools.

A stability evaluation confirms that for conditions within the plant allowable operating domain (i.e., power, flow, exposure, control rod patterns, power distribution), the plant is within the instability boundaries.

The three acceptance criteria can be simplified to two acceptance criteria by approximating the regional stability limit by a combination of the core-wide and channel stability limits. This approximation is based on the observation that the out-of-phase (regional) mode is more sensitive to the channel thermal-hydraulic component compared to the in-phase mode, where the core neutronic component is dominating. Therefore the channel stability margin is a reasonable approximation of the regional stability margin. Figure 3-3 depicts this simplified instability boundary.

3.2.1 Core-Wide Stability

Acceptance Criteria

The code-wide stability, as determined by the decay ratio in core average reactor power, will be less than one, including uncertainties.

RAMONA-3 Acceptance Limit

The acceptance limit for the RAMONA-3 code is a calculated core-wide decay ratio less than [Proprietary Information Deleted]

Measured Data Acceptance Limit

The acceptance limit for in-plant measured data is based on the specific uncertainties associated with the accuracy of the particular plant measurement and plant conditions.

Discussion

The RAMONA-3 stability analysis code is the intended tool for reload stability analysis of core-wide power oscillations. The code and qualification is presented in Reference 2 and summarized in Appendix B. Under certain circumstances, available in-plant stability measurements may be a desirable alternative reload stability analysis tool. Reload stability analysis core-wide oscillations using in-plant measurement data will be presented, as warranted, to the NRC under separate application.

3.2.2 Channel Stability

Acceptance Criteria

The channel stability, as determined by the most unstable channel throughout the core, for which flow oscillations as a result of only density-wave oscillations are damped, including uncertainties.

RAMONA-3 Acceptance Limit

The acceptance limit for the RAMONA-3 code is a calculated maximum channel decay ratio throughout the core less than [Proprietary Information Deleted]

NUFREQ-NPW Acceptance Limit

The acceptance limit for the NUFREQ-NPW code is the calculated maximum channel decay ratio less than or equal to the calculated maximum channel decay ratio of other licensed fuel designs.

Measured Data Acceptance Limit

The acceptance limit for measured data from in-plant or loop tests is that the relative margin to the instability limit for the reload fuel design is less than or equal to that for other licensed fuel designs.

Discussion

The RAMONA-3 stability analysis code is generally the intended tool for reload stability analysis of channel density-wave oscillations. The code and qualification is presented in Reference 2 and summarized in Appendix B. An alternative tool for relative comparison is the NUFREQ-NPW code documented in Reference 3. In some circumstances, test loop or in-plant measurements may be available as an alternative method for demonstrating relative channel stability margin. Channel stability oscillations using measured test loop or in-plant data, as warranted, will be presented to the NRC under separate application.

3.2.3 Core Regional Stability

Acceptance Criteria

The core regional stability, as determined by out-of-phase power oscillations between local regions of the core, will not be present when uncertainties are included.

RAMONA-3 Acceptance Limit

The acceptance limit for the RAMONA-3 code is:

[Proprietary Information Deleted]

Measured Data Acceptance Limit

The acceptance limit for in-plant measured data is based on the specific uncertainties associated with the accuracy of the particular plant measurement and plant conditions.

Discussion

The RAMONA-3 stability analysis code is the intended tool for reload stability analysis of core regional oscillations. The code and qualification is presented in Reference 2 and summarized in Appendix B. Under certain circumstances, available in-plant stability measurements may be a desirable alternative reload stability analysis tool. Reload stability analysis regional oscillations using in-plant measurement data or eigenvalue separation analysis will be presented, as warranted, to the NRC under separate application.

