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**ABB Combustion Engineering Nuclear Operations**

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Rec'd with letter dtd. 7/31/96

CENPD-300-NP-A

# Reference Safety Report for Boiling Water Reactor Reload Fuel

**July 1996**

**ABB Combustion Engineering Nuclear Operations**

ABB

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# CENPD-300-NP-A REPORT CONTENTS

# CENPD-300-NP-A REPORT

## Part I

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

ABB Combustion Engineering Nuclear Operations

ABB

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

MAY 2 4 1996

Derek Ebeling-Koning, Manager
Licensing and Safety Analysis BWR Fuel Operations
ABB CENO Fuel Operations
1000 Prospect Hill Road
Windsor, Connecticut 06095-0500

SUBJECT: CENPD-300-P "REFERENCE SAFETY REPORT FOR BOILING WATER REACTOR RELOAD FUEL," (TAC NO. M91197)

Dear Dr. Ebeling-Koning:

ABB Combustion Engineering, Inc. submitted the report CENPD-300-P by letter dated December 8, 1994. Related documents were submitted by letters dated August 28, 1995, September 12, 1995, November 8, 1995 and February 5, 1996. The staff has completed its review of the subject topical report and the related documents. The staff finds the report and related documents to be acceptable to the extent specified and under the limitations delineated in the NRC evaluation.

The staff does not intend to repeat the review of the matters that are described in the report and that were found acceptable when the report appears as a reference in license applications, except to ensure that the material presented is applicable to the specific plant involved. The staff's acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that ABB Combustion Engineering, Inc. publish an accepted version of this report within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation after the title page. The accepted version shall include an -A (designating accepted) following the report identification symbol.

Should the staff's criteria or regulations change so that its conclusions as to the acceptability of the report are invalidated, Combustion Engineering and/or applicants referencing the topical reports will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical reports without revision of their respective documentation.

Sincerely,

Timothy E. Collins /for

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

Enclosure:
CENPD-300-P Safety Report for BWR

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

ATTACHMENT 1

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO TOPICAL REPORT CENPD-300-P "REFERENCE SAFETY REPORT FOR BOILING WATER REACTOR RELOAD FUEL" TAC NO. M91197

## 1.0 INTRODUCTION

By letter dated December 8, 1994, ABB Combustion Engineering, Inc. (ABB/CE) submitted Topical Report CENPD-300-P, "Reference Safety Report (RSR) Boiling Water Reactor Reload Fuel" which describes the reload fuel design and safety analysis process used in specific plant applications. Specific topics related to the ABB BWR reload fuel design and safety analysis methodology are contained in numerous Licensing Topical Reports describing portions of the overall methodology. This RSR integrates all the separate reports into a single reload fuel design and safety analysis methodology. This report was supplemented with submittals dated August 28, 1995 CENPD-300-P-RAI, "Reference Safety Report for Boiling Water Reactor Reload Fuel: Response to Request for Additional Information" and September 12, 1995 CENPD-300-P-RAI Rev 1. Additional information discussed in meetings October 26, 1995 and February 1, 1996 was contained in submittals dated November 8, 1995 and February 5, 1996.

## 2.0 SUMMARY

The staff was assisted in its review of CENPD-300-P by International Technical Services, Inc. (ITS) under a technical assistance contract. The evaluation and findings are described in the attached ITS technical evaluation report (TER) which becomes a part of this report.

The purpose of this topical report (CENPD-300-P) was to provide a reference document containing ABB/CE's complete reload methodologies in support of licensing action, whether it be related to reload fuel, changes to the plant operating domain or equipment performance characteristics.

3.0 EVALUATION

ABB/CE provided additional information to supplement the information contained in CENPD-300-P in submittals CENPD-300-P-RAI (Reference 3) dated August 28, 1995, and CENPD-300-P-RAI Rev. 1 (Reference 4) dated September 12, 1995. These were in response to a request for additional information (RAI) prepared by the staff and their contractor (Reference 2). The contractor evaluation and findings are described in the ITS TER which is enclosed as a part of this evaluation. Additional information was discussed during meetings October 26, 1995 and February 1, 1996, (References 5 and 6) to address Conditions 4 and 5 in the ITS report. The topical report contained a generic reload analysis methodology and was therefore reviewed as being plant independent and fuel independent. The main body of the report consists of the reload fuel methodologies: mechanical, nuclear and thermal-hydraulic methodologies are employed to address different aspects of reload analysis, including fuel performance and design limits, core designs, hydraulic compatibility of fuels, determination of operating limits and performance of safety analysis. Appendix A covers the applicable computer codes, while Appendix B presents the format of a plant and cycle specific reload safety analysis summary report. Appendix C presents additional safety analyses which may be required if the licensee's existing required set of analyses is larger than ABB/CE's generic set of reload analyses or for gaining operating flexibility. Methodology for determining transient axial power distributions is presented in Appendix D and Appendix E illustrates by example how the content of the main report is intended to be used.

The staff has reviewed the contractor's evaluation as documented in the TER, and in general concurs with its findings. Differences are addressed below. In the text of the TER as well as in the listed limitations and restrictions in both the TER and in this SER, there are several references to additional justification or demonstration which needs to be provided on a plant specific basis. It is the staff's intention that such justification be part of the reload evaluation and that it be maintained and available for staff audit.

The staff has eliminated Condition 2 of the TER, because the CENPD-300-P provides the basic anslysis methodology, while deviations and changes addressed by Condition 2 would fall outside the analysis described in CENPD-300-P. Thus they would be evaluated by the normal 10 CFR 50.59 process and submitted. Condition 3 has been rewritten to specify that the Operating Limit MCPR must be calculated with Method A, since the statistical approach has not been justified.

We have reviewed the additional information regarding use of a time dependent axial power distribution and find it adequate. The staff finds the use of the time dependent axial power distribution acceptable, thus eliminating Condition 4 in the ITS report. Condition 6 has been rewritten as Condition 4 stating that for compliance with Appendix K, ABB/CE must use 1.2 times the ANS71 as stated in the current 10 CFR Part 50, Appendix K.

Condition 7 has been eliminated since it is already covered in the conditions stated in the SER on CENPD-292 on BISON. Condition 8 has been eliminated since it is already covered in Condition 1 and the other approved topical reports. Condition 10 has been rewritten in a simplier manner as Condition 5. Condition 11 has been rewritten as Condition 6 to more clearly state the intent.

Section 3.1.5.3 of the TER addresses the Rotated Fuel Assembly Accident which is discussed in Section 8.5.2 of the topical report. After reviewing this section of the TER and the topical report section, the staff finds that since the uncertainty analysis has not been validated a condition on the use of a variable water gap is necessary. This is Condition 8.

Section 3.1.5.4 of the TER addresses ATWS evalution. The staff does not agree with the details of the statement. As stated in Section 9.5.2 of CENPD-300-P, each new ABB/CE fuel design will be confirmed to comply with the design characteristic of the core assumed in the plant licensing basis ATWS analysis. Once confirmed not to have a significant impact on the current ATWS analysis, it is considered acceptable.

TER Section 3.2, Operational Flexibility, needs some clarification. For plants that have any of the plant flexibility options included in their current licensing basis, ABB/CE will perform the analyses associated with the particular option(s) as a part of the reload safety analysis using the approved methodology detailed in CENPD-300-P or previously approved topical reports. However, if a licensee desires to change its license basis to incorporate a new plant flexibility option, the change must be made in accordance with the requirements of 10 CFR 50.59.

Section 3.3 of the TER addresses transition cores, specifically the handling of CPR correlations for non ABB/CE fuel. Additional information dealing with the critical power ratio (CRP) correlation for transitions cores was discussed at both the October 26, 1995 and February 1, 1996 meetings. Based on this information we have eliminated Condition 5 from the ITS report and added Condition 7 listed below.

Finally, with respect to the contractor's observations and recommendations relating to ABB/CE's administrative control of the computer codes, the NRC staff performed an inspection of ABB Combustion Engineering Nuclear Operations (Reference 7). The inspection report contained a Notice of Nonconformance. ABB/CE responded to the Notice of Nonconformance (Reference 8) and the staff reviewed the information provided. As stated in the staff response (Reference 9), ABB/CE was responsive to the staff concerns. Furthermore, the staff plans an inspection later this year to ensure that full compliance has been achieved and will be maintained. Administrative control of computer codes and associated documentation will be part of this ABB/CE inspection. In addition, a plant specific audit is planned for the first use of the ABB/CE BWR methodology.

We find that ABB/CE has adequately described its BWR reload methodology subject to the limitations and restrictions described below:

1. Acceptability of this topical report is subject to review findings of the other relevant topical reports cited in the topical report, and all conditions set forth therein are applicable to this topical report.

Furthermore, acceptability of reload analysis is subject to conditions cited in the methodology topical reports.

2. ABB/CE's uncertainty analysis approach is not generically acceptable since the acceptability is highly application dependent. The Operating Limit MCPR must be calculated with Method A.

3. The use of ANS79 decay heat curve is not acceptable for LOCA analysis. For compliance with Appendix K, ABB/CE must use 1.2 times the ANS71 as stated in the current 10 CFR Part 50, Appendix K.

4. No evaluation of validity of sample analyses was performed. Furthermore, the approval recommended in this report does not imply any endorsement of analyses nor of the quantified uncertainties set forth in Appendix D. Therefore, no reference should be made to Appendix D as demonstration in support of any future reload.

5. At the minimum, each reload safety evaluation report should contain all the items referred to in Appendix B of the topical report.

6. ABB/CE must use 110% of vessel design pressure for the peak reactor vessel pressure limit unless otherwise governed by a plant specific licensing basis.

7. The ABB/CE methodology for determining the operating limit maximum critical power ratio (OLMCPR) for non-ABB/CE fuel as described in CENPD-300-P and additional submittals (References 1, 3, 4, 5 and 6) is acceptable only when each licensee application of the methodology identifies the value of the conservative adder to the OLMCPR. The correlation applied to the experimental data to determine the value of the adder must be shown to meet the 95/95 statistical criteria. In addition, the licensee's submittal must include the justification for the adder and reference the appropriate supporting documentation.

8. For the rotated fuel assembly analysis ABB/CE stated its intent to vary gap sizes to reduce conservatism in the analysis accompanied by uncertainty analyses to establish the impact. Since the acceptability of this approach depends upon the validity of the uncertainty analysis, which has not beeen validated this approach is not acceptable.

## 4.0 REFERENCES


1. Letter to NRC from D. Ebeling-Koning (ABB) dated December 8, 1994, Transmittal of CENPD-300-P.

2. Letter from M. Chatterton (NRC) to D.B. Ebeling-Koning (ABB), "Requests for Additional Information on review of CENPD-300-P," dated April 4, 1995.

3. Letter to NRC from D. Ebeling-Koning (ABB) dated August 28, 1995, Transmittal of CENPD-300-P-RAI.

4. Letter to NRC from D. Ebeling-Koning (ABB) dated September 12, 1995, Transmittal of CENPD-293-P-RAI Rev 1.

5. Letter to NRC from D. Ebeling-Koning (ABB) dated November 8, 1995, Presentation Slides from October 26, 1995, NRC Meeting, NFBWR-95-163.

6. Letter to NRC from D. Ebeling-Koning (ABB) dated February 5, 1996, Presentation Slides from February 1, 1996, NRC Meeting, NFBWR-96-012.

7. Letter to Donald E. Allen (ABB/CE) from Robert M. Gallo (NRC) dated December 29, 1994, Inspection Report No. 99900002/94-01; 99900102/94-01.

8. Letter from Donald E. Allen (ABB/CE) dated February 24, 1995, Response to a Notice of Conconformancein NRC Inspection Report 99900002/94-01; 99900102/94-01.

9. Letter to Donald E. Allen (ABB/CE) from Gregory C. Cwalina (NRC) dated March 30, 1995.

ITS/NRC/95-7

# TECHNICAL EVALUATION:
# REFERENCE SAFETY REPORT FOR BOILING WATER REACTOR RELOAD FUEL
# CENPD-300-P
# FOR
# ABB COMBUSTION ENGINEERING NUCLEAR OPERATIONS

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Prepared for
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under NRC Contract No. NRC-03-90-027
FIN No. L1318

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# TECHNICAL EVALUATION: REFERENCE SAFETY REPORT FOR BOILING WATER REACTOR RELOAD FUEL CENPD-300-P ABB COMBUSTION ENGINEERING NUCLEAR FUEL

## 1.0 INTRODUCTION

The topical report CENPD-300-P entitled "Reference Safety Report for Boiling Water Reactor Reload Fuel," dated November 1994 (Ref. 1) and response to requests for additional information (Ref. 2) in respect thereof were submitted to the NRC by ABB Combustion Engineering, Inc. (ABB/CE) for review. CENPD-300-P describes in general terms the ABB/CE reload fuel design and safety analysis methodology for boiling water reactors based upon analysis methodologies, from the fuel design methodology to LOCA methodology, documented in a series of other ABB submittals (Refs. 3 - 19). In addition, CENPD-300-P describes the general analysis approaches ABB/CE intends to use for reload.

The purpose of the subject topical report is for ABB/CE to provide a reference document containing its complete reload methodologies in support of licensing action, whether it be related to reload fuel, changes to the plant operating domain or equipment performance characteristics.

This evaluation report contains review findings regarding the subject topical report and its supporting documents with respect to the adequacy of ABB/CE's reload analysis methodology. Review findings are also presented regarding the adequacy, in a limited scope, of the sensitivity studies performed by ABB/CE in support of qualification of ABB's analysis approaches and techniques documented in CENPD-300-P. However, the review did not cover topics which were identified by ABB/CE as being documented in other topical reports.

## 2.0 SUMMARY OF TOPICAL REPORT

CENPD-300-P describes the reload fuel design and safety analysis process used by ABB/CE for boiling water reactors. Throughout this report ABB/CE referred to NRC approved reload analysis methodologies documented in other topical reports. In addition, ABB/CE described certain other analysis approaches and techniques, including those to compute time-dependent axial power distribution and to incorporate statistical uncertainty analysis for the determination of operating limits to account for uncertainties in system parameters.

The topical report is presented in a generic manner. The reload analysis methodologies are presented in the main body of the report. In addition, applicable computer codes are presented in Appendix A. In Appendix B the format for a plant and cycle specific reload safety analysis summary report is briefly summarized. Appendix C describes operating flexibility options which may have to be performed to demonstrate that proposed changes in the plant or in operations are acceptable. Appendix D illustrates, by sample applications to SVEA-96, how the content of the main report is intended to be used. A new method to determine transient axial power distributions is presented in Appendix E.

### 2.1 Related Topical Reports

Since CENPD-300-P refers principally to methodologies documented in other ABB/CE topical reports (Refs. 3 - 19), its acceptability is dependent upon the outcome of separate NRC review of these reports. Therefore, acceptability of models documented in these topical reports depends upon the outcome of the review of these reports. Those models are not reviewed herein.

## 3.0 EVALUATION

The topical report was assumed to contain a generic reload analysis methodology and was therefore reviewed as if its applicability was intended to be plant independent and fuel independent.

### 3.1 Reload Methodology

The reload fuel methodology consists of several analysis methodologies: mechanical, nuclear and thermal-hydraulic methodologies are employed to address different aspects of reload analysis, including fuel performance and design limits, core designs, hydraulic compatibility of fuels, determination of operating limits and performance of safety analysis. Each of these areas is separately discussed and evaluated as applicable.

#### 3.1.1 Fuel Mechanical Design

CENPD-287-P (Ref. 13) documents ABB/CE's mechanical design criteria to assure that the requirements of the Standard Review Plan are met. Therefore, acceptability of elements of this submittal related to ABB/CE's fuel design criteria is not addressed in this review. Similarly, CENPD-288-P (Ref. 14) documents the methodology to evaluate fuel assembly mechanical integrity and its effect on the reactor internals during postulated seismic and LOCA condition events. Thus acceptability of a particular fuel design with respect to structural integrity is addressed in that topical report and is not reviewed herewith.

Once a reload Reference Core is designed, ABB/CE is expected to prepare fuel mechanical design input for ABB fuel to be used in the reload design analysis (from nuclear analysis to LOCA) using NRC approved computer codes in a manner which meets the NRC approved QA procedure.

Data for the resident non-ABB fuel in a transition core are expected to be provided by the licensee.

### 3.1.2 Nuclear Design

The underlying nuclear methodology description and qualification were reviewed and approved in BR 91-402-P-A (Ref. 7).

#### 3.1.2.1 Reload Nuclear Design Methodology

The reload nuclear design methodology consists of: (i) establishment of the nuclear design bases, (ii) identification of nuclear design analyses for steady-state conditions, which include development of the Reference Core and treatment of the final loading pattern and (iii) development of an envelope for preparation of nuclear input to the other relevant computer codes for mechanical, thermal-hydraulic, anticipated operational occurrence (AOO), accident and special event analyses.

The ABB/CE nuclear design bases are established to satisfy the design bases for the applicable General Design Criteria (GDC) in 10CFR50 Appendix A. Furthermore, these bases are used for development of nuclear design acceptance limits.

A Reference Core which is intended to closely approximate the as-loaded core for the upcoming cycle is established, by an iterative procedure, to satisfy the economic objectives as well as to provide for the operational flexibility and to accommodate potential deviations discussed below. The Reference Core will be analyzed with cycle-specific safety analyses which will include determination of (1) shutdown margin requirement, (2) safety limit MCPR (SLMCPR), (3) cycle specific AOOs, (4) cycle-specific accidents and (5) special events.

A set of specific steps which ABB/CE will take to optimize the Reference Core is presented and depicted in Figure 4-1 of the topical report. Adjustments will be necessary to accommodate deviations in the initial assumptions due to end-of-cycle conditions which are different from the estimates: these may include assembly inventory, exposure due to a different cycle length, axial burnup distribution at the end of the previous cycle, and asymmetric loading of fuel assemblies.

Once the Reference Core is established, analysis will be performed to assure that parameters associated with the Reference Core satisfy the nuclear design bases and that various safety related limits are not violated.

Fuel rod power histories will be generated for the thermal-mechanical design evaluation. Conservative radial power distributions are to be computed using approved methods for input to the cycle specific safety limit MCPR (SLMCPR) calculation. For the input to the transient, accident, LOCA, CRDA and special event analyses, necessary nuclear cross-section data, various and appropriate power distributions, kinetic parameters, burn-up data, void histories, and fission product inventories will be computed using approved computer codes.

### 3.1.3 Thermal Hydraulic Design

Using a process similar to that followed in the nuclear design section, ABB/CE described the methods by which it will identify the thermal-hydraulic (TH) design bases, and the methodologies it will use to evaluate compliance with the design bases, and presented a description of TH input to the other analyses.

#### 3.1.3.1 Reload Thermal Hydraulic Design Methodology

The reload thermal hydraulic design methodology consists of (i) identification of design bases to establish thermal-hydraulic acceptance limits, (ii) evaluation of compliance with the thermal and hydraulic design bases, (iii) establishment of hydraulic compatibility of fuel, and (iv) development of input to the other analyses.

The ABB/CE thermal and hydraulic design bases are established to satisfy the design bases of the applicable General Design Criteria (GDC) in 10CFR50 Appendix A. Furthermore, these bases are used for development of thermal-hydraulic design acceptance limits.