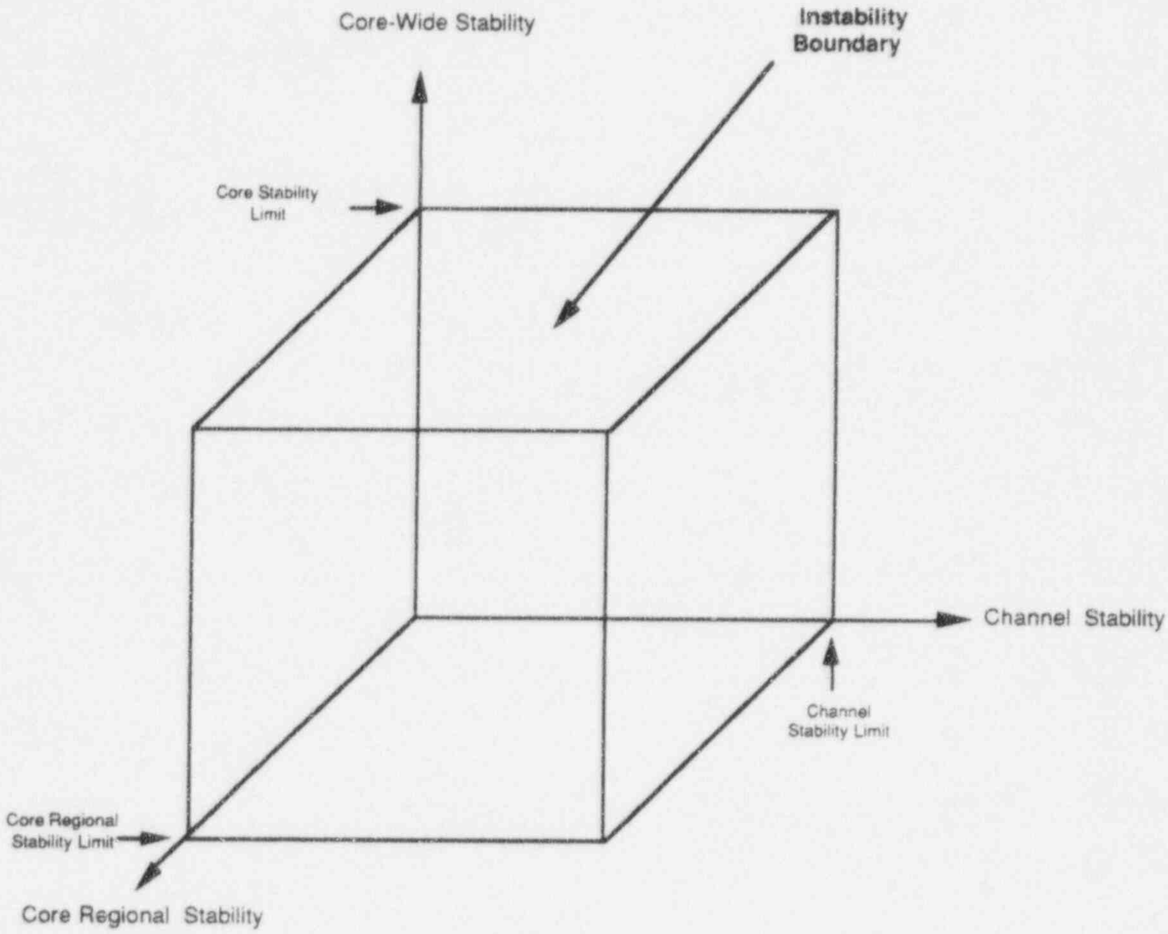

Figure 3-1 Thermal-Hydraulic Instability Boundaries

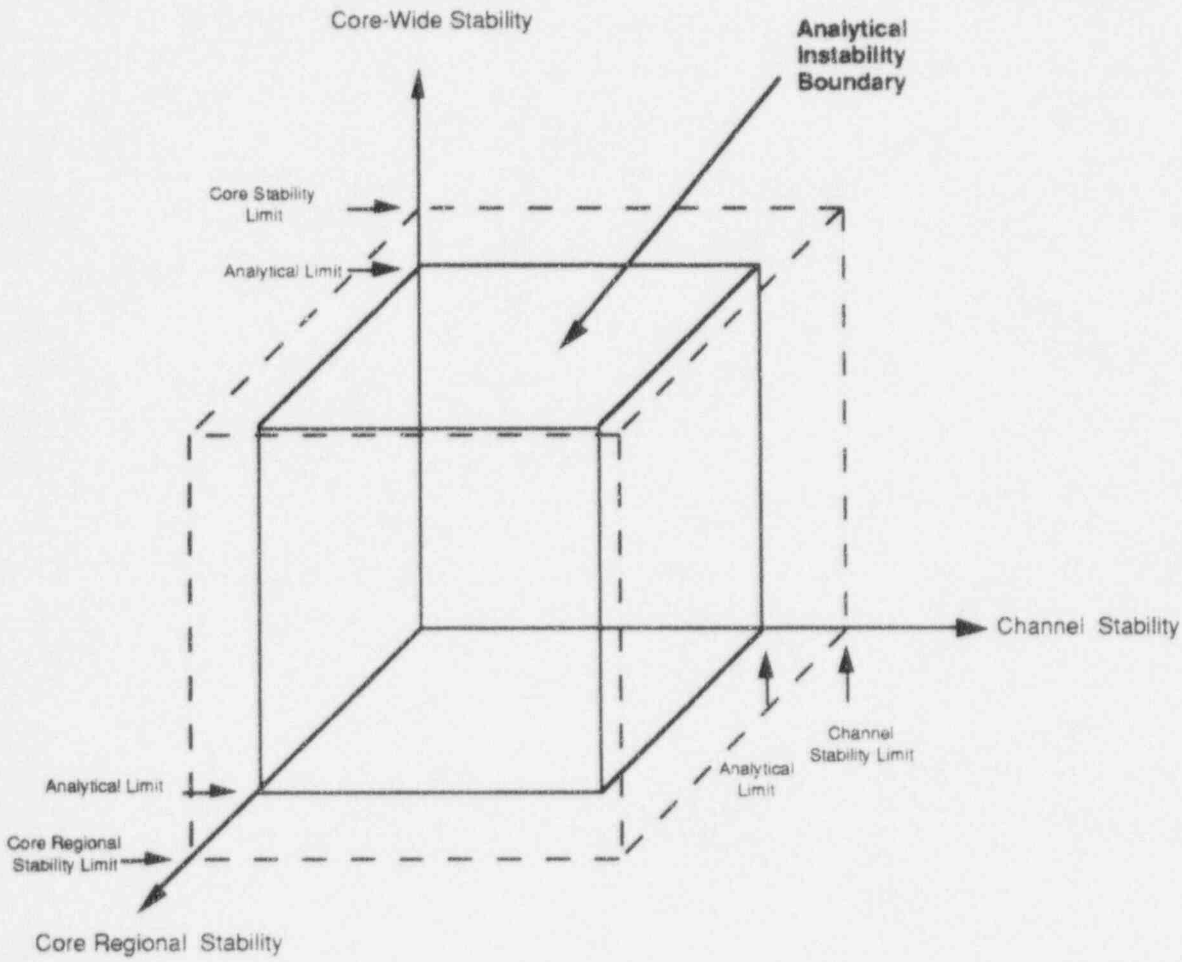

Figure 3-2 Analytical Thermal-Hydraulic Instability Boundaries

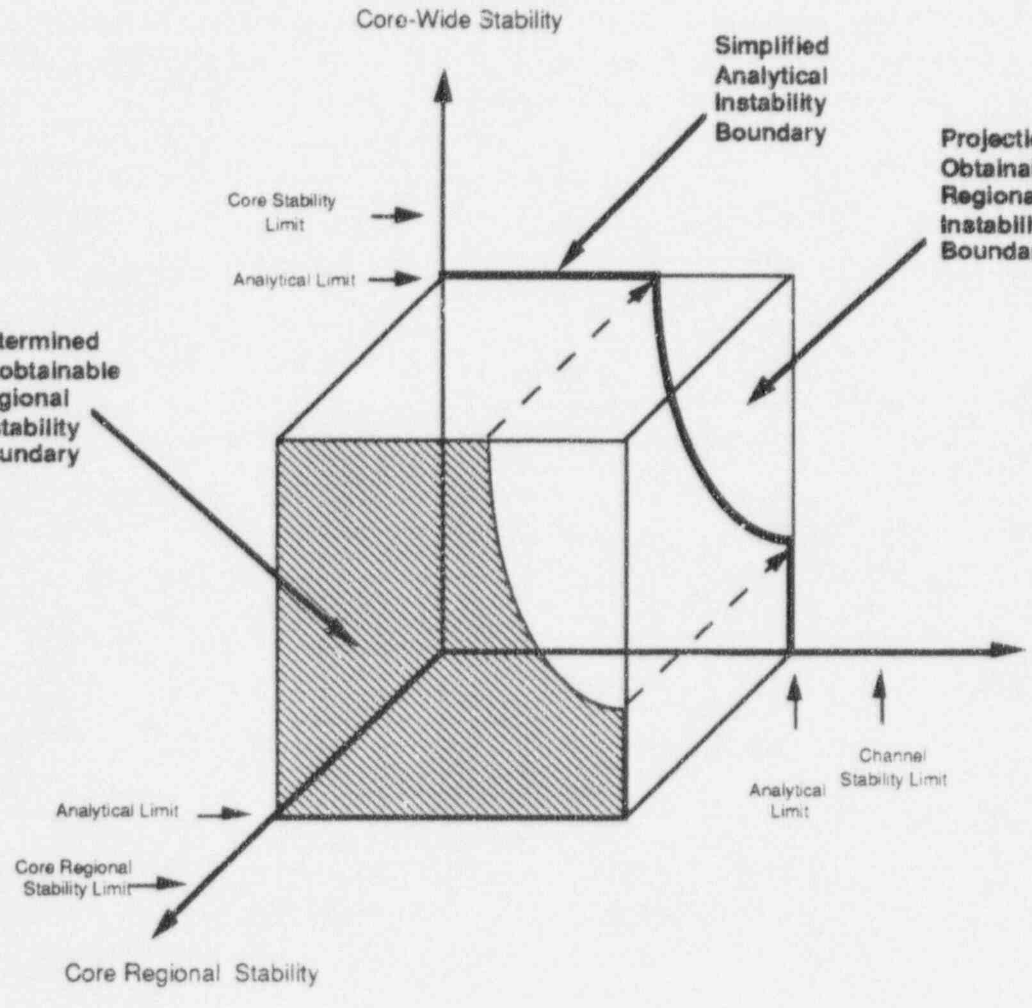

Figure 3-3 Simplified Thermal-Hydraulic Instability Boundary

4 STABILITY ANALYSIS TOOLS

ABB general stability analysis methodology relies on the NRC-approved codes such as RAMONA-3 and NUFREQ-NPW. RAMONA-3 is a time-domain transient systems code for the prediction of BWR dynamic behavior. NUFREQ-NPW is a frequency domain stability code used for stability trend evaluations.

If warranted by a specific application, the ABB stability analysis codes are augmented by stability performance data either from experimental test loop or in-plant measurements.

4.1 RAMONA-3

RAMONA-3 is a time-domain transient systems code used for predicting the dynamic behavior of BWRs. Because RAMONA-3 combines three-dimensional modeling of the core with the potential to be able to represent each bundle in the core, it is capable of predicting local effects resulting from transients. In particular, it is well suited for simulating core-wide and regional oscillations in the core, and for predicting thermal-hydraulic density-wave oscillations within any fuel channel.