ABB/CE developed a procedure to assure hydraulic compatibility of a new fuel with the resident fuel. This procedure, as presented, consists of very general requirements which are placed upon flow distribution and pressure differentials in the fuel assemblies. The iterative process of establishing hydraulic compatibility is presented in Section 5.3.3 of the topical report and is found to be reasonable.

Reload thermal-hydraulic design methodology begins with the development of steady-state computer models for the core and fuel assembly for a single channel and/or multi-channel analysis. Appropriate adjustments are made using NRC approved methodologies to simulate the phenomena in the appropriate areas of the core and fuel assembly. Test data for ABB/CE's reload fuel are available to support the computer model development.

The Safety Limit Minimum Critical Power Ratio (SLMCPR), to assure cladding integrity is not challenged by a boiling transition during steady-state operation and anticipated transients, is intended to be determined by a statistical combination of the uncertainties associated with the calculation of CPR. The statistical methodology used for this purpose will be discussed separately later in this evaluation report. The minimum CPR (MCPR) is computed using NRC approved codes, CPR correlations and methodology. For non-ABB fuel, information will be solicited from the utility to permit ABB/CE to adapt an accepted correlation for analysis. This "renormalization" process is discussed later in this evaluation report.

ABB/CE will perform plant and cycle specific analyses to determine the impact of the most limiting AOOs on the MCPR. The Operating Limit MCPR (OLMCPR) for each cycle and fuel type is defined in the plant licensee's Core Operating Limits Report (COLR). ABB/CE stated that, during the reload design phase, it will assure that for both ABB and non-ABB fuel the OLMCPR is satisfied during reactor operation.

Thermal-hydraulic input necessary to perform mechanical design evaluation, nuclear design analysis, transient, LOCA, CRDA and stability analysis is to be prepared using NRC approved methodologies and computer codes.

### 3.1.4 Reload Safety Analysis

The ABB/CE reload safety analysis process was described to consist of; (1) identification of generic BWR safety analysis events, (2) identification of potentially limiting events for reload, (3) development of design bases and acceptance limits, (4) analysis of plant allowable operating domain, and (5) evaluation of reload safety analyses.

The purpose of the process is to assure that: (1) all of the applicable regulatory requirements and guidance are satisfied; (2) the process is sufficiently flexible to incorporate the plant specific license commitments which potentially impact the reload safety analysis process; (3) the acceptability of the new core configuration is demonstrated to be consistent with operation in the allowable operating domain; and (4) the acceptability of plant modifications affecting the allowable plant operating domain is demonstrated through the process.

#### 3.1.4.1 Generic BWR Safety Analysis Events

Safety analysis events are categorized and potentially limiting events with respect to the plant design basis are identified. ABB/CE has identified a generic set of safety analysis events for BWRs. This set includes analyses which may not be necessary as part of reload evaluation as well as a generic set of potentially limiting events which will be evaluated at each reload. ABB/CE recognizes that there will be additional plant specific analyses which will be performed as part of reload analysis.

The generic set of analyses provides a framework in which reload analysis will be performed, in that potential single failure modes and operator actions are identified and a discussion of the impact of transient assumptions and plant parameter status is presented. This aspect was reviewed in the context that it represents a generic analysis approach and presents ABB/CE's understanding of the consequences and analysis objectives for each event. A recommendation of approval herein does not represent a generic endorsement of the ABB/CE reload safety analysis process, but rather indicates that the process is sufficiently thorough that there exists assurance that necessary analyses will be identified and performed at each reload.

#### 3.1.4.2 Potentially Limiting Events

The potentially limiting events are evaluated to establish cycle specific core operating limits. ABB/CE identified a generic minimum set of potentially limiting events which will be evaluated as part of reload analysis. The set will be supplemented with other event analysis on a plant specific basis. In the event of plant specific application, ABB/CE will evaluate all those events which are in the current plant safety analysis in order to assure that all applicable plant safety analysis events are considered.

3.1.4.3 Development of Design Bases and Acceptance Limits

In order to demonstrate conformity of the applicable event acceptance limits to the regulatory requirements and guidance, ABB/CE presented a general approach to identification of the event acceptance limits and identification of design bases accidents and applicable codes and regulatory requirements. The design bases event acceptance limits are summarized in Table 6-3 of the topical report.

ABB/CE stated that with respect to the reactor coolant boundary pressure limit ABB/CE will use 110% of the reactor pressure vessel design pressure unless there already exist other NRC limits (Ref. 2).

3.1.4.4 Plant Allowable Operating Domain

The allowable operating domain, provided by the plant licensee, serves as the basis for reload evaluation. ABB/CE will assure (Ref. 2), at each reload, that all necessary analyses covering all potentially limiting events and those necessitated by the licensee's requirement for operating flexibility options will be performed within the allowable operating domain. Any deviation from that domain will be treated as a plant modification and will be evaluated and submitted to the NRC for review.

3.1.4.5 Reload Safety Analysis Methodology

The reload safety analysis is performed to establish the operating limit MCPR and to demonstrate compliance to the event acceptance limits for several key accidents. The reload analysis is performed using the assumed Reference Core design. ABB/CE is required to use only computer codes and methodologies approved by the NRC.

3.1.5 Event Analysis Methodologies

In this section, ABB/CE's methodologies for performance of safety analyses are discussed and evaluated.

3.1.5.1 Anticipated Operational Occurrences

A generic set of potentially limiting AOO events is established. For this purpose, the bounding events are identified and analyzed based upon the transient characteristics (fast versus slow). On a transient-by-transient basis, ABB/CE presented analysis description, methodology and computer codes to be used.

Slow transients are analyzed as quasi-steady-state events, while fast transients are analyzed using transient methodologies. Intermediate transient are analyzed first using both techniques (Ref. 2). Then ABB/CE will select the method which results in conservative predictions.

For fast transients, analysis is provided in two steps: plant system response calculations and separate hot channel calculations driven by boundary conditions established from the system analysis. ABB/CE stated that in the hot channel analysis the use of the time-varying axial power shape distribution will result in more conservative predictions than does the use of a constant power shape. A limited amount of demonstration analysis was provided (Ref. 2) which supported ABB/CE's position. However, ABB/CE did not provide any description of underlying phenomena to support its position; therefore, the use of time-varying axial power distribution is acceptable for analysis of fast pressurization transients provided that its use is demonstrated to be conservative.

With the foregoing exception, transient analysis descriptions, assumptions and methodologies with variations for different BWR designs presented in the topical report are reasonable. Actual analysis approaches must be justified for each reload if they differ from the current plant specific FSAR approaches.

3.1.5.2 Determination of Operating Limits

The operating limit MCPR is the core operating limit to assure that there exists an adequate margin to boiling transition during normal operation and anticipated operational occurrences, and is composed of two parts: one part is established for the steady-state (SLMCPR) AOOs, and the other part is established for the dynamic state (delta CPR) with the calculation of the critical power ratio (CPR).

All potentially limiting AOO events are ordinarily analyzed for the limiting plant conditions accounting for uncertainties in the analysis computer codes, plant model input and plant operating state inputs by selecting these inputs at their limiting conditions to assure conservatism in the analysis results. However, analysis performed in this manner may result in establishment of operating limits which are more restrictive than those desired by the licensee. In that event, ABB/CE expects to examine uncertainties more closely in the reload analysis methodology. In those cases, ABB/CE plans to perform more realistic analysis using statistically combined uncertainties in inputs as discussed below.

ABB/CE presented four methods for the purpose of establishing operating limits. The first method, the most conservative method (referred to as Method A), is the traditional approach in which the operating limit is established in a deterministic fashion with each parameter at its own worst value. The second approach, Method B, is a simplified statistical analysis in which a series perturbation cases are run where certain of the key parameters are treated as variables with uncertainties considered to be in a normal distribution within a certain range about the nominal values. The results of differences are combined in the square root sum of the square. Inherent in this formulation is that the perturbed parameters are independent and the uncertainties are normally distributed.

In the third method, Method C, a response surface, defined by a polynomial equation, is developed with key parameters whose uncertainties are considered. In both Methods B and C, non-perturbed parameters are assumed at their conservative values. Once the response surface equation is determined, propagation of uncertainties using assumed probability distributions is performed using a sufficiently large number of Monte Carlo cases.

The fourth method, Method D, also utilizes the Month Carlo approach. However, ABB/CE stated that this method will not be used in licensing analysis (Ref. 2), and therefore it was not reviewed and is restricted from use until fully described and reviewed.

Although the methodology can be generically applicable to any fuel or plant, determination of which parameters are the key parameters and their probability distribution functions will vary from plant to plant. Therefore, in each of the statistical approaches, ABB/CE needs to provide justification at each reload application. In method A, ABB/CE needs to justify that the conservative values assumed for analysis are conservative enough.

In Method B, ABB/CE must: (1) justify its selection of key parameters; (2) demonstrate that the data base is large enough to obtain statistical significance and justify the applicability of normal distribution; and (3) justify the assumption that the input parameters are independent. Demonstration that the parameters are independent may not be achievable, in which case it would be acceptable for ABB/CE to demonstrate that the assumption of non-independent parameters produces conservative predictions.

For the use of the response surface method (Method C), the manner in which the surface is developed can define a range of applicability. ABB/CE must (1) justify the selection of parameters used in the development of the response surface, and (2) assure that the range of uncertainties used in the method encompasses the full operating range. Furthermore, depending upon the number of parameters considered in the development of the response surface, ABB/CE needs to provide justification of development of the composite design. In addition, since for Monte Carlo runs ABB/CE will assume a probability distribution function for each parameter, ABB/CE needs to justify the choice of distribution function selected by examination of the quality and quantity of data as discussed above.

For slow transient analysis, the OLMCPR is determined in the conventional deterministic manner. However, should this limit become too restrictive, ABB intends to select, on an optional basis, any one of the other approaches to obtain the limit. For fast transient analysis, ABB/CE intends to use one of the statistical approaches. Any such use is subject to all the aforesaid conditions.

The operating limit MCPR is set to be the most limiting value obtained from analysis of all AOO events and from analysis of the misplaced assembly accident and computed for full power operation throughout the range of allowable core flows and cycle burn-ups. However, since some AOOs are more restrictive at off-rated conditions, the OLMCPR may require adjustments for those conditions, typically at reduced power and flows. ABB/CE will perform additional calculations to justify any additional constraints necessary to accommodate these situations.

Operating Limit LHGR

The operating limit linear heat generation rate (LHGR) is established for each fuel type in a given reload cycle such that thermal/mechanical fuel limits are met during steady state operation and AOOs. In combination with the maximum average planar linear heat generation rate (MAPLHGR) limits (determined by the LOCA design bases), the operating limit LHGR helps ensure that compliance to the specified acceptance fuel design limits (SAFDLs) is maintained under all design basis conditions. The LHGR operating limit is generally exposure dependent. Adjustments may be necessary to accommodate off-rated operating conditions depending upon the specific fuel design and the plant allowable operating domain. ABB/CE will perform additional calculations to justify any additional constraints necessary to accommodate these situations.

3.1.5.3 Accident Analysis

ABB/CE identified four groups of accidents which generically require re-evaluation in the reload safety analysis process: (1) LOCA, (2) CRDA, (3) Fuel Handling Accident and (4) Misplaced Bundle Accident.

With respect to LOCA analysis, ABB/CE described the process of satisfying the design basis acceptance criteria established in 10CFR50 Appendix K. ABB/CE's Appendix K ECCS Evaluation Model has been reviewed and approved (Refs. 3 - 6). Parametric studies using the approved Appendix K ECCS Evaluation Model will be performed to identify the design basis event. Furthermore, analysis will be performed to determine the total hydrogen generation and the MAPLHGR operating limit to ensure compliance with the LOCA design bases.

In Appendix D and in Reference 2 ABB/CE stated its intent to use the ANS79 decay heat curve in place of ANS71 data in its ECCS analysis. Although it is recognized that there exists a large degree of conservatism in ANS71 data, its use is required in the Appendix K ECCS Evaluation Model. The use of ANS79 is acceptable in a best-estimate LOCA model; however, the discussion of LOCA analysis in the subject topical report is limited to the use of the Appendix K Evaluation Model. Therefore, if ABB/CE wishes to develop a best-estimate LOCA model with ANS79 decay heat data, ABB/CE should document and submit a full description of the model for NRC review. However, ABB/CE may not use ANS79 for Appendix K ECCS Evaluation Model.

CRDA analysis methodology is documented in CENPD-284-P (Ref. 16). Therefore, its acceptability should be determined as part of its review.

The Fuel Handling Accident is analyzed with respect to GDC19 and 10CFR100. The focus of the analysis is to determine if the existing analysis is bounding. However, ABB/CE asserts that the ABB fuel type has design advantage with respect to this type of accident relative to non-ABB fuel types. ABB/CE presented two approaches to estimating fission product inventories in the new fuel: one approach is simplistic in that estimates of fission product inventories are determined based upon the relative operating characteristics experienced by the two types of fuel assemblies, while the other more detailed approach is based upon the fission gas release assumptions in Regulatory Guide 1.25 in conjunction with the use of approved codes. ABB/CE stated that the results of the simplistic approach are more restrictive and that when the releases predicted with the simplistic method do not satisfy the event acceptance limits, more detailed analysis using the second method will be performed.

Both the mislocated and rotated fuel assembly accidents will be analyzed with respect to the SLMCPR limit. ABB/CE indicated that sensitivity studies will be performed to identify the limiting scenarios in both of these analyses.

In the case of rotated fuel assembly analysis, ABB/CE indicated the possibility that analysis using a constant interassembly gap size may result in determination of core operating limits which are more restrictive than the licensee may desire. In such instances, ABB/CE stated its intent to vary gap sizes to reduce the conservatism in the analysis accompanied by uncertainty analyses to establish the impact. The acceptability of this approach depends upon the validity of uncertainty analysis performed. When this (use of different gap size) becomes an issue, ABB/CE should submit a well detailed justification to the NRC for review.

#### 3.1.5.4 Special Events Analysis

Four special events are identified in the ABB/CE reload safety analysis process: (1) core thermal-hydraulic stability, (2) reactor overpressurization protection, (3) standby liquid control system capacity, and (4) anticipated transients without scram (ATWS). The first three events will be analyzed for each ABB/CE reload application while the ATWS is to be evaluated as necessary.

The ABB/CE frequency domain based stability methodology was approved in RPA-90-91-P-A (Ref. 17). The advanced stability analysis methodology based upon the time domain approach is documented in CENPD-295-P (Ref. 18) using the codes described in CENPD-294-P (Ref. 19). Therefore the acceptability of overall ABB/CE stability analysis methodology should be determined as part of the review of those topical reports.

Both the overpressurization MSIV closure event and the standby liquid control system capability will be analyzed with respect to meeting the respective applicable design limits and regulatory guidance.

ABB/CE is required to perform ATWS evaluation using NRC approved methodologies and computer code and to provide a full analysis description.

### 3.2 Operational Flexibility

Appendix C of the topical report presents additional safety analyses which may be required upon reload due to the fact that the licensee's existing required set of analyses is larger than ABB/CE's generic set of reload analyses, or for the reason of gaining operating flexibility.

ABB/CE does possess methodologies to perform analyses in this category, provided that all pending topical reports are found to be acceptable. However, this is highly plant specific: therefore, the adequacy and applicability of particular analyses can only be addressed at the time of reload application. Therefore, ABB/CE must provide for NRC review, at that time, full details of selected methodologies, computer codes, input preparation and assumptions, as well as justification of the extension. More specifically, ABB/CE should be required to demonstrate that: (1) the use of the standard reload analysis methodology is applicable at the extreme points on the extended allowable operating domain; (2) none of the SER conditions on use of relevant computer models are violated; and (3) the method used in determination of the core operating limits remains in the range of applicability; i.e., that there exist a sufficient (statistically meaningful) amount of data in support of the limiting points on the extended operating domain. Furthermore, any extension to the approved allowable operating domain must be approved by the NRC prior to use.

### 3.3 Transition Core

In analysis of a transition core, ABB/CE stated that all relevant analysis will be performed for both ABB and non-ABB fuel. Necessary data, including NRC approved CPR correlations, is expected to be provided by the licensee based upon the previous cycle's reload fuel. In the event that ABB/CE does not have access to the CPR correlation for particular non-ABB fuel, ABB/CE will "renormalize" for application to that fuel an NRC approved CPR correlation available to ABB/CE. This process includes data fitting by ABB/CE using its correlation to the computed data over the range of relevant conditions furnished by the licensee.

ABB/CE stated that it does not expect that the renormalized correlation will be relevant in determination of the delta CPR for the limiting assembly in the core (Ref. 2). While it is acceptable to use such a renormalized CPR correlation for analysis of non-ABB fuel in a non-limiting location, should a non-ABB fuel assembly be placed in the limiting location or used for determination of the operating limits, ABB must submit to the NRC for approval its demonstration of the applicability of the renormalized CPR correlation for that particular fuel.

### 3.4 Reload Safety Analysis Report Format

In Appendix B of the topical report, ABB/CE presented a representative format it intends to use in preparation of the reload safety analysis report (RSAR). The format described is only an outline of the RSAR. Therefore, at the time of preparation of any report, ABB/CE should include not only the items presented in Appendix B but also the relevant technical information in sufficient detail so that each topic is well substantiated. Any deviation from the current status, be it a change in the allowable operating domain due to reload, a plant modification or any other change, should be justified and accompanied by thorough analysis description including the input, transient assumptions and results.

## 4.0 CONCLUSIONS

ABB/CE's topical report "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P, dated November 1994 and supplemental information provided by the vendor in support of its submittal were reviewed. Some aspects of the reload methodology rely upon the outcome of review of other topical reports under separate review. Therefore, those models were not reviewed and their acceptability should be determined as part of the other ongoing reviews.

We find that ABB/CE has adequately described its BWR reload methodology subject to the limitations and restrictions described below:

1. Acceptability of this topical report is subject to review findings of the other relevant topical reports cited in the topical report, and all conditions set forth therein are applicable to this topical report. Furthermore, acceptability of reload analysis is subject to conditions cited in methodology topical reports.

2. When performing analyses, ABB/CE is required to identify and justify any changes from the existing set of licensing safety analysis assumptions and basic methodologies. The defined allowable operating domain should be limited to the existing approved set unless a change necessitated by plant modifications has been fully justified and supported by sensitivity analysis results.

3. Conceptually, the statistical uncertainty approach is acceptable. However, its acceptability is highly application dependant and therefore ABB/CE's uncertainty analysis approach is not generically acceptable. As stated in Section 3.1.5.2 of this report, before any of these methods is used, ABB/CE must justify the following for the parameters selected for statistical treatment: (1) those parameters are independent and uniformly distributed; (2) the range of applicability is not violated; (3) each selected probability function is adequately conservative and well supported by actual applicable data; and (4) the database is statistically significant.

4. As to the use of a time dependent axial power distribution, although limited comparative analyses were provided, discussion of the methodology was not complete. Therefore, the generic use of the time-dependant axial power distribution for analysis of fast pressurization transient is restricted until further qualification demonstrating that its use will result in conservative predictions for all conditions expected to be encountered.

   However, the use of ABB/CE's methodology for analysis of fast pressurization transients is otherwise acceptable provided that the use of the time-varying axial power distribution is demonstrated on a plant-by-plant basis to result in more conservative analysis than the constant axial power distribution.