A detailed code description is given in Reference 2. This reference also describes in detail the stability qualification data base for RAMONA-3 and discusses the code stability margin prediction uncertainty.

In the reload analysis methodologies described here, RAMONA-3 is used to calculate the core-wide, regional, and channel stability performance margins.

4.2 NUFREQ-NPW

NUFREQ-NPW is a frequency domain code for calculating stability performance margins in BWRs. The code has been approved by the NRC for determining relative changes in stability for different core configurations. Core-wide decay ratio is estimated from the power-to-external pressure perturbation transfer function, and channel decay ratio is based on the two-phase region pressure drop to single-phase region pressure drop transfer function.

Reference 3 describes the code models and provides a detailed discussion on code benchmarking and verification. Results of the review conducted by the NRC and recommendations on the use of the code are also contained in Reference 3.

In the reload analysis methodologies described here, NUFREQ-NPW is used to estimate the relative changes in channel stability.

4.3 Measured Stability Data

Under some circumstances, a reload safety evaluation of stability performance may require analysis of measured stability data to supplement analytical calculations. Measured stability data can be used to quantify conservative margins or to quantify relative performance characteristics for a specific plant or fuel applications. Measured stability data will be presented to the NRC separately, under the specific reload application for which the analysis is intended to be used. As an illustration, the following subsections outline three types of measured stability data available for plant-specific evaluations. Other types of stability data may become available in the future.

4.3.1 Experimental Loop Instability Limit Measurements

Test loop measurements of the instability limit for numerous different fuel designs, have been performed. The test loop measurements do provide an actual measure of the relative stability performance of different fuel designs.

For example, Section 5 of Reference 2 describes loop tests measurements performed for several fuel designs. Detailed examination of test loop data can demonstrate relative thermal-hydraulic channel stability performance of specific fuel designs.

4.3.2 In-plant Channel Flow Measurements

In-plant individual channel flow measurements have been made at several reactors designed by ABB Atom. Work has been done on correlating the noise in the measured channel flows to the relative channel stability performance (Reference 22). Detailed examination of in-plant flow measurement data can demonstrate the relative thermal-hydraulic channel stability performance of specific fuel designs.

4.3.3 In-plant Core Stability Measurements

Extensive in-plant core stability measurements have been taken in European plants. Measurements have been performed for numerous core configurations (See Section 6 of Reference 2). Plant measurements provide an actual measure of the relative core stability performance for varying plant conditions. Furthermore, plant measurements in a specific plant also provide a reference for more precise quantification of generic analytical model uncertainties.

Detailed examination of in-plant stability data can demonstrate the relative core stability performance for specific core designs.

5 RELOAD ANALYSIS METHODOLOGY

5.1 General Reload Methodology

The following steps are performed in a reload analysis for core thermal-hydraulic stability:

(1) Determine the plant stability licensing design basis.

(2) Confirm that the current plant stability limits are applicable for the new reload cycle, following the plant licensing base reload review procedures.

(3) As required, perform stability analysis for the new reload cycle in accordance with the plant licensing base. For example: (a) evaluate the reload core and fuel design cycle-specific instability power-flow boundaries, or (b) confirm fuel design and/or core design relative stability performance.

Each U.S. licensee has as part of the plant licensing basis, a stability application methodology accepted by the NRC. The licensing basis includes processes for evaluating acceptable plant stability performance for each new reload cycle. The following sections discuss the general process used to evaluate stability performance and specifically to evaluate the power-flow stability boundaries. The general process can be applied under the constraints and requirements of a specific application methodology adopted by the licensee. Several NRC accepted application methodologies are discussed in Section 5.2 outlining specifics of how the general reload methodology is applied, as required.

5.1.1 Instability Boundary

The power-flow instability boundary is determined as the limiting boundary of the core-wide, channel, and regional stability boundaries. Each boundary defines power and flow conditions that comply with the stability acceptance limit throughout the plant allowable operating domain. The steps in constructing the instability boundary are:

[Proprietary Information Deleted]

5.1.2 Important Stability Parameters

The plant parameters that are important to thermal-hydraulic stability performance are shown in Table 5-1. Listed in the table are the key Physical Parameters for four groups in increasing order of frequency of change. The groups are:

(1) Plant Design Parameters

(2) Core and Fuel Design Parameters

(3) Cycle Exposure Parameters

(4) Operating Condition Parameters

Each Physical Parameter influences one or more Stability Parameters. Stability Parameters are plant, core and fuel characteristics that have a known influence on core-wide, regional, and channel stability. [Proprietary Information Deleted]

5.1.3 Analysis Conditions

Plant reload stability evaluations encompass the plant normal and anticipated operating domain. Stability analyses are performed for a set of conditions that captures all expected and anticipated plant operating conditions. [Proprietary Information Deleted]

5.1.4 Stability Margin Calculation

An NRC approved analysis code is used for core-wide, regional, and channel stability margin calculations. The evaluation process for calculating the stability margins using the stability tool, RAMONA-3, is discussed below. The evaluation process using the NUFREQ-NPW code is discussed in Reference 3. Specific evaluations using other stability analysis tools will be presented to the NRC, with the specific reload application for which the analysis is to be applied.

5.1.4.1 Core-Wide Stability

Core-wide stability reload evaluations are performed with RAMONA-3. The steps in evaluating the core-wide decay ratio are:

[Proprietary Information Deleted]

5.1.4.2 Channel Stability

Channel stability reload evaluations are performed with RAMONA-3. The steps in evaluating the channel decay ratio are:

[Proprietary Information Deleted]

5.1.4.3 Core-Regional Stability

Regional stability reload evaluations are performed with the RAMONA-3. [Proprietary Information Deleted]

5.2 Application Methodologies

Each U.S. BWR plant has a licensing basis for plant stability performance using NRC approved code methods and methodology. A safety evaluation of the plant stability performance is performed for each reload application following an evaluation process consistent with the plant specific licensing bases. Reload safety evaluations performed by ABB shall be consistent with the approved methodology adopted by the plant in question, and, as warranted, will use NRC-approved stability analysis methods.