5. This review was performed with the understanding that non-ABB fuel will not be located in the limiting positions of the core so that if ABB/CE has no access to the CPR correlation for that particular fuel, its use of a renormalized CPR correlation will not impact the calculation of safety limits. However, if this is not the case, the use of a renormalized CPR correlation is restricted unless it has been demonstrated to produce adequately conservative results.

6. The use of ANS79 decay heat curve is acceptable only for best-estimate LOCA analysis and not for the Appendix K ECCS evaluation model. For compliance with Appendix K, ABB/CE should continue to use 1.2 times the ANS71 as stated in the current 10CFR50 Appendix K. However, if ABB/CE desires to obtain approval of a best-estimate LOCA model, ABB/CE must submit a full description and qualification for NRC review.

7. In the qualification analysis of the ABB/CE-developed vessel water level algorithm provided in Reference 2, ABB/CE stated that the method is able to predict the level relatively well as long as the level is changing slowly. The use of this water level algorithm is covered by the condition cited in the SER on BISON regarding the use of the control systems. ABB/CE should demonstrate that its use results in conservative prediction at each reload.

8. Input must be prepared and analysis performed using NRC approved compute codes and in a manner consistent with governing approved topical reports; specifically, correlations and models must be used within their ranges of applicability and subject to respective SER conditions. Additionally, data preparation and documentation should be made in accordance with the NRC approved QA procedure.

9. No evaluation of validity of sample analyses was performed. Furthermore, the approval recommended in this report does not imply any endorsement of analyses nor of the quantified uncertainties set forth in Appendix D. Therefore, no reference should be made to Appendix D as demonstration in support future reload.

10. At the minimum, each RSAR should contain all the items referred to in Appendix B of the topical report. However, since there are several application-dependent items ABB/CE must provide and justify, it is recommended that the cycle-specific Reload Safety Analysis Summary Report be written in such a way to convey the technically significant results due to the reload, with particular emphasis in sufficient detail on differences and changes introduced, in core performance between the existing and new cycle and their impact on transient/accident analysis and operating safety limits.

11. ABB/CE must use 110% of vessel design pressure for the peak reactor vessel pressure limit unless otherwise governed by another pre-existing NRC approved limit.

Finally, it appears that ABB/CE's licensing computer codes may not be maintained in a manner consistent with 10CFR50 Appendix B and Regulatory Guide 1.64, specifically with respect to administrative control of the computer codes. Therefore it is recommended that a QA audit be performed to assure that the control of computer codes and associated documentation are acceptable and the code inputs are prepared and maintained acceptably.

## 5.0 REFERENCES


1. "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P, November 1994.

2. "Reference Safety Report for Boiling Water Reactor Reload Fuel: Response to Request for Additional Information," CENPD-300-P-RAI, Rev. 1, August 1995.

3. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 1 to Code Description and Qualification," CENPD-293-P, August 1994.

4. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," RPB 90-93-P-A, October 1991.

5. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel," CENPD-283-P, March 1993.

6. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," RPB 90-94-P-A, October 1991.

7. "ABB Atom Nuclear Design and Analysis Programs for Boiling Water Reactors: Programs Description and Qualification," BR 91-402-P-A, May 1991

8. "BISON - One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification," CENPD-292-P, August 1994.

9. "BISON - One Dimensional Dynamic Analysis Code for Boiling Water Reactors," RPA-90-90-P-A, December 1991.

10. "CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors," BR-91-255-P-A, Rev. 1, May 1991.

11. "SVEA-96 Critical Power Experiments on a Full Scale Sub-bundle," ABB Atom Report UR 89-210-P-A, October 1993.

12. "Fuel Rod Design Methods for Boiling Water Reactors," CENPD-285-P, May 1994.

13. "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors," CENPD-287-P, June 1994.

14. "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel," CENPD-288-P, May 1994.

15. "ABB Atom High Worth Control Rods for US BWRs, Rod Drop Accident Analysis," RPA-89-053, August 1989.

16. "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," CENPD-284-P, October 1993.

17. "NUFREQ-NPW: An ABB Atom Computer Code for Core Stability Analysis of Boiling Water Reactors," RPA-90-91-P-A, December 1991.

18. "Thermal-Hydraulic Stability Analysis Methodology for Boiling Water Reactors," CENPD-295-P, to be submitted to the NRC for review.

19. "Thermal-Hydraulic Stability Analysis Methods for Boiling Water Reactors," CENPD-294-P, to be submitted to the NRC for review.

# CENPD-300-NP-A REPORT

## Part II

## Body of Report

ABB Combustion Engineering Nuclear Operations

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# LIST OF ACRONYMS

| | |
|---|---|
| 1D | One-Dimensional |
| 3D | Three-Dimensional |
| AOO | Anticipated Operational Occurrences |
| APRM | Average Power Range Monitor |
| ARO | All Rods Out |
| ASME | American Society of Mechanical Engineers |
| ATWS | Anticipated Transient Without Scram |
| BOC | Beginning of Cycle |
| BOLFFP | Beginning of Low Flow and Full Power |
| CPR | Critical Power Ratio |
| CRDA | Control Rod Drop Accident |
| EFPH | Effective Full Power Hours |
| ELLLA | Extended Load Line Limit Analysis |
| EOC | End of Cycle |
| EOFP | End of Full Power |
| EOLFFP | End of Low Flow and Full Power |
| FFWTR | Final Feedwater Temperature Reduction |
| FGR | Fission Gas Release fraction |
| FMCPR | Final Minimum CPR |
| FRAD | Radial Power Form Factor |
| FRSC | Failure of RHR Shutdown Cooling |
| FSAR | Final Safety Analysis Report |
| FWCF | Feedwater Controller Failure |

LIST OF ACRONYMS (Continued)

| | |
|---|---|
| gap HTC | Heat Transfer Coefficient between pellet and cladding |
| GLRNB | Generator Load Rejection No Bypass |
| GLRWOB | Generator Load Rejection Without Bypass |
| GLRWB | Generator Load Rejection With Bypass |
| ICF | Increased Core Flow |
| ICPR | Initial CPR |
| IMCPR | Initial Minimum CPR |
| LHGR | Linear Heat Generation Rate |
| LOAP | Loss of Auxiliary Power |
| LOCV | Loss of Condenser Vacuum |
| LOFH | Loss of Feedwater Heating |
| LOOP | Loss of Offsite Power |
| LPRM | Local Power Range Monitor |
| MAPLHGR | Maximum Average Planar Linear Heat Generation Rate |
| MCPR | Minimum CPR |
| MELLLA | Maximum Extended Load Line Limit Analysis |
| MEOD | Maximum Extended Operating Domain |
| MOC | Middle of Cycle |
| MSIV | Main Steam Isolation Valve |
| MSIVC | Main Steam Isolation Valve Closure |
| NBR | Nuclear Boiler Rated |

LIST OF ACRONYMS (Continued)

| | |
|---|---|
| OL | Operating Limit |
| OLMCPR | Operating Limit MCPR |
| PCI | Pellet Clad Interaction |
| RBM | Rod Block Monitor |
| RCPR | ΔCPR/ICPR |
| RPR | Recirculation Pump Runup |
| RPT | Recirculation Pump Trip |
| RWE | Rod Withdrawal Error |
| SAFDL | Specified Acceptable Fuel Design Limit |
| SLCS | Standby Liquid Control System |
| SLMCPR | Safety Limit MCPR |
| TBV | Turbine Bypass Valve |
| TCV | Turbine Control Valve |
| TMOL | Thermal-Mechanical Operating Limit |
| TSV | Turbine Stop Valve |
| TT | Turbine Trip |
| TTMOL | Transient TMOL |
| *UNC* | Uncertainty |
| ΔCPR | CPR variation during a transient |
| ΔMCPR | Minimum CPR variation during a transient |

# 1 INTRODUCTION

ABB Combustion Engineering (ABB) is a supplier of reload fuel in the United States. This Reference Safety Report (RSR) for boiling water reactor (BWR) reload fuel describes the reload fuel design and safety analysis process used in specific plant applications. Specific topics related to the ABB BWR reload fuel design and safety analysis methodology are contained in numerous Licensing Topical Reports describing portions of the overall methodology. This RSR integrates all the separate reports into a single comprehensive reload fuel design and safety analysis methodology. Between the contents of the separate Licensing Topical Reports and contents of this RSR the code methods, code qualification, design bases, methodology, and sample applications are described for all fuel design and safety analyses performed in support of plant modifications requiring a safety evaluation of the fuel, core, reactor coolant pressure boundary, or containment systems, including BWR reload fuel applications.

## 1.1 Background

The United States licensing of the ABB BWR reload fuel safety methodology started in 1982 with the submittal of Licensing Topical Reports (References 1 through 11) by Westinghouse Electric Corp. describing code methods and methodology developed by ABB Atom (formerly ASEA Atom) of Sweden. Many of these report were reviewed and approved by the U.S. NRC (References 12 through 18). In 1988, ABB Atom continued the licensing of the ABB BWR reload methodology, started by Westinghouse, directly with the NRC. The transfer of the licensing effort was formally facilitated by ABB resubmitting NRC approved Licensing Topical Reports under the ABB ownership (References 19 through 25), and the NRC acknowledged the transfer of the Licensing Topical Reports approvals (Reference 26). In the ongoing effort to license a complete BWR Reload methodology, ABB Atom submitted several additional Licensing Topical Report (References 27 through 29). As a result of the acquisition of Combustion Engineering, Inc. by the parent company of ABB Atom, the U.S. operations of ABB Atom were consolidated within Combustion Engineering, Inc. (Reference 30). The ABB Combustion Engineering Nuclear Operations Division of Combustion Engineering, Inc. is the cognizant origination for BWR reload fuel application in the United States. Quality control, maintenance, and implementation for the complete ABB U.S. reload fuel licensing methodologies resides with ABB Combustion Engineering Nuclear Operations.

Subsequent to the consolidation, the NRC has issued approval for one ABB Licensing Topical Report (References 31 and 32), and several additional Licensing Topical Reports have been submitted to the NRC for review (References 33 through 41). This document, the "Reference Safety Report for BWR Reload Fuel" integrates the ABB BWR reload methodology intended to be used for ABB U.S. reload and plant operational modification applications.

## 1.2 BWR Reload Licensing Documents

The ABB BWR reload fuel safety analysis methodology is contained in a series of Licensing Topical Reports. Each report describes for one or more disciplines the code methods, code qualification, design bases, analysis methodology and/or sample applications. Table 1-1 summarizes the Licensing Topical Reports comprising the overall reload methodology. Table 1-2 identifies the scope of each report and the discipline(s) it covers.

## 1.3 Report Overview

This document describes the ABB reload fuel safety analysis methodology for boiling water reactors. The structure of this RSR is shown on Figure 1-1. Section 2 provides a summary of the report purpose, content, and conclusions. The reload fuel and core design process are discussed in Sections 3, 4, and 5. The fuel thermal-mechanical design process is described in Section 3. The fuel and core nuclear design is described in Section 4. The thermal-hydraulic design is described in Section 5. Emphasis in the reload fuel and core design process is placed on the inputs and interfaces of the design with the reload safety analysis. The reload safety analysis methodology is discussed in the remainder of the report. Section 6 provides an introduction to the required safety analyses for a reload fuel or plant operational modification. Section 7 presents the transient analysis methodology for anticipated operational occurrences (transient analyses), with Appendix E providing qualification of the fast transient analysis methodology. Section 8 presents the methodology for accident analysis, specifically: loss of coolant accident, control rod drop accident, fuel handling accident, and fuel loading errors. Finally, Section 9 discusses Special Events addressed in the reload fuel safety analysis i.e., thermal-hydraulic stability, reactor vessel overpressure protection, standby liquid control system performance, and anticipated transients without scram.

Appendix A to this report provides a brief description of the computer codes used in ABB reload analysis methodology. Appendix B provides an example of a cycle specific Reload Safety Analysis Summary Report. Appendix C outlines how the differing plant operational features and analysis options are addressed in the overall ABB reload analysis methodology. Appendix D provides illustrations of the ABB reload fuel design and safety analysis methodology described in this document by presenting sample applications. Several different application examples are used to best illustrate the generic methodology for each discipline.

## TABLE 1-1

## ABB BWR RELOAD FUEL LICENSING TOPICAL REPORTS

| Report Number | Report Title | Discipline |
|---|---|---|
| CENPD-285-P-A | Fuel Rod Design Methods for Boiling Water Reactors | Mechanical |
| CENPD-287-P-A | Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors | Mechanical |
| CENPD-288-P-A | ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel | Mechanical |
| BR 91-402-P-A | ABB Atom Nuclear Design and Analysis Programs for Boiling Water Reactors: Programs Description and Qualification | Nuclear |
| BR 91-255-P-A, Rev. 1 | CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors | Thermal-Hydraulic |
| UR 89-210-P-A | SVEA-96 Critical Power Experiments on a Full Scale 24-rod Sub-Bundle | Thermal-Hydraulic |
| RPA 90-90-P-A | BISON - A One Dimensional Dynamic Analysis Code for Boiling Water Reactors | AOO: Fast Transients |
| CENPD-292-P-A | BISON - One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification | AOO: Fast Transients |
| RPB 90-93-P-A | Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification | Accidents: LOCA |
| RPB 90-94-P-A | Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity | Accidents: LOCA |
| CENPD-293-P-A | Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 1 to Code Description and Qualification | Accidents: LOCA |
| CENPD-283-P-A | Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel | Accidents: LOCA |
| CENPD-284-P-A, RPA 89-112-A, and RPA 89-053-A | Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification | Accidents: CRDA |
| RPA 90-91-P-A | NUFREQ-NPW: A Computer Code for Core Stability Analysis of Boiling Water Reactors | Special Events: Stability |
| CENPD-294-P-A | ABB Advanced Stability Methods for Boiling Water Reactors | Special Events: Stability |
| CENPD-295-P-A | ABB Advanced Stability Methodology for Boiling Water Reactors | Special Events: Stability |
| CENPD-300-P-A | Reference Safety Report for Boiling Water Reactor Reload Fuel | Reload Analysis |

## TABLE 1-2

## LICENSING TOPICAL REPORT SCOPE

| Discipline | Design Bases and Methodology | Code Methods | Qualification | Application |
|---|---|---|---|---|
| Mechanical | CENPD-287-P-A (Normal Operation/AOO)<br><br>CENPD-288-P-A (Accidents) | CENPD-285-P-A (Fuel Rod)<br><br>CENPD-287-P-A (Fuel Assembly) | CENPD-285-P-A | CENPD-287-P-A |
| Nuclear | CENPD-300-P-A | BR 91-402-P-A | BR 91-402-P-A | CENPD-300-P-A |
| Thermal-Hydraulic | CENPD-300-P-A | BR 91-255-P-A, Rev. 1<br><br>UR 89-210-P-A (CPR Correlation) | BR 91-255-P-A, Rev. 1<br><br>UR 89-210-P-A (CPR Correlation) | CENPD-300-P-A |
| AOO: Fast Transients | CENPD-300-P-A | RPA 90-90-P-A<br><br>CENPD-292-P-A | | CENPD-300-P-A |
| AOO: Slow Transients | CENPD-300-P-A | BR 91-402-P-A | BR 91-402-P-A | CENPD-300-P-A |
| Accidents: LOCA | RPB 90-94-P-A<br><br>CENPD-283-P-A<br><br>CENPD-300-P-A | RPB 90-93-P-A<br><br>CENPD-293-P-A | RPB 90-93-P-A | CENPD-300-P-A |
| Accidents: CRDA | CENPD-284-P-A | BR 91-402-P-A | CENPD-284-P-A | CENPD-284-P-A |
| Accidents: Others | CENPD-300-P-A | CENPD-300-P-A | CENPD-300-P-A | CENPD-300-P-A |
| Special Events: Stability | CENPD-300-P-A (Current)<br><br>CENPD-295-P-A (Advanced) | RPA 90-91-P-A (NUFREQ-NPW)<br><br>CENPD-294-P-A (RAMONA-3) | RPA 90-91-P-A (NUFREQ-NPW)<br><br>CENPD-294-P-A (RAMONA-3) | CENPD-300-P-A (Current)<br><br>CENPD-295-P-A (Advanced) |
| Special Events: Overpressure Protection | CENPD-300-P-A | CENPD-300-P-A | CENPD-300-P-A | CENPD-300-P-A |

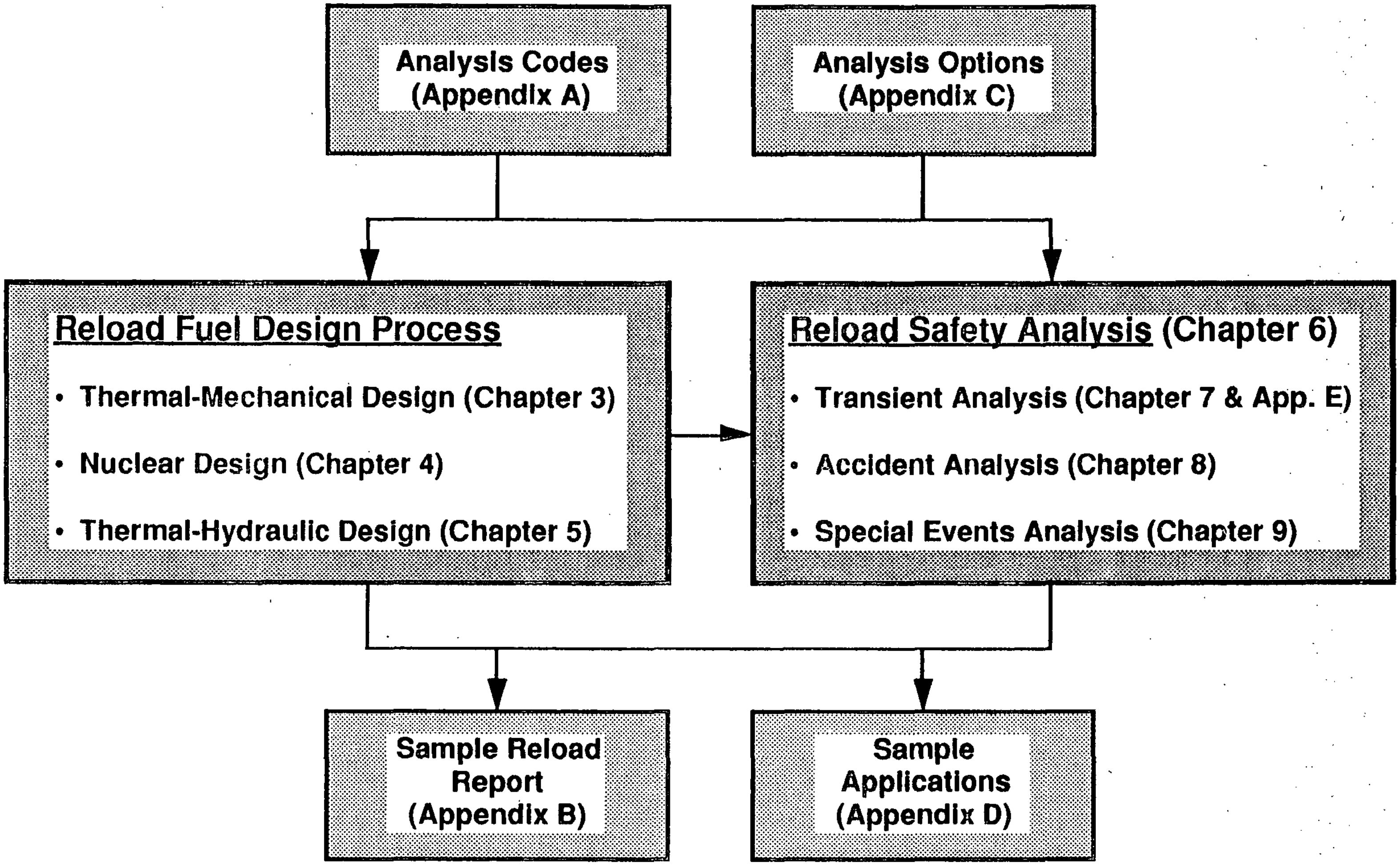



**Figure 1-1. Reference Safety Report Structure**

# 2 SUMMARY AND CONCLUSIONS

## 2.1 Summary

This Reference Safety Report (RSR) for boiling water reactor (BWR) reload fuel describes the reload fuel design and safety analysis process used by ABB in specific plant applications. The objective of the reload fuel design process is to provide a reload fuel and core design, consistent with the utility energy utilization plan, that will reliably satisfy the operational objectives of the plant. The objective of the reload fuel safety analysis is to demonstrate that the plant using the reload fuel and core design can operate without undue risk to the health and safety of the public. To satisfy these two primary objectives, ABB has developed a single highly interrelated process for reload fuel applications that covers all of the required subjects for the reload fuel design and safety analysis.

Consistent with the reload fuel design process, this RSR has separated the discussion of the process into the three key disciplines: (1) thermal-mechanical (Section 3); (2) nuclear; (Section 4) and (3) thermal-hydraulic (Section 5). The thermal-mechanical design discussion includes the fuel assembly and fuel rod performance analyses, the definition of the specified acceptable fuel design limits, and the identification of the control rod insertability and core coolability requirements. The nuclear design discussion includes a description of the process used to determine the number and enrichment of the reload fuel assemblies, the development of a realistic nuclear core model that can be utilized for core follow and support, the methodology used to develop the reference core loading pattern and target control rod sequences, and the development of the nuclear parameters. The thermal-hydraulic design discussion includes the methodology for establishing the minimum critical power ratio safety limit, the analysis process for demonstrating hydraulic compatibility between the reload fuel and resident fuel assemblies, and the development of the thermal-hydraulic design parameters. For each of these disciplines, the applicable design bases and criteria, analysis methodology, and inputs to the other design disciplines and safety analysis are described.

Consistent with the reload safety analysis process, this RSR has separated the discussion of the process into an overview (Section 6) and the analysis of the three categories of safety analysis events: (1) anticipated operational occurrences or transients (Section 7); (2) accidents; (Section 8), and (3) special events (Section 9). Anticipated operational occurrences are those conditions of normal operation which are expected to occur one or more times during the life of the plant and include but are not limited to generator load rejection, turbine trip, isolation of the main condenser, and loss of feedwater heating. Accidents are those postulated events that potentially affect one or more of the barriers to the release of radioactive materials to the environment. These events are not expected to occur during the plant lifetime, but are used to establish the design basis for many systems. Special events are postulated occurrences that are analyzed to demonstrate different plant capabilities required by the regulatory requirements and guidance, industry codes and standards, and licensing commitments applicable to the plant. For the potentially limiting events in each of the event categories, the applicable design bases and evaluation methodology are described.

Specific topics related to the ABB BWR reload fuel design and safety analysis methodology have been provided in individual Licensing Topical Reports. These individual Licensing Topical Reports have been the subject of independent regulatory review and approval. The status of these individual Licensing Topical Reports is not impacted by the information contained in this RSR, and it is not considered necessary to re-review the information contained in previously approved Licensing Topical Reports. This RSR provides an integrated summary of the applicable parts of the separate reports in a single comprehensive reload fuel design and safety analysis methodology and describes how the individual methodologies are applied in the reload fuel design and reload safety analysis process. It is the application of these methodologies that is considered unique to this RSR and subject to regulatory authority approval.

## 2.2 Conclusions

Based on the information provided in this report, it is concluded that:

(1) The ABB reload design and safety analysis process and methodology satisfies all of the applicable regulatory requirements and is consistent with regulatory requirements and guidance.

(2) The ABB reload fuel design and safety analysis methodology is sufficiently flexible to be applied to the spectrum of BWR plant types and can satisfy the plant specific license commitments.

(3) The ABB reload fuel design and safety analysis methodology can be used to demonstrate the acceptability of a plant operating with ABB reload fuel in a new core configuration consistent with operation in the allowable operating domain.

(4) The ABB reload fuel design and safety analysis methodology can be used to demonstrate the acceptability of plant modifications affecting the allowable plant operating domain.

(5) The ABB reload fuel thermal-mechanical design satisfies applicable regulatory requirements and guidance, including

the identification of the specified acceptable fuel design limits of General Design Criteria (GDC) 10 (Reference 42, 10CFR50 Appendix A), the rod insertability requirements of GDC 27, the core coolability requirements of GDC 35, and the fuel thermal-mechanical acceptance requirements identified in Standard Review Plan, Section 4.2 (Reference 43).

(6) The ABB reload fuel nuclear design satisfies the applicable regulatory requirements and guidelines, including those identified in the applicable General Design Criteria (Reference 42) and Section 4.3 of the Standard Review Plan (Reference 43).

(7) The ABB reload fuel thermal-hydraulic design satisfies the applicable regulatory requirements and guidelines, including those identified in the applicable General Design Criteria (Reference 42) and Section 4.4 of the Standard Review Plan (Reference 43).

(8) The ABB reload fuel safety analysis methodology has established appropriate design bases for the evaluation of all events considered a part of the plant safety analysis.

(9) The ABB safety analysis methodology can be applied to the analysis of anticipated operational occurrences, accidents, and special events to demonstrate compliance with applicable design bases and to establish the acceptable core operating limits.

Therefore, the ABB reload safety analysis methodology can be used to update the current plant safety analysis consistent with the requirements of 10CFR50.59 (Reference 42).

# 3 MECHANICAL DESIGN

The fuel assembly and fuel rod mechanical design bases and methodology are described in Reference 37 and are, therefore, not repeated in this document. Therefore, this section describes the mechanical design and fuel rod performance data provided to disciplines supporting the reload design and safety analysis methodology.

## 3.1 Summary

The ABB methodology for the fuel assembly and fuel rod mechanical evaluation identified in Section 4.2 of the Standard Review Plan, NUREG-0800 (Reference 43) is provided in Reference 37. Specifically, Reference 37 contains mechanical design criteria which assure that the requirements of Reference 43 are satisfied, the methodology for performing mechanical design evaluations relative to those criteria, and an application of that methodology to the ABB SVEA-96 fuel assembly which demonstrates that the SVEA-96 assembly satisfies the design criteria. Therefore, this information is not repeated in this document. The interface between the mechanical design and other disciplines supporting the reload design methodology is not described in Reference 37 and is, therefore, discussed in this section.

This section provides the interface between the mechanical design of ABB fuel and the other design activities. Specifically, the type of mechanical data provided to the nuclear, thermal-hydraulic, and safety analysis processes, as well as the methodologies for determining that data, are provided as required. For example, the methods used to establish the fuel rod performance parameters for transient (anticipated operational occurrences (AOOs)) analysis, loss of coolant accident (LOCA) analysis, control rod drop accident (CRDA) analysis, and thermal hydraulic stability analysis are provided.

## 3.2 Design Criteria

The design criteria for the fuel assembly and fuel rod performance analyses are provided in Reference 37. An overview of those criteria is provided in this section.

The principal objective of the SVEA mechanical design criteria is to assure compliance with the specified acceptable fuel design limits of General Design Criteria (GDC) 10, the rod insertability requirements of GDC 27, and the core coolability requirements of GDC 35, which are provided in 10CFR50, Appendix A (Reference 42). To accomplish these objectives, the fuel is designed to meet the acceptance requirements identified in Standard Review Plan (SRP), Section 4.2 (Reference 43), relative to:

1. No calculated fuel system damage for normal operation and anticipated operational occurrences which includes no predicted fuel rod failure (defined as exceeding the fuel cladding plastic strain design limits and fuel centerline melting temperature), fuel system dimensions remain within operational tolerances, and fuel system functional capabilities not reduced below those assumed in the safety analysis; and

2. Retention of fuel coolability and control rod insertion when required during postulated accidents which includes retention of rod-bundle geometry with adequate coolant channels to permit removal of residual heat considering the potential for cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme co-planar fuel rod ballooning.

The mechanical integrity design criteria are provided in three categories in Reference 37:

1. General design criteria to assure that all required fuel system damage, fuel rod failure, and fuel coolability issues are addressed for new assembly designs and design changes,

2. Specific design criteria for the assembly components, other than fuel rods, to assure that the general design criteria are satisfied, and

3. Specific design criteria for the fuel rods to assure that the general design criteria are satisfied.

The mechanical design criteria for normal operation and anticipated operational occurrences are provided in Section 3 of Reference 37.

The nuclear fuel assembly is classified as a Seismic Category I component. To ensure compliance with the requirements of U.S. NRC Standard Review Plan, Section 4.2 (Reference 43), the fuel assembly is designed to withstand a Safe Shutdown Earthquake (SSE) in conjunction with structural and hydraulic loads from the worst LOCA. The postulated design base SSE and LOCA events are described in Section 3 of Reference 38. A set of specific acceptance criteria are established to demonstrate that the design bases given in Section 3 of Reference 38 are satisfied. These acceptance criteria are provided in Section 4 of Reference 38.

### 3.3 Design Methodology

The ABB methodology for evaluation of fuel assembly mechanical integrity for normal operation and AOOs relative to the design criteria is provided in Section 4 of Reference 37. In addition, an evaluation of the fuel assembly relative to the design criteria provided in Section 3 of Reference 37 is performed for each plant application. If appropriate conditions such as plant operating conditions, burnup requirements, and assembly design do not change, a single evaluation can be applied to all cycles for a given plant for many of the criteria. Therefore, whenever possible, bounding conditions are assumed for a specific plant to accommodate conditions from cycle-to-cycle.

In addition to the methodology description, the ABB methodology described in Reference 37 is applied to the SVEA-96 design as an illustration in Reference 37. This illustration is provided to help the reader understand the methodology and to provide an indication of the margins relative to the design criteria inherent in the SVEA-96 design.

The general methodology used to evaluate a BWR fuel assembly mechanical integrity and its effect on the reactor internals, including control rods, when subjected to a postulated seismic and LOCA event, is described in Section 5 of Reference 38. Where appropriate, a general discussion of the methodology is included. Specific applications which illustrate this general methodology are presented in Section 6 of Reference 38.

## 3.4 Methodology for Mechanical Design Input to Reload Design and Safety Analysis

### 3.4.1 Mechanical Design Input to Nuclear Design Analyses

This section describes the methodology for providing mechanical design input to the nuclear design analysis. The nuclear design analyses require input regarding detailed dimensions of the fuel assemblies used in the core from ABB and other vendors.

All mechanical design data for ABB reload assemblies required for the nuclear design is formally provided internally. These data include:

1. A complete dimensional description of the assembly,
2. Assembly materials properties information,
3. Assembly materials composition data, and
4. Assembly and component masses.

In addition, criteria and limits required for the satisfactory mechanical performance of the ABB-designed assembly are provided to assure that the nuclear design of the Reference Core is such that these criteria and limits can be satisfied in operation. This information includes:

[ Proprietary Information Deleted ]

### 3.4.2 Mechanical Design Input to Thermal Hydraulic Design Analyses

A complete dimensional description of the assembly is required for the thermal-hydraulic description and design evaluation of the assembly. This information includes:

1. Assembly and component dimensions,
2. Assembly and component flow areas, and
3. Any additional mechanical data required for the SLMCPR evaluation. For example, uncertainties in assembly flow areas to support the SLMCPR evaluation.

All mechanical design data for ABB reload assemblies required for the thermal-hydraulic design and design evaluation are formally provided internally. ABB obtains from the utility the required data for non-ABB fuel which resides in the reactor with ABB fuel and which supports the thermal-hydraulic design of the Reference Core. In general, the same dimensional data required for the ABB assembly design are required for the non-ABB fuel assemblies.

### 3.4.3 Mechanical Design Input to the Transient Analyses

All mechanical design data for ABB reload assemblies required for the transient analyses are formally provided internally. ABB obtains from the utility the required data for non-ABB fuel which resides in the reactor with ABB fuel and which supports the transient analyses from the utility. In general, the same dimensional data required for the ABB assembly design are required for the non-ABB fuel assemblies.

<u>Assembly Input Data</u>

The same assembly dimensional data required for the nuclear and thermal-hydraulic analyses are made available for the transient analyses. The same Linear Heat Generation Rate (LHGR) limits which assure that mechanical design criteria will be satisfied under transient conditions which are provided for the nuclear design are utilized in the transient analyses.

[ Proprietary Information Deleted ]

### 3.4.4 Intentionally Deleted

### 3.4.5 Mechanical Design Input to LOCA Analyses

The LOCA analysis requires virtually the same mechanical assembly, core, and plant dimensional data as for the transient analyses.

As for the transient analyses, fuel rod performance data for the LOCA analyses are calculated using a fuel rod performance code accepted by the NRC (see Appendix A). Inputs to the fuel rod performance code include fuel rod dimensional data, enrichments, pellet density, initial rod pressurization, and power history. [ Proprietary Information Deleted ]

### 3.4.6 Mechanical Design Input to CRDA Analyses

The methodology for analyzing the Control Rod Drop Accident is described in Reference 33. The description in Reference 33 includes the treatment of mechanical input data, such as gap HTCs, and, therefore, is not repeated in this document.

### 3.4.7 Mechanical Design Input to Stability Analyses

Virtually the same mechanical assembly, core, and plant dimensional data are required for the input to the stability analysis codes as for the transient analyses.

Fuel rod performance data for the stability analyses are calculated using a fuel rod performance code accepted by the NRC (see Appendix A). Inputs to the fuel rod performance code include fuel rod dimensional data, enrichments, pellet density, initial rod pressurization, and power histories. [ Proprietary Information Deleted ]

# 4 NUCLEAR DESIGN

## 4.1 Summary and Conclusions

This section provides the ABB BWR nuclear design bases and describes the methodology used to demonstrate compliance with those bases under steady-state conditions and to generate nuclear data for other disciplines. Sample applications of the methodologies for the ABB SVEA-96 reload fuel are provided in Appendix D.3. Conformance with the design bases is demonstrated for the SVEA-96 assembly in the sample applications in Appendix D.3.

Specifically, this section contains the following:

- The ABB nuclear design bases;

- The ABB methodology used to evaluate compliance with the nuclear design bases for steady-state conditions, including the development of the Reference Core and the treatment of the final loading pattern;

- The methodology for enveloping the nuclear input to the mechanical, thermal and hydraulic, AOO, accident, and special event analyses.

A description of the nuclear characteristics of ABB SVEA-96 fuel and a sample application of the nuclear design methodology is provided in Section D.3. The examples are for the SVEA-96 fuel assemblies in a 764-assembly BWR/5 core.

The objective of the nuclear design process for a given cycle is to establish the following information consistent with the constraint that thermal (e.g. MCPR and LHGR) and reactivity (e.g. shutdown margin) limits can be satisfied:

(1) Number and enrichment of the feed fuel assemblies that meet the required energy output and cycle length,

(2) A realistic nuclear core model that can be utilized for core follow and support of subsequent cycles,

(3) Reference Core loading pattern, target control rod sequences, and expected core power, burnup, and void history distributions to support the cycle Reload Safety Analysis.

(4) Nuclear-related parameters required for the Reload Safety Analysis. Such key safety parameters include reactivity coefficients, cross sections, control rod reactivity worths, and local peaking factors that are used as input assumptions to the

analyses of Anticipated Operational Occurrences (AOOs), special event analyses, and accident analyses.

Discussions of the methods used to accomplish these ends are provided in Sections 4.3 and 4.4.

The information in this section supports the following conclusions regarding the ABB nuclear design bases and methodology:

(1) The design bases identified are sufficient to assure that the applicable General Design Criteria (GDC) in 10CFR50, Appendix A (Reference 42) as well as the requirements and guidelines for assembly nuclear design identified in Section 4.3 of NUREG-0800 (Reference 43) will be satisfied.

(2) The methodology described in this section for evaluating the nuclear performance of BWR fuel is adequate for evaluation of reload fuel evaluation relative to the design bases. This methodology is acceptable for design and licensing application. Specifically, the methodology described in this section for determining nuclear parameters such as power, burnup and void-history distributions, reactivity coefficients, shutdown margin, and cross section data for ABB as well as non-ABB fuel is acceptable for design and licensing applications.

## 4.2 Design Bases

This section describes the ABB reload fuel nuclear design bases and relates these design bases to the General Design Criteria (GDC) in 10CFR50, Appendix A (Reference 43).

### 4.2.1 Fuel Burnup

Basis

The nuclear design basis is to install sufficient reactivity in the reload fuel to meet design lifetime requirements while satisfying the fuel rod and fuel assembly design bases and assuming the shutdown margin requirements are satisfied.

Discussion

The fuel rod and assembly design bases and their dependence on burnup are discussed in Section 3 and Reference 37.

This basis, in conjunction with the design basis in Section 4.2.3, Control of Power Distribution, assures that GDC-10 is satisfied for the cycle under consideration.

The ABB methodology for evaluating conformance to this design basis is discussed in Section 4.3.

### 4.2.2 Reactivity Coefficients

Basis

The Doppler fuel temperature and moderator void coefficients of reactivity shall be negative while in the power operating condition, thereby providing negative reactivity feedback characteristics for normal operation and AOOs.

The reactivity feedback shall be sufficiently negative to provide adequate control and maneuvering of the core power in the power range.

Discussion

This design basis assures that GDC-11 is satisfied for normal operation and AOOs for the cycle under consideration. Design criteria assuring sufficient negative reactivity under accident conditions (e.g., the Control Rod Drop Accident) are addressed in Section 8.

Compensation for a rapid increase in reactivity is provided by two basic phenomena. These phenomena are the resonance absorption associated with changing fuel temperature, or Doppler effect, and the impact on neutron spectrum resulting from changing moderator density. The use of low enrichment uranium ensures that the Doppler coefficient of reactivity is negative. This coefficient provides the most rapid negative reactivity compensation. The core is also designed to have an overall negative moderator void coefficient of reactivity so that the coolant void content provides another rapid negative reactivity feedback mechanism. Power operation is permitted only in a range of overall negative moderator void coefficient. The negative moderator void coefficient is assured through the geometry of the fuel itself and through the selection of the fuel assembly enrichment and burnable absorber distribution.

The ABB methodology for evaluating conformance to this design basis is discussed in Section 4.3.

### 4.2.3 Control of Power Distribution

Basis

The nuclear design bases on core power distribution are:

(1) During normal operation, the nuclear design will be such that the Linear Heat Generation Rate (LHGR) limits established to meet the mechanical fuel rod design bases are not exceeded.