The following subsections describe the most common plant specific stability licensing bases and how, as required, the ABB stability analysis methods and ABB general reload analysis methodology are applied. The stability licensing bases discussed are:

- Plant Implementation of BWROG Interim Corrective Actions,
- BWROG Long Term Solution Option Enhanced I-A,
- BWROG Long Term Solution Option I-D,
- BWROG Long Term Solution Option II,
- BWROG Long Term Solution Option III, and
- Other Stability Reload Application Methodologies

5.2.1 Plant Implementation of BWROG Interim Corrective Actions

The current plant licensing base in most plants is the plant specific implementation of the BWROG recommended interim corrective actions (References 7 and 8). The BWROG recommended interim corrective actions identifies regions of the power-flow map restricted from plant operation, and identifies other plant operation restrictions in the vicinity of the restricted domain (e.g., power distribution controls). A licensee stability licensing basis may be a plant specific implementation of the BWROG interim corrective actions recommendations. For this plant licensing basis, any reload specific analyses performed in developing the plant operating boundaries are confirmed for new reload. As required, the general reload methodology is used with the specific assumptions adopted by the plant. For example, some plants have adopted generic power-flow boundaries applicable for all reloads. Other plants have power-flow boundaries that are demonstrated to be applicable to a specific reload cycle.

5.2.2 BWROG Long Term Solution Opti›n Enhanced I-A

BWROG Long Term Solution Option Enhanced I-A solution concept and supporting methodology is documented in NEDO-31960 (Reference 9). Additional supporting information is documented in NEDO-32339 (Reference 11). The Option Enhanced I-A solution includes modifications to the flow-biased APRM neutron flux scram signal used to detect and suppress oscillations. In addition, power distribution controls and core stability performance monitoring hardware are used to avoid core-wide and regional power oscillations.

The NRC-approved analysis methodology prescribes:

- The process for validating an NRC-approved stability analysis method for a specific plant application (Appendix A of Reference 11),
- The plant event assumptions to be used as the operating conditions in the analysis of reload stability performance (Appendix A of Reference 11), and
- The process for reviewing stability performance in subsequent reloads (Appendix B of Reference 11).

For plants with the BWROG Option Enhanced I-A methodology as the stability licensing basis, the ABB stability analysis tools (e.g., RAMONA-3) will be used in implementing the methodology.

Currently, the procedures prescribed in the approved methodology are followed with RAMONA-3 code. The analysis conditions are as prescribed in Reference 11. The acceptance limits and analysis process are as described in Section 3.2 and 5.1 of this document.

5.2.3 BWROG Long Term Solution Option I-D

BWROG Long Term Solution Option I-D concept and supporting methodology is documented in NEDO-31960 (Reference 9). Additional supporting information is documented in NEDO-32465 (Reference 10), and the first application of the generic methodology (to Vermont Yankee Nuclear Power Station) is addressed in References 12, 13 and 14.

BWROG Option I-D introduces an administratively controlled exclusion region to prevent oscillations and the automatic function through the flow-biased APRM neutron flux scram signal to detect and suppress oscillations. This generic option was designed for BWR plants with relatively tight fuel inlet orificing. These plants have a small relative probability of regional oscillation, and hence large probability of automatic detection and suppression with the current flow-biased APRM signal.

For plant specific implementation of Option I-D, typical supporting analyses, include:

(1) determining the administratively controlled exclusion region,

(2) demonstrating that the flow-biased APRM neutron flux scram signal suppresses oscillations before exceeding design acceptance limits.

(3) demonstrating regional oscillations are of sufficiently low probability,

(4) showing sufficient power distribution controls in a "restricted" region adjacent to the exclusion region.

Reload safety evaluations for Option I-D implementation at a specific plant generally will confirm the supporting licensing analysis and region boundaries remain valid for the new reload application. As required in the reload evaluation process, stability analysis shall be performed using the general reload analysis methodology described in Section 5.1.