(2) For anticipated operational occurrences, the fuel peak power will not cause the Specified Acceptable Fuel Design Limits (SAFDLs) to be exceeded.

(3) The nuclear design will be such that the fuel will not operate with a power distribution that violates the Cladding Integrity Design Basis for both normal operation and for AOOs.

(4) The nuclear design will be such that the fuel will be operated at or below specified Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits under normal operating conditions which ensure compliance with the Loss of Coolant Accident (LOCA) criteria in 10 CFR 50.46.

Discussion

This design basis assures that GDC-10 is satisfied for normal operation and AOOs for the cycle under consideration.

The SAFDLs are identified in Section 6.

The ABB methodology for evaluating conformance to this design basis is discussed in Section 4.3.

### 4.2.4 Shutdown Margin

Basis

The core shall be subcritical in its most reactive condition with all control rods fully inserted except for the single control rod with the highest reactivity worth, which is assumed to be in its full-out position.

The Standby Liquid Control System shall be capable of shutting the reactor down to the cold condition from the most reactive reactor operating state at any time in cycle life.

Discussion

This design basis assures that GDC-26 and GDC-27 are satisfied for the cycle under consideration.

Two independent reactivity control systems are provided in U.S. plants. These control systems are the control rods and soluble boron in the coolant from the Standby Liquid Control System. The control rod system by itself is designed to compensate for the reactivity effects of the fuel and moderator temperature and density changes accompanying power level changes over the complete range from cold, clean, zero-power to full power, equilibrium xenon conditions without the benefit of the Standby Liquid Control System (SLCS). The fuel bundle and loading pattern design must be such that the control rod system itself provides the minimum shutdown margin (SDM) under all operating conditions and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits assuming that the highest worth control rod is stuck out upon trip. This capability must be available at all times in core life at all operating states. The ABB methodology for evaluating conformance to this design basis is discussed in Section 4.3.

The Standby Liquid Control System (SLCS) provides an alternate means of attaining and maintaining the reactor in the cold shutdown state by the injection of soluble boron. At any time in core life, the SLCS must be capable of bringing the reactor to a shutdown condition from any operating state, assuming no movement of the control rods. Thus, backup and emergency shutdown provisions are provided by this chemical poison system. The ABB methodology for evaluating conformance to this design criterion is discussed in Section 9.

### 4.2.5 Stability

Basis

The bundle and loading pattern design shall be such that the potential for growing or limit cycle power oscillations are sufficiently minimized that power oscillations that can result in conditions exceeding the SAFDLs do not occur or are readily detected and suppressed.

Discussion

This design basis assures that GDC-12 is satisfied for the cycle under consideration.

In principal, power oscillations can be caused by spatial xenon and void feedback effects. However, the negative void coefficient associated with the boiling condition in a BWR provides a much larger and more rapid feedback effect than that caused by variations in xenon concentration. In addition, the void feedback rapidly damps any xenon oscillations. Therefore, a specific evaluation of xenon oscillations is not required on a reload-specific basis. The ABB treatment of hydrodynamic stability is discussed in Section 9. It should be noted that the ABB time domain methods referred to in Section 9 treat variations in xenon concentration as well as the effects of void feedback.

## 4.3 Reload Nuclear Design Methodology

### 4.3.1 Reference Core

The ABB BWR safety analyses methodology uses the Reference Core approach. This approach requires the development of a Reference Core design (e.g. loading pattern, batch sizes, and control rod sequences) which is designed with the intent that it will model as closely as possible the as-loaded core for the upcoming cycle. The cycle-specific safety analyses are performed for the Reference Core. The Reference Core thus forms the licensing basis for the upcoming cycle.

The Reference Core loading pattern is designed with five primary goals:

(1) To meet the customer cycle energy requirements;

(2) To meet all licensing requirements;

(3) To optimize operating margin and flexibility;

(4) To make the most efficient use of the energy available in the expected inventory of partially burned fuel and the feed fuel; and

(5) To provide sufficient flexibility to accommodate, with only minor loading pattern changes, the degree of variation in bundle inventory and current cycle length changes usually associated with scheduler or energy requirement changes.

The Reference Core is developed on a schedule which supports the cycle-specific Reload Safety Analysis and required documentation for utility and regulatory authorities. The Reference Core is based on the best estimates of:

(1) The cycle energy requirements in the next cycle;

(2) The end of cycle exposure conditions for the previous cycle; and

(3) Bundle inventory at refueling.

The Reference Core is designed such that, if all the estimates that went into its development are accurate, it would be the design for the upcoming cycle. Hence, in addition to safety analyses considerations, considerations regarding operability and economy are reflected in the design of the Reference Core loading pattern. The design of the Reference Core pattern also accommodates plausible deviations from the estimated conditions at the end of the ongoing cycle. This includes consideration of a target exposure window with regard to design parameters (such as shutdown margin) which is exposure dependent as explained below.

Since the Reference Core is intended to be the core design used in the upcoming cycle, the Reference Core is subjected to all cycle-specific analyses and evaluations required to assure that the design will comply with all applicable design bases. These analyses set the operating limits of the upcoming cycle. These analyses and evaluations include the following items:

(1) Shutdown margin requirement,

(2) Determination of Safety Limit MCPR (SLMCPR),

(3) Cycle specific AOOs,

(4) Cycle specific accidents, and

(5) Special events such as the Standby Liquid Control System (SLCS) capability requirement, stability, and reactor overpressure protection.

Item (1) is addressed in Section 4.3.2. Item (2) is addressed in Sections 4.4.2 and 5. Items (3), (4) and (5) are addressed in Sections 7, 8, and 9, respectively.

Deviations from the Reference Core due to changes in cycle length or fuel inventory require reevaluation to assure that the actual as-loaded core will meet safety limits. Guidelines for the evaluation of deviations from the Reference Core are discussed in Section 4.3.1.3.

#### 4.3.1.1 Bundle Design Cross Section Calculations

The determination of the $UO_2$ enrichment distribution and burnable absorber design is an iterative process with the loading pattern determination and control rod sequence determination described in Section 4.3.1.2. Ultimately, the bundle design must support the definition of a satisfactory loading pattern meeting all applicable limits and design bases in a manner which optimizes fuel efficiency.

[ Proprietary Information Deleted ]

Preliminary bundle designs established in this manner are utilized in the three-dimensional calculations described in Section 4.3.1.2 to establish a satisfactory loading pattern. Based on these calculations the bundle designs are optimized to meet the design goals and applicable design bases.

A nuclear design code system accepted by the NRC is utilized for the bundle and loading pattern design and determination of target control rod sequences. ABB currently utilizes the system of codes documented in Appendix A. The two-dimensional lattice physics code is used to calculate the nuclear data (e.g. cross sections, local peaking factors, MCPR subchannel factors, detector constants, etc.) required for the three-dimensional nodal core simulator input as well as the transient and accident computer codes.

### 4.3.1.2 Loading Pattern and Control Rod Sequences

Loading patterns and control rod sequences are established by an iterative process which is illustrated in Figure 4-1. [ Proprietary Information Deleted ]

### 4.3.1.3 Deviations from the Reference Core

The Reference Core design, upon which the Reload Safety Analysis is based, is established based on a set of assumed core conditions at the end of the ongoing cycle. The actual end-of-cycle conditions may differ from the estimates, however, and the as-loaded core loading arrangement may be different from the Reference Core loading pattern. Deviations from the Reference Core can include:

(1) Different assembly inventory;

(2) Different end-of-cycle exposures due to a shorter or longer cycle length than planned;

(3) Different exposure distributions than used in the Reference Core Reload Safety Analysis, particularly the axial exposure distributions; or

(4) Deviations in the as-loaded core which do not preserve the symmetry of the Reference Core.

A major deviation from the Reference Core is explicitly treated by repeating affected parts of the Reload Safety Analysis calculations to confirm that the conclusions based on the Reference Core are valid or to modify them appropriately. The following guidelines are utilized to increase the probability that any deviation from the Reference Core can be shown to be acceptable without a major reanalysis. Regardless of the deviation from the reference loading pattern, a shutdown margin calculation is performed for the as-loaded core.

<u>Assembly Inventory</u>

The following guidelines address deviations in the assembly inventory from that assumed for the Reference Core:

[ Proprietary Information Deleted ]

It should be noted that any deviation in the reload core inventory is evaluated even if it falls within these guidelines. Adherence to these guidelines increase the probability that a major reanalysis will not be required.

Different Core Average Exposure

The Reference Core analysis for cycle N+1 is typically based on the assumption that the core average exposure at the end-of-cycle (EOC) N remains within an allowed deviation from the expected EOC N exposure. A nominal allowed deviation, or burnup window, is selected based on sensitivity studies that demonstrate that the safety criteria for the Reference Core design are met for deviations of the core EOC N exposure within this nominal burnup window. Should the actual exposure fall outside of the exposure window, the cycle specific safety analysis is evaluated and augmented as required to cover the actual cycle exposure. The magnitude of the burnup window can be core-and cycle-specific.

Different Axial Exposure Distribution

A comparison is made between the core average axial burnup distribution actually realized near the end-of-cycle compared with that assumed for the Reference Core safety analysis. Any deviation which adversely affects the operating limits established by Reload Safety Analysis significantly is evaluated, and affected parts of the Reload Safety Analysis calculations are repeated or modified to confirm that all applicable limits are still satisfied.

Deviations from Assumed Core Symmetry

The Reference Core is designed with a symmetry which supports the specific cycle, utility, plant process computer, and core requirements. Core asymmetries that involve the asymmetric loading of fuel assemblies can be accommodated. Fuel assembly loading-related asymmetries, or that due to an asymmetric control rod pattern, are evaluated for their impact on the operating margins relative to the operating limits and as to whether they invalidate any of the safety analysis conclusions. Any deviation which adversely affects the operating limits established by Reload Safety Analysis significantly is evaluated, and affected parts of the Reload Safety Analysis calculations are repeated or modified to confirm that all applicable limits are still satisfied.

### 4.3.1.4 Reload Cycle Design Model

When the actual characteristics of the reload cycle and the previous cycle are sufficiently well established, the Reference Core three-dimensional nodal simulator model is modified accordingly to obtain an accurate representation of the reload core referred to as the Reload Cycle Design Model. This model is utilized for support of any required revisions to the Reload Safety Analysis based on the Reference Core, for a revised shutdown margin calculation, and as a core-follow model to be used as the reload cycle depletes. The Reload Cycle Design Model is also utilized to provide projections for the design and Reload Safety Analyses of the next cycle.

## 4.3.2 Performance Relative to Nuclear Design Bases and Calculation of Selected Parameters

### 4.3.2.1 Fuel Burnup

The core design lifetime or design discharge burnup is achieved by establishing a bundle design and developing a loading pattern that simultaneously satisfies the energy requirements and satisfies all safety related criteria in each cycle of operation.

The bundle and loading pattern design must be sufficient to maintain core criticality at full power operating conditions throughout the cycle with burnable poison concentration, equilibrium xenon, samarium, and other fission products present.

The Reference Core calculations are utilized to confirm that cycle energy requirements and fuel burnup limitations are satisfied. Reference values of $k_{effective}$ established from plant data are utilized to conservatively establish the end-of-full power reactivity level which will be predicted by ABB methods to assure that cycle energy requirements are satisfied.

The Reference Core calculations are used to confirm that burnup limitations will not be exceeded. Burnup limitations are established by fuel rod and fuel assembly considerations discussed in Reference 37.

### 4.3.2.2 Reactivity Coefficients

Reactivity void and Doppler coefficients are reviewed qualitatively during the Reference Core design to confirm that they are negative and that they are in an appropriate range to provide adequate reactivity feedback to support conformance with thermal, reactivity, and thermal-mechanical limits addressed in the Reload Safety Analysis.

[ Proprietary Information Deleted ]

In addition to the void and Doppler coefficients, values of the following parameters are also required for the evaluation of AOOs, accidents, and special events:

(1) Delayed Neutron Fractions

(2) Inverse Neutron Velocities and Prompt Neutron Lifetimes

(3) Energy Deposition Fractions

Therefore, the methodology for evaluating these parameters is also provided in this section. These calculations are performed with approved nuclear design code systems. ABB currently utilizes the nuclear design code system documented in Appendix A.

Moderator Void Reactivity Coefficient

The void coefficient of reactivity is defined as the change in reactivity per unit change in the core average void fraction. The value of this coefficient is sensitive to changes in the moderator density, the moderator temperature (keeping the density constant), the fuel burnup, and the presence of control rods and burnable poisons.

[ Proprietary Information Deleted ]

Doppler Coefficient of Reactivity

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per unit temperature change in fuel temperature. The core-average Doppler coefficient can be calculated by a combination of two- and three-dimensional methods by:

[ Proprietary Information Deleted ]

Delayed Neutron Fractions

Effective delayed neutron fractions vary with isotopic composition and, therefore, with such parameters as burnup and void history. The core-average effective delayed neutron fraction can be calculated by a combination of two- and three-dimensional methods by:

[ Proprietary Information Deleted ]

Inverse Velocities and Prompt Neutron Lifetimes

Fast and thermal neutron inverse velocities are obtained in the same manner as the delayed neutron fractions described above. Inverse velocities for each fuel type are calculated with the two-dimensional lattice code (Appendix A), and a core-average values are calculated as weighted averages of the fuel-type specific values using the three-dimensional core simulator results to determine the weighting factors.

Core-average prompt neutron lifetimes are calculated from the core-average inverse velocities using standard expressions.

Energy Deposition Fractions

The fraction of power released or generated outside the fuel material is required for steady-state and dynamic calculations. For most purposes, generic average values are acceptable. For example, [ Proprietary Information Deleted ] the fission energy is typically assumed to be deposited in the fuel with about half of the remaining energy deposited in the coolant and the other half deposited in the interassembly bypass, the internal bypasses (e.g. water cross), and Zircaloy cladding and channel envelope materials of the fuel assembly for steady-state calculations. A total percentage of fission energy deposited outside of the fuel is provided for rapid transient events, and the fraction of energy deposited in the coolant is calculated as part of the transient analysis.

[ Proprietary Information Deleted ]

#### 4.3.2.3 Control of Power Distribution

*Methodology*

The four design bases listed in Section 4.2.3 are satisfied during core operation by requiring conformance to those limits and monitoring that conformance with the Core Supervision System. During the design phase, the Reference Core is designed in a manner which provides a high level of confidence that power distributions during core operation can be conveniently maintained within the limits required by Design Basis 4.2.3. Design methodology to achieve this goal is discussed in this section in the order in which the corresponding design bases are presented in Section 4.2.3.

(1) The reload fuel feed bundle and Reference Core reload pattern and control rod sequences are specifically designed such that during normal operations the Linear Heat Generation Rate (LHGR) limits established to meet the mechanical fuel rod design bases are not exceeded. As discussed in Section 4.3.0 of Reference 37, a Design Power History (DPH) is established for which all mechanical design bases are satisfied [ Proprietary Information Deleted ] Confirmation that the DPH is not exceeded demonstrates that all mechanical fuel rod design bases are satisfied. [ Proprietary Information Deleted ]

(3) The reload fuel feed bundle and Reference Core reload pattern and control rod sequences are specifically designed such that, to a high level of confidence, the fuel will not experience power distributions which could credibly lead to a violation of the Cladding Integrity Design Basis for both normal operation and for AOOs. [ Proprietary Information Deleted ]

(4) The reload fuel feed bundle and Reference Core reload pattern and control rod sequences are specifically designed such that the fuel can be conveniently operated at or below specified Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits under normal operating conditions to a high level of confidence. During the design of the Reference Core, the peak Average Planar Linear Heat Generation Rates (APLHGRs) are compared to the MAPLHGRs at each statepoint to confirm that the design provides sufficient margin to the MAPLHGRs to assure that the MAPLHGR will not be approached during normal operation in the plant application.

*Discussion*

The sample applications discussed in Appendix D.3 provide examples of the comparisons to the TMOL and the OLMCPR. Sample discussions of the margin to the SAFDLs under transient conditions are provided in Appendix D.6.

#### 4.3.2.4 Shutdown Margin

The ABB methodology for evaluating the shutdown capability of the Standby Liquid Control System is discussed in Section 9. This section provides the methodology for demonstrating that the core can be made subcritical with the most reactive control rod assumed to be fully withdrawn.

*Methodology*

The reload fuel feed bundle and Reference Core reload pattern are specifically designed such that the core will be subcritical in its most reactive condition with all control rods fully inserted with the exception of any single control rod in the core. This highest worth rod is assumed to be in its full-out position.

[ Proprietary Information Deleted ]

*Discussion*

The sample application discussed in Appendix D.3 provides sample results of applying this methodology of the SDM evaluation.

## 4.4 Nuclear Design Input to Other Disciplines

### 4.4.1 Nuclear Design Input to Mechanical Design

*Methodology*

Fuel rod power histories are provided for the thermal-mechanical design evaluation of the fuel rods for each plant application as described in Section 4.3.0 of Reference 37. These calculations are performed to confirm that the Design Power History is in fact bounding for a specific application and to identify the level of conservatism associated with that power history. [ Proprietary Information Deleted ]

*Discussion*

An example of the selection of limiting fuel rods and the resulting power histories is provided in Section 4.3.0 of Reference 37.

### 4.4.2 Nuclear Design Input to Thermal-Hydraulic Design

Conservative radial power distributions are provided for the cycle-specific SLMCPR calculation discussed in Section 5. These radial bundle power distributions are based on the Reference Core three-dimensional core simulator calculations discussed in Section 4.3.1. The term "conservative" refers in this case to the radial power distribution which places a larger number of fuel rods with a higher probability of experiencing boiling transition than radial power distributions which could lead to limiting MCPR situations during plant operations. [ Proprietary Information Deleted ]

### 4.4.3 Nuclear Design Input to Transient Analyses

The AOOs discussed in Section 7 can be categorized as "fast" or "slow". The "slow" transient events are analyzed using the three-dimensional core simulator. These events include transients which can be adequately modeled with steady-state methods because of the relatively long time frame of the transient and quasi steady-state conditions existing throughout the transient. Such transients include the Loss of Feedwater Heating and the Rod Withdrawal Error. These AOOs are evaluated directly with the Reference Core three-dimensional nodal simulator model discussed in Section 4.3.1.