5.2.4 BWROG Long Term Solution Option II

BWROG Long Term Solution Option II solution concept and supporting methodology is documented in NEDO-31960 (Reference 9). The application of the generic methodology (to Oyster Creek Nuclear Power Station) is addressed in Reference 15.

BWROG Option II demonstrates that the existing quadrant-based APRM system is sufficient to detect and suppress power oscillations. This generic option was designed for the BWR/2 class of plants, which have quadrant-based APRM system. These plants have an flow-biased APRM system that will detect regional oscillations in addition to detecting core-wide oscillations.

For plant specific implementation of Option II, typical supporting analyses:

(1) demonstrating that the flow-biased APRM neutron flux scram signal suppresses oscillations before exceeding design acceptance limits, and

(2) demonstrating acceptable channel stability performance.

Reload safety evaluation for Option II implementation at a specific plant generally will confirm the supporting licensing analysis remain valid for the new reload application. Typical analysis is for channel stability performance. As required in the reload evaluation process, stability analysis shall be performed using the general reload analysis methodology described in Section 5.1.

5.2.5 BWROG Long Term Solution Option III

BWROG Long Term Solution Option Enhanced III solution concept and supporting methodology is documented in NEDO-31960 (Reference 9). Additional supporting information is documented in NEDO-32465 (Reference 10). The Option Enhanced III solution introduces stability reactor protection hardware that detects and subsequently suppresses core-wide or regional power oscillations.

The supporting analysis methodology is performed for each plant implementing the hardware. Reload safety evaluations confirm that the supporting licensing analysis remain valid for the new reload application. The analysis methodology includes reload review procedures to confirm applicability of the supporting analysis for each new fuel and reload core design. Cycle specific stability analysis is not intended to be performed for plant that have implemented Option III.

5.2.6 Other Stability Reload Application Methodologies

A licensee may have a plant-specific licensing analysis basis that deviates or augments the general NRC-approved methodologies discussed in Sections 5.2.1 through 5.2.5. For example, a licensee may have a requirement to also evaluate the core stability performance for each new reload for specific set of analysis conditions. The ABB general reload analysis methodology will be used to perform any additional reload-specific stability evaluations.

6 REFERENCES

1. "Reference Safety Report for Boiling Water Reactors Reload Fuel," ABB Report CENPD-300-P-A (proprietary), CENPD-300-NP-A (non-proprietary), July 1996.

2. "Thermal-hydraulic Stability Analysis Methods for Boiling Water Reactors," ABB Report CENPD-294-P-A (proprietary), CENPD-294-NP-A (nonproprietary), July 1996.

3. NUFREQ-NPW: An ABB Atom Computer Code for Core Stability Analysis of Boiling Water Reactors, ABB Report RPA-90-91-P-A (proprietary), RPA-90-91-NP-A (nonproprietary), December 1991.

4. General Electric Service Information Letter 380, SIL-380, Rev. 0 (1982), Rev. 1 (1984).

5. Technical Resolution of Generic Issue B-19-Thermal Hydraulic Stability, NRC Generic Letter 86-02 (January 23, 1986).

6. Power Oscillations in Boiling Water Reactors (BWRs), NRC Bulletin 88-07 (June 15, 1988) and Supplement 1 (December 30, 1988).

7. Implementation Guidance for Stability Interim Corrective Action, Letter BWROG-92030 to BWR Owners' from R. D. Binz (BWROG), March 18, 1992.

8. BWR Owners' Group Guidelines for Stability Interim Corrective Action, Letter BWROG-94079 to M. J. Virgilio (NRC) from L. A. England (BWROG), June 6, 1994.

9. BWR Owners' Group Long-Term Stability Solution Licensing Methodology General Electric Report NEDO-31960, May 1991.

   BWR Owners' Group Long-Term Stability Solution Licensing Methodology (Supplement 1), General Electric Report NEDO-31960 Supplement 1, March 1992.

10. BWR Owners' Group Reactor Stability Detect and Suppress Solution Licensing Basis Methodology and Reload Applications, General Electric Report NEDO-32465, May 1995.

11. Reactor Stability Long Term Solution: Enhanced Option I-A, General Electric Report NEDO-32339, March 1994.

12. Licensing Topical Report, Application of the "Regional Exclusion with Flow Based Scram APRM Neutron Flux Scram" Stability

Solution (Option 1-D) to Vermont Yankee Nuclear Power Plant, GENE-637-018-0793, General Electric, July 7, 1993.