The "fast" transient events are evaluated with a transient analysis code system accepted by the NRC. ABB currently utilizes the code documented in Appendix A for this purpose. This one-dimensional axial space-time kinetics transient analysis code computes the overall reactor response during a transient event. The change in critical power ratio (ΔCPR) for the limiting fuel assembly in the core is evaluated with a supplemental "slave channel" model. [ Proprietary Information Deleted ]

### 4.4.4 Nuclear Design Input to the Accident Analyses

#### 4.4.4.1 LOCA Analysis

As discussed in Section 8.2, ABB utilizes the series of codes documented in References 21, 35, and 40 for the LOCA analysis. Since this code system utilizes a point kinetics model, point kinetics parameters are required. Therefore, the following parameters are provided at required statepoints:

a. Moderator Void Reactivity Coefficient,

b. Fuel Temperature (Doppler) Coefficient,

c. Delayed Neutron Fractions and Decay Constants,

d. Prompt Neutron Generation Time,

e. Energy Deposition Fractions

These parameters are calculated as described in Section 4.3.2.2. [ Proprietary Information Deleted ]

In addition to the point kinetics parameters, the LOCA analysis also requires the following power distribution information:

[ Proprietary Information Deleted ]

#### 4.4.4.2 Nuclear Design Input to CRDA Analyses

The CRDA methodology is described in Reference 33. As discussed in Reference 33, a CRDA analysis is fundamentally a two-step approach. The first step involves determination of possible candidates for the control rod which would cause the most severe consequences resulting from a CRDA. [ Proprietary Information Deleted ]

The second step is simulation of the dynamic response to the identified worst dropped control rod(s) and the subsequent consequences to the fuel. This evaluation is performed with a three dimensional systems transient code approved for this purpose (Reference 33). [ Proprietary Information Deleted ]

### 4.4.4.3 Nuclear Design Input to Fuel Handling Accident Analyses

[ Proprietary Information Deleted ]

### 4.4.4.4 Mislocated and Rotated Fuel Assembly Analyses

Mislocated and Rotated Fuel Assembly Analyses are performed with the three-dimensional nodal simulator and two-dimensional lattice physics codes as discussed in Section 8.

## 4.4.5 Nuclear Design Input to Special Events Analyses

### 4.4.5.1 Stability Analysis

As discussed in Section 9, the ABB stability analysis methodologies utilizes the frequency domain code and the time domain code described in References 24, 44, and 45.

Frequency Domain Methodology Input

A point kinetics model is utilized in the frequency domain code. Therefore, the following parameters are provided at statepoints required by the analysis:

(a) Moderator Void Reactivity Coefficient,

(b) Fuel Temperature Coefficient,

(c) Delayed Neutron Fractions and Decay Constants, and

(d) Prompt Neutron Generation Time.

These parameters are calculated as described in Section 4.3.2.2. [ Proprietary Information Deleted ]

Advanced Stability Methodology Input

This evaluation is performed with the three dimensional systems transient code described in Reference 44. Appropriate files from the three-dimensional core simulator provide the nodal burnups and void histories for the specific state point considered in the three dimensional systems transient code calculation as shown in Figure 4-2. [ Proprietary Information Deleted ]

### 4.4.5.2 Overpressurization Protection

This analysis is performed with the same dynamic analysis models utilized for the fast transient AOOs discussed in Section 4.4.3.

Therefore, the input to these analyses is the same as the input to the fast transient AOOs described in Section 4.4.3.

#### 4.4.5.3 Standby Liquid Control System

The Standby Liquid Control System (SLCS) evaluation is performed with the three-dimensional nodal simulator and two-dimensional lattice physics codes as discussed in Section 9.

## FIGURE 4-1 THROUGH FIGURE 4-2

Proprietary Information Deleted

# 5 THERMAL-HYDRAULIC DESIGN

## 5.1 Summary and Conclusions

### 5.1.1 Summary

This section provides the ABB BWR thermal and hydraulic design bases and describes the methodology used to demonstrate compliance with those bases. Sample applications of the thermal and hydraulic methodologies are provided in Section D.4.

Specifically, this section contains the following:

- The ABB thermal and hydraulic design bases,
- The ABB methodology used to evaluate compliance with the thermal and hydraulic design bases for steady-state conditions. The methodology for treating Anticipated Operational Occurrences (AOOs) and postulated accident conditions are addressed in Sections 7 and 8, respectively. The methodology for the treatment of undamped oscillations and other thermal-hydraulic instabilities is discussed in Section 9.
- Thermal and hydraulic input to the mechanical, nuclear, AOO, accident, and special event analyses.

### 5.1.2 Conclusions

The information contained in this section supports the following conclusions regarding the ABB thermal-hydraulic methodology and the thermal-hydraulic characteristics of the SVEA-96 assembly:

(1) The design bases identified are sufficient to assure that the requirements and guidelines for assembly thermal-hydraulic performance identified in Section 4.4 of NUREG-0800 will be satisfied.

(2) The methodology described in this section for evaluating the thermal and hydraulic performance of BWR fuel fulfills the design bases and is acceptable for design and licensing application. Specifically, the methodology described in this section for evaluating Critical Power performance and hydraulic compatibility for ABB as well as non-ABB fuel is acceptable for design and licensing applications.

(3) The ABB methodology for establishing hydraulic compatibility of different fuel types is illustrated in Appendix D.4 for two sample 764 assembly BWR/5 cores containing SVEA-96 and 8x8 and 9x9 fuel. These examples illustrate that the flexibility

of the SVEA-96 fuel design is sufficient to assure hydraulic compatibility with other fuel types in a mixed core.

(4) The ABB methodology for determining the safety limit for SVEA-96 fuel, as well as mixed cores containing other fuel types, is illustrated by application to a 764 assembly BWR/5 reactor.

## 5.2 Design Bases

The principal objective of the thermal and hydraulic design is to assure that the relevant requirements of General Design Criteria (GDC) 10 in 10CFR50, Appendix A (Reference 42) are satisfied. To accomplish this objective, the fuel is designed to meet the acceptance requirements outlined in Standard Review Plan (SRP), Section 4.4 (Reference 43), to assure that acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences (AOOs).

### 5.2.1 Cladding Integrity

*Basis*

The minimum value of CPR is established such that at least 99.9% of the fuel rods in the core would not be expected to experience boiling transition during normal operation or anticipated operational occurrences.

*Discussion*

The multiple-barrier concept has been adopted by the nuclear industry to prevent the escape of radioactive fission products to the environment. The first of these barriers is the fuel rod cladding. A potential failure mechanism of the fuel rod cladding is the overheating of the cladding due to inadequate heat transfer. Therefore, adequate margin must be maintained during the reactor steady-state and transient operations to ensure cladding integrity.

Compliance with this design basis also assures that Design Criterion 3.3.8 of Reference 37 for cladding temperature is also satisfied.

The design limit which protects the fuel cladding from overheating is the Critical Power Ratio (CPR). CPR is the ratio of the critical power to the actual power in an assembly. The critical power is defined as the power at which the transition from nucleate boiling to film boiling would occur in the most limiting rod in that assembly for a given pressure, flow, inlet enthalpy and axial power shape. This transition to film boiling is conservatively assumed to be the point of cladding failure. Therefore, the critical power is the maximum power at which an assembly could be operated. However, because of uncertainties in the instrumentation readings and process measurements, variations in as-built core design parameters and inaccuracies in calculation methods used in the assessment of thermal margin, the CPR must be maintained above 1.0 in practice.

Section 4.4 of Reference 43 requires that these uncertainties be treated such that there is at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience boiling transition during normal operation or anticipated operational occurrences. This requirement is achieved for BWR fuel by establishing the Safety Limit MCPR (SLMCPR), such that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The methodology for establishing SLMCPR values is provided in Section 5.3. As described in Section 7, plant and cycle specific analyses are performed to determine the impact of the most limiting AOOs on the MCPR. The Operating Limit Minimum MCPR (OLMCPR) is set such that the worst AOO does not violate the SLMCPR. The OLMCPR value for each cycle and fuel type is typically defined in the plant Licensee's Core Operating Limits ("COLR") Report. The treatment of MCPR for ABB and non-ABB fuel to assure that the OLMCPR is satisfied during reactor operation and during the reload design phase are discussed in Section 5.3.

### 5.2.2 Hydraulic Compatibility

*Basis*

Reload fuel shall be designed to be hydraulically compatible with the Resident fuel in the core when the Reload fuel is installed and compatible with the hydraulic characteristics of the core. Specifically,

(1) At reactor rated power and flow conditions, the total interassembly bypass flow will be maintained within the design range of the plant. This range is typically 8 to 12% of the total core flow. If the interassembly bypass flow for a specific reload is outside of the design range of the plant, the safety significance of plant operations will be specifically evaluated in accordance with 10CFR50.59.

(2) Hydraulic compatibility will be demonstrated at rated conditions and for the allowable flow domain.

*Discussion*

The "Reload" fuel assembly refers to a new fuel assembly installed in a core containing "Resident" fuel assemblies of the same or a different design. The "Reload" fuel assembly will be an ABB fuel assembly. The SVEA-96 assembly is the ABB BWR fuel assembly currently being marketed in the U.S. The Resident fuel assemblies can be ABB assemblies of a different design than the Reload fuel or fuel manufactured by a vendor other than ABB.

[ Proprietary Information Deleted ]

The methodology for assuring sufficient hydraulic compatibility is discussed in Section 5.3.

### 5.2.3 Bypass, Water Rod and Water Cross Flow

*Basis*

The fuel assembly shall be designed to maintain the interassembly bypass flow within the same range as the original plant design or within the same range provided by the current Resident fuel. The flow to the interior assembly flow bypass channels of the Reload fuel is maintained such that significant boiling will not occur.

*Discussion*

Design Basis 5.2.2 addresses interassembly bypass flow to assure acceptable flow distributions. This design basis is intended to assure that sufficient interassembly and interior assembly bypass flows are maintained at acceptable levels. By satisfying this design basis, assurance is provided that there is sufficient active coolant flow to assure that CPR margins on the fuel is maintained and that there is sufficient cooling flow to the neutron detectors. This design basis also provides assurance that the neutron kinetics parameters are maintained within the range consistent with the safety analysis.

The methodology used to assure sufficient flow to the interassembly bypass and interior assembly flow channels is provided in Section 5.3.

## 5.3 Methodology for Reload Thermal and Hydraulic Design

### 5.3.1 Thermal-Hydraulic Design Models

Accurate computer models simulating the thermal-hydraulic behavior of the core and Resident and Reload fuel assemblies are established for the following purposes:

(1) Evaluate and establish thermal-hydraulic compatibility of the Reload fuel with the Resident fuel and the core,

(2) Establish and evaluate margin to thermal limits, and

(3) Provide a consistent thermal-hydraulic data base for the mechanical and nuclear design evaluation as well as for the

evaluation of the fuel during AOOs, accidents, and special events.

[ Proprietary Information Deleted ]

Computer codes accepted for licensing applications by the NRC are used for all thermal-hydraulic analyses. The steady-state thermal hydraulics performance code is discussed in Appendix A. The thermal and hydraulic models in this code are also incorporated into the ABB BWR three dimensional nodal simulator discussed in Appendix A.

#### 5.3.1.1 Core and Assembly Models

The core is divided into groups of vertical parallel flow channels. A single flow channel is typically used to represent the outer bypass regions between the fuel assemblies. Separate flow paths are typically utilized to describe flow to the interassembly bypass upstream and downstream of the inlet orifice.

The different fuel assembly types are represented as separate flow channels. A flow channel can represent an individual fuel assembly or a group of fuel assemblies having the same thermal and hydraulic characteristics (e.g. same geometry with same radial and axial power distributions).

Figure 5-1 illustrates typical fuel assembly hydraulic components. The fuel bundle and fuel support assembly consists of three regions representing a lower region, a center region and an upper region. The lower region consists of the fuel support piece (inlet orifice), the transition piece (or bottom nozzle), and bypass flow holes. The center region consists of the bundle active flow and internal bypass flow paths. Internal bypass flow paths are typically modeled as one or two separate paths depending on the design. The upper region represents the upper tie plates and section of the channel above the upper tie plates.

The core inlet orifice, bottom nozzle, lower tie plate, spacer grid, upper tie plate, internal bypass flow inlets and exits, and the bottom nozzle bypass flow holes are hydraulically described as local form losses. Single phase friction pressure drops are computed with well established functions of fluid properties. Two-phase multipliers based on well-established phenomenological models and/or experimental data are used to calculate the two-phase friction and spacer pressure drops. Void-quality correlations are based on experimental data. Models which have been reviewed and accepted by the NRC are utilized.

Conservation of energy is required during the pressure drop calculations. A small fraction of the energy produced by the fission reaction inside the fuel rods is deposited directly into the internal and interassembly bypass regions as well as the active flow region. The remaining energy is transferred to the active flow via convective heat transfer from the fuel rods. The fractions of energy deposited directly into the internal and interassembly bypass and active flow regions are included in the model. The heat transfer from the active flow area through the channel wall to the internal and external bypass regions are also accounted for.

The enthalpy rise and quality in the active flow region are calculated from an energy balance relation. Void formation in the flow channel is based on an experimental correlation accepted by the NRC.

[ Proprietary Information Deleted ]

### 5.3.1.2 Plant and Resident Fuel Hydraulic Data

Pressure drop and flow split information for the core and resident fuel is obtained from the plant licensee for each application of ABB reload fuel. [ Proprietary Information Deleted ]

### 5.3.1.3 Hydraulic Data for Reload Fuel

Extensive test loop data are used to verify the validity of the analytical modeling of the ABB fuel. Specifically, test data are used to verify the modeling of SVEA-type design watercross and water wing modeling, design of bypass holes, tie plates, spacers, and flow distribution to the SVEA-type design subbundles as well as friction pressure drop multipliers. ABB tests are used to establish loss coefficients for these components and orifices as well as to establish the relationships between holes sizes and loss coefficients required to translate the hydraulic design parameters into dimensions for engineering drawings.

An illustration of the scope of the ABB test program is provided by the hydraulic testing of various ABB bundle designs summarized in Tables 5-1 and 5-2. Descriptions of the SVEA-100, SVEA-96, and SVEA-96+ designs are provided in Reference 37 and summarized in Section 3. The designations "SVEA-64", "SVEA-64A", and "SVEA-64B" refer to variations on the basic SVEA-64 design introduced in this country by Westinghouse as QUAD+. The variations primarily involved spacer design and channel dimensions. Much of the SVEA-64 hydraulic test program is relevant to, and supports, the SVEA-96 and SVEA-100 designs.

### 5.3.2 Thermal Design

#### 5.3.2.1 Safety Limit Minimum Critical Power Ratio

This section describes the methodology used to determine the safety limit MCPR (SLMCPR) and the uncertainties considered in the process.

Since the SLMCPR methodology is completely general and not specific to the SVEA-96 design, the methodology is acceptable for design and licensing purposes for all BWR cores containing ABB fuel, as well as for mixed cores containing both ABB and non-ABB fuel assemblies, provided adequate input data are available.

For ABB fuel assemblies in BWRs, thermal margin is described by the Critical Power Ratio (CPR) which is calculated using a CPR correlation obtained by adjusting a phenomenologically-based expression to critical power data. The SLMCPR is established to protect the fuel from boiling transition during steady state operation and anticipated transients. The SLMCPR is established to provide that at least 99.9% of the fuel rods avoid boiling transition.

*Methodology*

[ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

#### 5.3.2.2 Monte Carlo Safety Limit Evaluation

*Methodology*

[ Proprietary Information Deleted ]

Treatment of Mixed Cores

[ Proprietary Information Deleted ]

#### 5.3.2.3 Channel Bow Evaluation

The influence of channel bow on CPR performance is accounted for in the SLMCPR evaluation. Nominal gaps are assigned to the fuel assemblies, and the Monte Carlo method is used to evaluated the impact of deviations in these gaps on CPR in establishing the SLMCPR. The required sensitivity of CPR on gap size is determined using the approved nuclear design codes (see Appendix A).

#### 5.3.2.4 Minimum Critical Power Evaluation for Reload Fuel

For reload applications, an ABB MCPR correlation accepted by the NRC is utilized in the plant on-line core supervision system for monitoring thermal limits as well as for design and licensing analyses. The correlation is provided to the utility for installation in the core supervision system (i.e. Plant Process Computer). The same correlation is utilized for design and licensing application in the thermal-hydraulic, nuclear, transient, and safety analyses.

For example, the CPR correlation for the SVEA-96 assembly currently being marketed in the U.S. for BWR applications has been accepted by the NRC and is documented in Reference 27.

#### 5.3.2.5 Minimum Critical Power Evaluation for Resident Fuel

*Methodology*

If the Resident fuel is an ABB design, the MCPR is treated in the same manner as for the Reload fuel assembly. An ABB MCPR correlation accepted by the NRC is utilized in the plant on-line core supervision system for monitoring against thermal limits as well as for design and licensing analyses.

If the Resident fuel is not an ABB design, an MCPR correlation provided by the fuel vendor is utilized in the plant on-line core supervision system for monitoring relative to thermal limits. Utilization of this correlation in the core supervision system is handled by the utility and the manufacturer of the resident fuel.

If the Resident fuel is not an ABB design, ABB may or may not have direct access to the accepted correlation for the Resident fuel. If ABB does have direct access to that correlation, it is used for design and licensing analyses. [ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

### 5.3.3 Hydraulic Compatibility

The process used to establish hydraulic compatibility of the Reload (ABB) fuel assembly and the Resident fuel in the Plant in which the Reload fuel is being installed can be summarized as follows:

[ Proprietary Information Deleted ]

### 5.3.4 Bypass, Water Cross, and Water Rod Flow

The bypass flow fraction is a function of the size of the bypass flow holes in the bottom nozzle. [ Proprietary Information Deleted ]

## 5.4 Methodology for Thermal Hydraulic Design Input to Reload Design and Safety Analyses

### 5.4.1 Thermal Hydraulic Design Input to Mechanical Design

Thermal-hydraulic information to support the following mechanical design evaluations described in Reference 37 are required for each plant application for the Reload fuel:

[ Proprietary Information Deleted ]

### 5.4.2 Thermal Hydraulic Design Input to Nuclear Design

The thermal and hydraulic models in the steady-state thermal and hydraulic code are incorporated in the ABB three-dimensional core simulator.

### 5.4.3 Thermal Hydraulic Design Input to Transient Analyses

In order to assure that the hydraulic modeling in the transient analyses calculational models are consistent with the nuclear and thermal hydraulic models, a matrix of calculated results for applicable core power and flow conditions using the models described in Section 5.3.1 are provided for verification of the transient analysis methods.

The burnup distributions and void histories from the nuclear design calculations at a given state point are used to provide one-dimensional cross section data for the transient analysis calculations. Power distributions and hydraulic information from the nuclear design calculations are used to initialize the transient analysis calculations. Therefore, the nuclear data and initial conditions in the transient analyses calculations are consistent with the predictions of the thermal and hydraulic models described in Section 5.3.1.

### 5.4.4 Thermal Hydraulic Design Input to LOCA Analyses

In order to assure that the hydraulic modeling in the LOCA analyses calculational models are consistent with the nuclear, thermal hydraulic, and transient analysis models, a matrix of calculated results for applicable core power and flow conditions using the models described in Section 5.3.1 are provided for verification of the LOCA analysis methods.