13. D. A. Reid (Vermont Yankee) Letter to U.S. NRC, "Proposed Change No. 173. BWR Thermal Hydraulic Stability and Plant-Information Requirements for BWROG Option 1-D Long Term Stability Solution, March 31, 1994.

14. D. A. Reid (Vermont Yankee) Letter to U.S. NRC, "Response to NRC Generic Letter 94-02 Regarding Stability Interim Corrective Actions and Plans for Implementation of BWROG Long Term Stability Solution 1-D at Vermont Yankee," September 9, 1994.

15. J. J. Barton (GPU) to U.S. NRC, Technical Specification Change Request No. 191, October 9, 1991.

16. U.S. NRC Letter to L. A. England (BWROG) "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owners' Group Long-Term Stability Solution Licensing Methodology" (TAC No. M75928) July 12, 1993 .

17. U.S. NRC Letter to R.A. Pinelli (BWROG) from R. C. Jones, Jr. (NRC) "Acceptance for Referencing of Topical Report NEDO-32339, "Reactor Stability Long Term Solution: Enhanced Option I-A," April 24, 1995.

18. U.S. NRC letter from D. H. Dorman (NRC) to D. A Reid (VYNPC), "Thermal Hydraulic Stability - Vermont Yankee Nuclear Power Station (TAC NO. M87091), March 30, 1995.

19. U.S. NRC letter from A. W. Dromerick (NRC) to J. J. Barton (GPU), "Issuance of Amendment (TAC NO. M81944), December 29, 1994.

20. Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors, NRC Generic Letter 94-02, July 11, 1994.

21. Code of Federal Regulations, Energy, Article 10, Office of The Federal Register, U.S. National Archives and Records Administration.

22. J. Blomstrand, et al. "Noise Analysis of Core Coolant Channel Flow Signals, Recorded in Swedish and Finnish BWRs," Progress in Nuclear Energy, Vol. 21, pp. 777-787, 1988.

TABLE 5-1 THROUGH 5-3

PROPRIETARY INFORMATION DELETED

APPENDIX A: RELOAD APPLICATION EXAMPLES

A.1 Example 1 - General Evaluation of Stability Boundary

This section shows an example of generating the instability boundary for a plant. The plant is a 764 assembly, C-lattice General Electric BWR/5 plant. Table A-1 summarizes the plant design.

A.1.1 Plant Licensing Basis

In this example, it is assumed that the a power-flow stability boundary has been established in the previous reload cycle. The boundary is to be confirmed for the current reload cycle. The power-flow boundary is shown in Figure A-1. [Proprietary Information Deleted]

A.1.2 Reload Description

[Proprietary Information Deleted]

A.1.3 Cycle Exposure Evaluation

[Proprietary Information Deleted]

A.1.4 Stability Power-Flow Boundary Evaluation

[Proprietary Information Deleted]

TABLE A-1 THROUGH A-6

PROPRIETARY INFORMATION DELETED

FIGURE A-1 THROUGH A-10

PROPRIETARY INFORMATION DELETED

APPENDIX B: SUMMARY OF RAMONA-3 ACCEPTANCE LIMITS QUALIFICATION

The following subsections summarize the qualification bases for the acceptance limits presented in Section 3. Details of the analysis and calculations supporting the chosen acceptance limits are presented in Reference 1.

B.1 Core-wide Stability Acceptance Limit Qualification

[Proprietary Information Deleted]

B.2 Channel Stability Acceptance Limit Qualification

[Proprietary Information Deleted]

B.3 Core Regional Stability Acceptance Limit Qualification

[Proprietary Information Deleted]

FIGURE B-1 THROUGH B-4

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APPENDIX C: DETAILED DESCRIPTION OF BWR/5 STABILITY CALCULATION

This appendix provides a detailed description of the calculated core-wide stability for two operating points.

These operating points are:

Case 1: 23.8% Core Flow, 35.3% Power and,

Case 2: 45.0% Core Flow, 68.5% Power

The Tables C-1 through C-11 and Figures C-1 through C-3 provide detailed information related to the Case 1 and 2 points.

TABLE C-1 THROUGH C-11

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FIGURE C-1 THROUGH C-3

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