### 5.4.5 Thermal Hydraulic Design Input to CRDA Analyses

Direct input to the CRDA analysis is not routinely provided from the thermal and hydraulic models described in Section 5.3.1. [ Proprietary Information Deleted ]

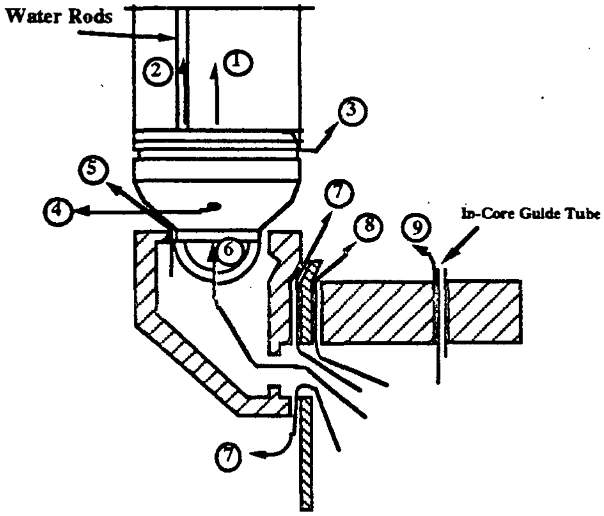
### 5.4.6 Thermal Hydraulic Design Input to Stability Analyses

In order to assure that the hydraulic modeling in the stability analyses calculational models are consistent with the nuclear, thermal hydraulic, and transient analysis models, a matrix of calculated results for applicable core power and flow conditions using the models described in Section 5.3.1 are provided for verification of the stability analysis methods. [ Proprietary Information Deleted ]

**TABLE 5-1 THROUGH TABLE 5-3**

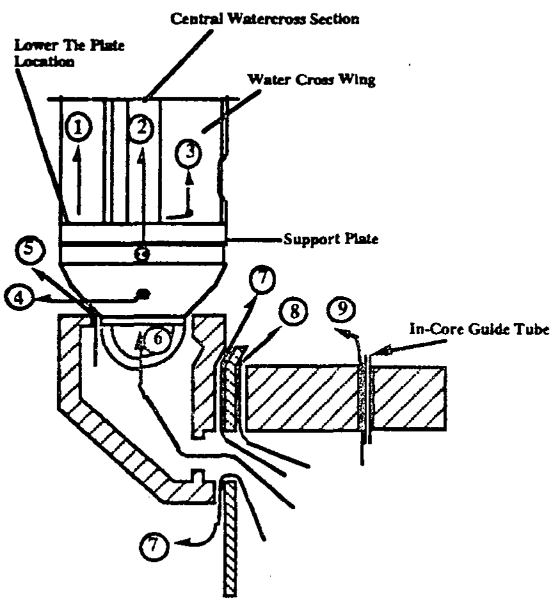
Proprietary Information Deleted

**RESIDENT ASSEMBLY**




1. Active coolant flow
2. Water rod flow
3. Leakage between channel and bottom nozzle
4. Bottom nozzle bypass holes
5. Leakage between bottom nozzle and fuel support piece
6. Bottom nozzle inlet flow
7. Leakage between control rod guide tube and fuel support piece
8. Leakage between control rod guide tube and core support plate
9. Leakage between in-core instrumentation guide tubes and core support plate

**SVEA-96 ASSEMBLY**




1. Active coolant flow
2. Flow through central canal
3. Flow through water cross wings (separate inlets)
4. Bottom nozzle bypass holes
5. Leakage between bottom nozzle and fuel support piece
6. Bottom nozzle inlet flow
7. Leakage between control rod guide tube and fuel support piece
8. Leakage between control rod guide tube and core support plate
9. Leakage between in-core instrumentation guide tubes and core support plate

Figure 5-1 Schematic of Flow Paths

**FIGURE 5-2**

Proprietary Information Deleted

# 6 RELOAD SAFETY ANALYSIS

## 6.1 Summary and Conclusions

*Summary*

This section describes the ABB reload safety analysis process for reload core applications and plant modifications. It also details the ABB reload safety analysis methodology used for boiling water reactors (BWR) in the United States.

The objective of the plant safety analysis is to demonstrate that the plant can operate without undue risk to the health and safety of the public. To assure that the plant safety analysis is comprehensive, a wide spectrum of events is evaluated as a part of the overall plant safety analysis. Each of these evaluations demonstrates conformance to the applicable event design bases and acceptance limits. The reload safety analysis process is used to update the plant safety analysis and can be used to demonstrate the acceptability of plant operation for any plant modification that requires a safety evaluation of the fuel, core, reactor coolant pressure boundary, or containment systems to satisfy the requirements of 10CFR50.59 (Reference 42), including the safety analysis required for the installation and operation of the plant core reload (see Figure 6-1).

The ABB reload safety analysis process categorizes safety analysis events and identifies potentially limiting events with respect to the plant design basis. The ABB reload safety analysis methodology defines the process of evaluating the potentially limiting events against acceptance limits and determining acceptable plant operating limits. (see Figure 6-2).

The reload safety analysis process uses the ABB BWR reload safety analysis design bases, methods, and methodology described in Sections 7, 8, and 9 (see also Table 1-2). The reload safety analysis is performed for the fuel and core design developed with the methods and methodology described previously in Section 3, 4, and 5.

*Conclusion*

It is concluded that:

(1) The ABB reload safety analysis process and methodology satisfies all of the applicable regulatory requirements and is consistent with regulatory requirements and guidance.

(2) The ABB reload safety analysis methodology is sufficiently flexible to incorporate the plant specific license commitments which potentially impact the reload safety analysis process.

(3) The ABB reload safety analysis methodology can be used to demonstrate the acceptability of the new core configuration consistent with operation in the allowable operating domain.

(4) The ABB reload safety analysis methodology can be used to demonstrate the acceptability of plant modifications affecting the allowable plant operating domain.

Therefore, the ABB reload safety analysis methodology can be used to update the current plant safety analysis consistent with the requirements of 10CFR50.59 (Reference 42).

## 6.2 Reload Safety Analysis Process

The plant safety analysis contains an analysis of the overall plant design and performance to determine the margin of safety during normal plant operation and transient conditions expected during the plant lifetime (anticipated operational occurrences) and demonstrates the adequacy of the plant design for the prevention of accidents and the mitigation of their consequences, should they occur. The plant safety analysis also contains the results of other analyses evaluated to demonstrate the plant capability to respond to selected events, performed in response to regulatory requirements and guidance and to specific licensing commitments. The results of the current plant safety analysis are contained in the updated final safety analysis for the plant as required by 10CFR50.71 (Reference 42, 10CFR50.71(e)). The event analyses contained in the updated final safety analysis report are used as a key input to the ABB reload safety analysis process.

The ABB reload safety analysis process is shown in Figure 6-2. The ABB reload safety analysis methodology builds on the current plant safety analysis to demonstrate that the plant can meet all of the applicable regulatory requirements and guidance, and plant specific licensing commitments, for the ABB reload application. This is accomplished through a reload safety analysis process that combines the results of generic safety analysis assessments and plant specific licensing commitment assessments. The ABB reload safety analysis process is intended to be consistent with licensee application of 10CFR50.59 (Reference 42). If the safety evaluation of the reload fuel and core design or plant operational modification demonstrates that there is no unreviewed safety question or required technical specification change, a written safety evaluation is prepared for retention by the plant licensee. If there is an unreviewed safety question or a technical specification change required, a license amendment request is prepared in accordance with the requirements of 10CFR50.90 (Reference 42).

Event assessment for reload safety analysis consists of the event categorization process and selection of potentially limiting events. The event categorization process uses the results of the typical plant event analyses and sensitivity studies using ABB reload safety analysis methodology to establish the events that are potentially limiting; that is those events that pose the most severe challenge to the event design bases and acceptance limits.

The event acceptance limits are those figures of merit that are used in the safety analysis process to demonstrate that the results of the specific analyses are acceptable. It is these potentially limiting events that are analyzed in the reload safety analysis process, using the ABB reload safety analysis methodology, to demonstrate the acceptability of the specific plant reload application or modified plant operational domain are acceptable.

Reload safety analyses methodology used for the plant specific reload safety evaluation includes the development of analysis inputs, use of analysis methods, and the evaluation of events supporting the allowable operating domain. The reload safety analyses inputs are based on inputs derived from the core and fuel design, as well as inputs provided by the plant licensee, that define plant and system performance. The analyses cover the allowable plant operating domain consistent with the current plant safety analyses and technical specifications. Changes to the allowable plant operating domain necessitated by the change to the core or fuel design or requested by the plant licensee are made in accordance with the requirements of 10CFR50.59 or 10CFR50.90, as applicable.

Key features of the ABB reload safety analysis process are summarized in Figure 6-2 and described in more detail below.

## 6.3 Reload Safety Analysis Events Assessment

In the reload safety analysis process, an assessment is made of safety analysis events. The generic assessments of safety analysis events are limited to the evaluation of anticipated operational occurrences, accidents, and other events that represent challenges to the fuel, core, reactor coolant pressure boundary, or containment systems. The list of generic safety analysis events that can potentially challenge the fuel, core, reactor coolant pressure boundary, or containment systems is provided in Table 6-1. In the generic assessment, the potentially limiting events for the typical plant safety analysis are identified. Identified are those events that can be impacted by a reload application or plant operational modification. It is the potentially limiting events that are evaluated as a part of the plant specific reload safety analysis.

In addition to the generic list of events identified in Table 6-1, it must be recognized that individual plants may have incorporated in their individual safety analysis an assessment of other events. These additional safety analysis events are reviewed for each plant specific application to determine if they can be potentially limiting with respect to the ABB reload application. The assessment of plant specific events is limited to events that have the potential to challenge the fuel, core, reactor coolant pressure boundary, or containment systems. Any of these additional events that are identified as potentially being limiting are included in the evaluations performed as a part of the plant specific reload safety analysis.

### 6.3.1 Event Categorization

As discussed in Section 6.2, the plant safety analysis contains the evaluation of a wide spectrum of postulated events and are consistent with the applicable event design bases and acceptance limits. Based on the relative event probabilities and failure assumptions, these events have been separated into three categories:

(1) Anticipated Operational Occurrences,

(2) Accidents, and

(3) Special Events.

Each of these event categories is initiated from some mode of normal planned operation. Planned Operation and each of these event categories are described in more detail below.

In the safety analysis process, the concept of design basis or potentially limiting events is frequently used. Design basis events are the events analyzed in the plant safety analysis that have the potential to establish design parameters for the plant or place constraints on plant operation. This event categorization is in accordance with the current regulatory requirements, including the General Design Criteria (Reference 42, Part 50, Appendix A). Further, it can be incorporated into other event categorizations such as that identified in Regulatory Guide 1.70 (Reference 47), which suggest events be categorized as incidents of moderate frequency, infrequent events, and limiting faults. The event categorization used in the ABB reload safety analysis process has been chosen because it is consistent with the selection of the event acceptance limits. These event acceptance limits (detailed in Section 6.4) are consistent with the relative event probabilities based on the applicable regulatory requirements.

Anticipated Operational Occurrences (AOOs) mean those conditions of normal operation which are expected to occur one or more times during the life of the plant and include but are not limited to generator load rejection, tripping of the turbine, isolation of the main condenser, and loss of all offsite power. To aid in the specific analysis, anticipated operational occurrences are evaluated based on a systematic evaluation enveloping credible events in this category.

Accidents are those postulated events that affect one or more of the barriers to the release of radioactive materials to the environment. These events are not expected to occur during the plant lifetime, but are used to establish the design basis for many systems.

Special Events are postulated occurrences that are analyzed to demonstrate different plant capabilities required by regulatory requirements and guidance, industry codes and standards, and licensing commitments applicable to the plant. As a result, they are not considered design basis events.

Planned Operation refers to normal plant operation under planned conditions within the normal operating envelope or planned operating domain in the absence of significant abnormalities. Following an event (Anticipated Operational Occurrence, Accident, or Special Event) Planned Operation is not considered to have resumed until the plant operating state is identical to a planned operating mode that could be attained had the event not occurred. As defined, Planned Operation can be considered as a chronological sequence:

- refueling outage
- criticality
- heatup
- power operation
- shutdown
- cooldown
- refueling outage.

Because Planned Operation provides the operating domain bounds for the initial conditions, it is an inherent part of the evaluation of each event and is not treated independently.

This section identifies all of the generic Anticipated Operational Occurrences, Accidents, and Special Events that are considered part of the ABB reload safety analysis process. The generic safety analysis events that are covered in the ABB reload safety analysis process are identified in Table 6-1. The potentially limiting events in each category are also identified and have been included in Table 6-2. It is these potentially limiting events that are evaluated for each plant reload application or change in plant operating domain, using the ABB reload safety analysis methodology. The results of these evaluations are included in the plant specific reload safety evaluation.

In addition, the plant safety analysis that incorporates an ABB reload safety analysis supporting a specific reload core or a change to the plant operating domain is reviewed to identify any events different than those generic events identified in Table 6-1 which may be potentially limiting. If any plant unique events are identified through this process, they are evaluated consistent with the plant specific commitments to determine if they can establish any plant operational constraints for the plant reload application or change in plant operating domain. The results of these evaluations are also included in the plant specific reload safety evaluation.

The next three sections discuss the categorization of events in the three groups: Anticipated Operational Occurrences, Accidents, and Special Events.

#### 6.3.1.1 Anticipated Operational Occurrences

To select the anticipated operational occurrences to be analyzed as a part of the plant safety analysis, eight nuclear system parameter variations are considered in the generic plant safety analysis process as possible initiating causes of challenges to the core, fuel, reactor coolant pressure boundary, and containment systems. These parameter variations are:

(1) Reactor Vessel Pressure Increase

(2) Reactor Core Coolant Temperature Decrease

(3) Reactor Core Positive Reactivity Insertion

(4) Reactor Vessel Coolant Inventory Decrease

(5) Reactor Core Coolant Flow Decrease

(6) Reactor Core Coolant Flow Increase

(7) Reactor Core Coolant Temperature Increase

(8) Reactor Vessel Coolant Inventory Increase

The eight parameter variations listed above include all the effects within the reactor system caused by anticipated operational occurrences that can challenge the integrity of the reactor fuel or other fission product barriers. The variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, challenges to barrier integrity are evaluated by groups according to the parameter variation initiating the plant challenge, which typically dominates the event response. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from reactor system pressure increases.

Event Single Failures

The specific events identified as anticipated operational occurrences in the safety analysis are generally associated with transients that result from single active component failures or single operator errors that reasonably can be expected during any mode of Plant Operation or are a conservative representation of those events.

Examples of single active component failures are:

(1) Opening or closing of any single valve (a check valve is not assumed to close against normal flow).

(2) Starting or stopping any single component.

(3) Malfunction or misoperation of any single control device.

(4) Any single electrical failure.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single reasonably expected erroneous decision. The set of actions is limited as follows:

(1) Those actions that could be performed by only one person.

(2) Those actions that would have constituted a correct procedure had the initial decision been correct.

(3) Those actions that are subsequent to the initial operator error and that affect the designed operation of the plant, but are not necessarily directly related to the operator error.

Examples of operator errors are:

(1) An increase in power above the established power flow limits by control rod withdrawal in the specified sequences.

(2) The selection of and attempt to completely withdraw a single control rod out of sequence.

(3) An incorrect calibration of an average power range monitor.

(4) Manual isolation of the main steam lines caused by operator misinterpretation of an alarm or indication.

Reactor Vessel Pressure Increase Events

Reactor vessel pressure increase events are initiated by a sudden reduction in steam flow such as a rapid valve closure. Increasing pressure collapses voids in the reactor core and increases core reactivity. This results in a positive feedback mechanism that further increases reactor system pressure and core power level which challenges the fuel and reactor coolant pressure boundary event acceptance limits. Examples of these events are:

- Generator Load Rejection with Bypass
- Generator Load Rejection without Bypass
- Turbine Trip with Bypass
- Turbine Trip without Bypass
- Pressure Regulator Failure - Closed
- Closure of One MSIV
- Closure of All MSIVs
- Loss of Condenser Vacuum

General plant performance to a sudden decrease in steam flow is an increase in reactor vessel and system pressure and core power. The initiating event usually will be terminated by a reactor trip. Scram is initiated by stop valve closure for a turbine trip, turbine control valve fast closure for generator load rejection, main steam line valve closure for isolation all of main steam lines, and neutron flux for pressure regulator failure - closed. The safety/relief valves and turbine bypass valves (unless assumed to be inoperative as a part of the event definition) will operate to limit the reactor system pressure rise.

This category of events can establish plant operating limits (i.e., minimum critical power ratio (MCPR)). [ Proprietary Information Deleted ] The actual event that is the most limiting is dependent on the plant specific performance characteristics and is determined specifically for each plant. The most limiting of these two events for specific plants is used as part of the process to establish the operating limit in the ABB reload safety analysis process. Also, for BWR/6 plants, it has been determined that the pressure regulator failure - closed also has the potential to establish the operating limits (i.e., MCPR). For BWR/6 plants, this event is evaluated as part of the process to establish the operating limits. Events other than the load rejection without bypass or the turbine trip without bypass and pressure regulator failure - closed events are not evaluated as part of the standard ABB reload safety analysis process.

Reactor Core Coolant Temperature Decrease Events

Decrease in core coolant temperature are those events that either increase the flow of cold water or reduce the temperature of the water being delivered to the reactor vessel. Core coolant (moderator) temperature reduction results in an increase in core reactivity, increasing the power level which threatens overheating of the fuel. Examples of these events are:

- Loss of Feedwater Heating
- Inadvertent RHR Shutdown Cooling Operation
- Inadvertent HPCI Start

General plant performance due to a core coolant temperature decrease is a corresponding increase in core power due to a negative core moderator void reactivity. Reactivity will increase when moderator voids decrease as the core coolant inlet temperature is reduced. A scram may occur on high thermal power or neutron flux. If no scram occurs, a new steady state power level will be reached and the operator will take steps to return to the operating conditions.

Large changes in core coolant temperature (e.g., 100 °F change in feedwater temperature or inadvertent HPCI system start) can lead to significant changes in critical power ratio (CPR). [ Proprietary Information Deleted ] Therefore, evaluation of the loss of feedwater heater in the ABB reload safety analysis process is considered necessary to determine if it is limiting and could be used to establish the operating limits. Analysis of the other events in this category demonstrates that they are easily controlled by operator action and do not pose a significant challenge to the event acceptance limits. Therefore, none of the other events in this category are evaluated as part of the standard ABB reload safety analysis process.

Reactor Core Positive Reactivity Insertion Events

Positive reactivity insertion events are generally caused by errors in the movement of control rods or in the loading of fuel assemblies during the refueling process. Localized positive reactivity insertions cause anomalies in power distribution and an increase in core power level which can potentially overheat the fuel. Examples of these events are:

- Control Rod Withdrawal Error (throughout Planned Operation)
- Control Rod Misoperation
- Incorrect Fuel Assembly Insertion

The plant performance due to reactivity and power distribution anomalies varies depending on the plant initial conditions and actual event. For the control rod withdrawal error, the assumed error is the continuous withdrawal of the maximum worth control rod with the core at rated conditions and in a state which maximizes the control rod worth. It is assumed that the operator has fully inserted the maximum worth control rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits in the fuel assemblies in the vicinity of the control rod to be withdrawn. The reactivity insertion rate is relatively slow, and the event is terminated either by the rod block monitor system or by the complete withdrawal of the control rod if the rod block monitor setpoint is not reached. The control rod withdrawal error may establish the MCPR operating limit. Therefore, this event is evaluated in the ABB reload safety analysis process.

The incorrect fuel assembly insertion is the erroneous insertion of a fuel assembly into an incorrect location or orientation. The error is identified and corrected during the core verification process. The reactor remains subcritical throughout the event. (The fuel loading error is an design base accident discussed in Section 6.3.1.2) Control rod misoperation is the erroneous drifting of a control rod during normal plant operation due to a failure in the control rod control system. This event is alarmed and terminated by operator action. [ Proprietary Information Deleted ]

Events in this category other than the control rod withdrawal error are not evaluated in the standard ABB reload safety analysis process.

Reactor Vessel Coolant Inventory Decrease Events

Reactor vessel coolant inventory decrease events are the result of a situation where the steam flow rate is greater than the feedwater input flow. Losses in reactor coolant inventory cause a decrease in reactor water level, which threatens overheating of the fuel, and a decrease in coolant temperature, which leads to a mild depressurization. Examples of these events are:

- Inadvertent Safety/Relief Valve Opening
- Pressure Regulator Failure - Open

- Loss of AC Power
- Loss of Feedwater Flow

General plant performance for this category of events is a decrease in reactor vessel water level and a decrease in core coolant temperature as a result of the steam and feedwater flow mismatch which leads to a mild depressurization. The event may be terminated by a scram on low water level if feedwater cannot respond to maintain level. If feedwater maintains level, a new steady state operating condition is established until operator action is taken to control the event and return to Planned Operation.

This category of events is less severe than others and is generally considered non-limiting. This conclusion is verified by the evaluations performed in the plant safety analysis. Therefore, none of these events are evaluated as part of the standard ABB reload safety analysis process.

Reactor Core Coolant Flow Decrease Events

Reactor core coolant flow decrease events decrease the ability of the reactor coolant to remove the heat generated in the core which has the potential for overheating of the fuel. Examples of these events are:

- Trip of One Recirculation Pump
- Trip of All Recirculation Pumps
- Recirculation Flow Control Failure - Decreasing Flow

General plant performance with a decrease in reactor coolant flow rate is a decrease in core power level due to increased moderator voids, and an increase in water level due to the swelling effects of increasing moderator voids. The vessel water level increase may be sufficient to cause a turbine trip through actuation of the turbine protection features. The turbine trip will cause a reactor scram, terminating the event. For most events, the feedwater controller will prevent high water level and avoid the turbine trip. Any increase in system pressure is limited by the turbine bypass system or safety/relief valve operation. If no scram occurs, the power level will drop to a value that maintains a reactivity balance for the new steam void content. [ Proprietary Information Deleted ]

This category of events is less severe than others and is generally considered non-limiting. This conclusion is verified based on the analyses performed as a part of the plant safety analysis process. Therefore, none of these events are analyzed as part of the standard ABB reload safety analysis process.

Reactor Core Coolant Flow Increase Events

Reactor core coolant flow increase events result in an increase in recirculation flow rate. Increases in reactor core coolant flow rate result in a decrease in core voids and an increase in core reactivity. An increase in core reactivity increases core power level and threatens overheating of the fuel. Examples of these events are:

- Recirculation Flow Controller Failure - Increasing Flow
- Startup of an Idle Recirculation Loop

General plant performance for an increase in reactor coolant flow is a corresponding increase in core reactivity and power due to the reduction in voids as the coolant flow increases. If the reactivity increase is rapid, such as for the startup of an idle recirculation loop, the event will be terminated by a scram on high neutron flux. If the reactivity increase is slower due to a slow increase in recirculation flow, a new steady state operating condition can be established until operator action is taken to terminate the event.

[ Proprietary Information Deleted ] Therefore, for this category of events, only the slow increase in recirculation flow rate is considered in establishing core operating limits at reduced core flow and power levels and is evaluated as a part of the reload safety analysis process.

Reactor Core Coolant Temperature Increase Events

Core coolant temperature increase events are those that increase the temperature of the water being delivered to the reactor vessel. An increase in core coolant temperature increases reactor pressure and threatens the reactor coolant pressure boundary. These events could also lead to fuel cladding damage due to overheating. An example of this type of event is:

- Failure of RHR Shutdown Cooling

General plant performance for a failure of the shutdown cooling mode of the Residual Heat Removal (RHR) system is a slow increase in pressure followed by isolation of the shutdown cooling system. The event is terminated by operator action.

Loss of shutdown cooling is easily controlled by operator action. This conclusion is verified based on the analyses performed as a part of the plant safety analysis. Therefore, this event is not evaluated as part of the standard ABB reload safety analysis process.

Reactor Vessel Coolant Inventory Increase Events

Excess of coolant inventory events can result from a feedwater flow increase greater that the steam production rate due to a feedwater controller failure in maximum demand position. Increasing the reactor vessel water level could result in excessive moisture carryover to the main turbine, which results in the actuation of the turbine protective devices (e.g., turbine trip). In addition, the coolant inventory increases result in a core wide power increase prior to the turbine trip due to the transient effect of adding cooler water and reducing steam content in the core region. [ Proprietary Information Deleted ] An example of this event is:

- Feedwater Controller Failure - Maximum Demand

General plant performance for the feedwater controller failure to the maximum demand position is similar to a combination of a decrease in coolant temperature followed by a pressurization event. There is initially a core wide power increase due to the effects of the increased feedwater flow. This is followed by a turbine trip initiated by high water level.

The event is essentially the same as a turbine trip with bypass initiated from a higher power level, and it may establish the operating limits. Therefore, feedwater controller failure - maximum demand is evaluated in the ABB reload safety analysis process.

#### 6.3.1.2 Design Bases Accidents

Accidents are defined as those postulated events that affect one or more of the radioactive material barriers. These events are not expected to occur during the plant lifetime, but are used to establish the design basis for certain systems. Accidents have the potential for releasing radioactive material as follows:

(1) From the fuel with the reactor system process barrier, primary containment, and secondary containment initially intact.

(2) Directly to the primary containment.

(3) Directly to the secondary containment with the primary containment initially intact.

(4) Directly to the secondary containment with the primary containment not intact.

(5) Outside the secondary containment.

The effects of the various accident types are investigated, with a consideration for the full spectrum of plant conditions, to examine events that result in the release of radioactive material. The accidents resulting in radiation exposures greater than any other accident considered under the same general accident assumptions are typically designated design basis accidents. Examples of accident types are as follows.

(1) Component Mechanical Failure: Mechanical failure of various components leading to the release of radioactivity from one or more radioactivity release barriers. These components encompass components that do not act as radioactive material barriers. Examples of mechanical failures are breakage of the coupling between a control rod drive and the control rod, failure of a crane cable, and failure of a spring used to close an isolation valve.

(2) Overheating Fuel Barrier: This type includes overheating as a result of reactivity insertion or loss of cooling. Other radioactive material barriers are not considered susceptible to failure from any potential overheating situation.

(3) Pressure Boundary Rupture: Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the reactor system process barrier. Such rupture is assumed only if the component postulated to rupture is subjected to significant pressure.

The accidents considered in the generic plant safety analysis that can be significantly impacted by the introduction of reload fuel or a change to the plant operating domain include:

(1) Pipe Breaks Outside of Primary Containment

(2) Loss of Coolant Accident

(3) Control Rod Drop Accident

(4) Fuel Handling Accident

(5) Fuel Loading Errors

(6) Recirculation Pump Failure Accident

(7) Instrument Line Breaks

Additional Single Failure

To increase the conservatism in the evaluation of accidents, an Additional Single Failure in a component that is intended to mitigate the consequences of the postulated event is assumed to occur coincident with the initiation of the accident. This single failure is in addition to the failures that are an inherent part of the postulated accident definition. The single failures considered include occurrences such as electrical failure, instrument error, motor stall, breaker freeze-in, or valve malfunction. Highly improbable Additional Single Failures, such as pipe breaks, are not assumed to occur coincidentally with the postulated accident. The single failures are selected to be sufficiently conservative so that they include the range of potential effects from any other single failure. Thus, there exists no other Additional Single Failure of the types under consideration that could increase the calculated radiological effects of the design basis accidents.

Pipe Breaks Outside Primary Containment

Pipe breaks outside primary containment can result in the release of radioactivity directly to the environment. These piping systems which penetrate the primary and secondary containments are connected to the reactor coolant pressure boundary during normal operation. These pipe breaks include both main steam and feedwater systems. The consequences of the spectrum of postulated pipe break locations is bounded by the main steam line break.

The main steam line break is the postulated instantaneous complete severance of one main steam line. This accident results in the maximum amount of reactor coolant being released directly to the environment. The initial plant response to a main steam line break is a rapid depressurization of the reactor and closure of the MSIVs due to high steam flow. The reactor is initially shut down by the increase in void fraction due to the depressurization. The event is terminated by closure of the MSIVs and trip of the reactor due to the main steam line isolation valve closure.

The change in core thermal hydraulic conditions represents a challenge to the fuel cladding, and the release of coolant directly to the environment represents a significant radiological effect. Therefore, the analysis of this event in the plant safety analysis is required to demonstrate conformance to accident limits. For reload fuel applications, sensitivity studies have demonstrated that there are no significant changes to the core thermal hydraulic conditions. Further, the core coolant activity is limited by the plant technical specifications, which are not changed as a result of the reload. Therefore, this event is not evaluated as a part of the standard ABB reload safety analysis process.

Loss of Coolant Accident

The loss of coolant accident has been selected to bound the consequence of events that release radioactivity directly to the primary containment as a result of pipe breaks inside the primary containment. The reactor coolant pressure boundary contains a number of different sizes, lengths, and locations of piping. Failure of this piping results in loss of coolant from the reactor and discharge of the coolant directly to the primary containment.

The loss of coolant accident is the postulated break of any size piping in the reactor coolant pressure boundary up to and including the rapid circumferential failure of the reactor recirculation system piping. By evaluating the entire spectrum of postulated break sizes, the most severe challenge to the emergency core cooling system (ECCS) and primary containment can be determined.

The initial plant response to a large loss of coolant accident is a depressurization of the reactor and decrease in water level followed by a trip of the reactor, closure of the primary containment isolation valves, initiation of the ECCS, and a low reactor water level or high containment pressure that causes isolation of the secondary containment (if applicable) and initiation of the standby gas treatment system. The reactor is initially shut down by the increase in void fraction due to the depressurization which is followed by the automatic insertion of the control rods. The event is terminated by the closure of the containment isolation valves, actuation of the ECCS and operation of the other required safety systems.

The loss of coolant can lead to significant fuel cladding failures and the release of substantial amounts of radioactivity to the primary containment. The performance of the ECCS is critical in limiting the fuel failures, and the performance of the primary and secondary containments is key in limiting the dose consequences. Therefore, analysis of this event in the plant safety analysis is required to demonstrate conformance to accident limits. This event is evaluated for each plant modification with potential to significantly change the core thermal hydraulic or radiological input parameters, or significantly change the ECCS, primary containment, or secondary containment performance characteristics.

For the introduction of each new reload fuel type, appropriate analyses must be performed to establish the core operating limits for the new fuel. If no new fuel types are introduced, an evaluation of the loss of coolant accident is not required by the ABB reload safety analysis process.

Control Rod Drop Accident

The control rod drop accident represents the greatest potential for adding reactivity to the core at a relatively high rate. Therefore, the control rod drop accident has been chosen to bound the consequences of the reactivity insertion events.

The control rod drop accident is the postulated dropping of a fully inserted and decoupled control rod at its maximum velocity. The dropped control rod is assumed to have the maximum incremental worth rod consistent with the constraints on control rod patterns. It is assumed that the event can occur in any operating mode in which the reactor is not shutdown.

The initial plant response to a control rod drop accident is a prompt power burst which is terminated initially by the core negative reactivity feedback due primarily to Doppler. Final reactor shutdown is achieved by control rod scram initiated by high neutron flux.

The postulated rapid insertion of large amounts of reactivity can lead to significant fuel cladding failures and increases in reactor pressure. Therefore, analysis of this accident in the plant safety analysis is required to demonstrate conformance to accident acceptance limits. The radiological consequences assumed by plant safety analysis and the fuel integrity acceptance limits are confirmed acceptable for ABB reload applications. If required, plant safety analysis is modified to reflect the radiological consequences of the accident. In the ABB reload safety analysis process, the control rod drop accident is evaluated for each reload to demonstrate conformance to the applicable event acceptance limits.

Fuel Handling Accident

Fuel handling accidents can occur which will release radioactivity directly to the plant confinement (primary containment, secondary containment, or fuel building depending on the containment design). The fuel handling accident, or refueling accident, is consistent with the licensing basis for fuel handling equipment which considers failures such as the postulated dropping of a fuel assembly and the fuel grapple mast and head from the maximum height allowed by the fuel handling equipment. For the limiting event, the fuel assembly and fuel grapple are assumed to drop onto the core causing the maximum damage to the highly exposed fuel.

The plant response to this event is the isolation of the containment or building and initiation of the standby gas treatment system.

The postulated fuel handling accident can lead to a significant number of fuel failures and subsequent release of radioactivity to the containment or building. Therefore, analysis of this event in the plant safety analysis is required to demonstrate conformance to accident limits. In the ABB reload safety analysis process, the fuel handling accident is analyzed for each new fuel design to establish the maximum number of fuel rods that can be damaged as a result of this accident. The plant safety analysis then can be modified, if necessary, to reflect the radiological consequences of this event. This event is not reanalyzed for a specific reload unless a new fuel design is introduced or a modification is made to the fuel handling equipment that can increase the severity of this event.

Fuel Loading Errors

The fuel loading error (also specified as a misplaced assembly accident in other sections of this report) is the postulated occurrence of loading one fuel assembly in an improper location (mislocated) or in an improper orientation (rotated). Further, it is assumed that the improper loading of a fuel assembly is not discovered and corrected as a result of the core verification program, and the plant is operated throughout the operating cycle assuming that the design core configuration has been correctly implemented. Because of the low probability of these events, they are considered accidents in the safety analysis process. [ Proprietary Information Deleted ] These events are considered as potentially limiting events in the plant safety analysis. In the ABB reload safety analysis process, fuel loading errors are evaluated for each reload to demonstrate conformance to the applicable event acceptance limits.

Recirculation Pump Failure Accident

Recirculation pump seizure or recirculation pump shaft break accidents are the events which result in the most rapid rate of coolant flow reduction in a BWR. Therefore, the recirculation pump seizure and shaft break accidents have been selected to represent accidents in this category.

The recirculation pump seizure or recirculation pump shaft break result in a rapid decrease in core flow due to the large hydraulic resistance introduced by the recirculation pump failure. The initial plant response is a rapid reduction in core flow with a corresponding reduction in core power level. The plant will generally settle out at a new steady state condition until operator action is taken to terminate the event.

[ Proprietary Information Deleted ] Therefore, the ABB reload safety analysis process does not require reanalysis of these events.

Instrument Line Breaks

Instrument line breaks are potentially non-isolable small line breaks that can result in the release of radioactivity directly to the reactor building. The instrument lines are connected to the reactor coolant pressure boundary during normal operation.

This accident results in the maximum amount of reactor coolant being released from a non-isolable line. The initial plant response to an instrument line break is continued power operation until operator action can be taken to limit the fluid loss. Once the operator has identified the occurrence of an instrument line break, action will be taken to shut the reactor down and, if necessary, depressurize the reactor to limit the loss of inventory.

[ Proprietary Information Deleted ] Therefore, this event is not evaluated as a part of the standard ABB reload safety analysis process.

### 6.3.1.3 Special Events

Special events are evaluated to demonstrate plant capabilities required by regulatory requirements and guidance, industry codes and standards, and licensing commitments. The special events considered in the plant safety analysis are dependent on the goals of the analysis. The following special analyses are considered a part of the generic plant safety analysis that can be impacted in by a ABB reload application.

(1) Core Thermal-Hydraulic Stability

(2) Reactor Overpressure Protection

(3) Shutdown Without Control Rods

(4) Anticipated Transients Without Scram

Core Thermal-Hydraulic Stability

Core thermal-hydraulic stability analyses are performed to satisfy the regulatory requirement that no divergent power oscillations occur that cannot be detected or suppressed before exceeding specific acceptable fuel design limits. There are three sources of core thermal-hydraulic stability: (1) plant system, (2) coupled nuclear/hydrodynamic; and (3) channel hydrodynamic. Stability is evaluated for each plant modification with potential to significantly change the core thermal hydraulic performance characteristics. The plant safety analysis demonstrates that stability due to the plant system is not significantly changed by the introduction of reload fuel. In the ABB reload safety analysis process, core thermal-hydraulic stability evaluations are performed as required by the plant specific stability licensing bases. As required for the specific plant reload application, coupled nuclear/hydrodynamic (core) and channel hydrodynamic (channel) stability are evaluated to demonstrate conformance to the applicable event acceptance limits. Where applicable, plant specific licensing commitments are followed with regards to stability evaluations.

Reactor Overpressure Protection

The overpressure protection analysis is performed to demonstrate conformance to the ASME Code overpressure requirements (Reference 49). The overpressure protection analysis is the simulation of the most severe pressurization event with no credit allowed for a scram associated with the initiating event. In the plant safety analysis process, a closure of all MSIVs with a neutron flux scram (MSIV position scram assumed failed) is analyzed, unless a plant-specific licensing commitment has been made to analyze a different event.

The plant performance for this event is a rapid increase in reactor vessel pressure and core power. The reactor is scrammed on high neutron flux and the recirculation pumps are tripped on high pressure. The safety/relief valves operate to limit the reactor vessel pressure rise.

The event is analyzed in the plant safety analysis to demonstrate conformance to the ASME Code overpressure limits for the reactor vessel and reactor coolant pressure boundary. Therefore, it is not necessary for this event to assess the effects on the fuel or other components. This event is evaluated for plant modifications with potential to significantly change the core thermal hydraulic performance characteristics or changes the characteristics of the safety/relief valves. In the ABB reload safety analysis process, the overpressure protection capability is evaluated for each reload application to demonstrate conformance to the applicable event acceptance limits.

Shutdown Without Control Rods

For the shutdown without control rods event, the standby liquid control system capability analysis is performed to demonstrate that the core can be made subcritical in the cold condition without movement of the control rods. In this analysis, it is assumed that the core is made subcritical (in a xenon free state) from full power and minimum control rod inventory (at equilibrium xenon) by action of the standby liquid control system to inject liquid poison into the reactor.

The standby liquid control system capability analysis is required in the plant safety analysis to demonstrate the capability of the plant to reach cold shutdown without dependence on the control rods. This analysis demonstrates compliance with General Design Criteria 26 and 27 (Reference 42, Part 50, Appendix A). This event is evaluated for plant modifications with the potential to significantly change the core overall core reactivity or to change the standby liquid control system performance characteristics. In the ABB reload safety analysis process, the standby liquid control system capability is evaluated for each reload application to demonstrate conformance to the applicable event acceptance limit.

Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) are defined as the postulated occurrence of an anticipated transient which reaches a reactor protection system setpoint (or requires a manual scram to terminate the event) and for which there is a failure of sufficient control rods to insert to shut the reactor down. For the purpose of this set of events, anticipated transients are generally defined as those conditions of operation expected to occur one or more times during the service life of the plant. Because an ATWS event would require multiple failures, it is considered beyond the plant design basis and is analyzed to demonstrate conformance to 10CFR50.62 (Reference 42).

By its definition, ATWS represents a spectrum of events due to the number of different potential event initiators. The spectrum of event initiators is generically evaluated to establish which ones are potentially most limiting. The most limiting initiators generally are caused by a rapid reduction in steam flow (rapid pressurization events) or events that can evolve to a rapid pressurization event during the course of the transient. These potentially limiting transients are analyzed as part of the plant safety analysis.

Plant performance for ATWS events is highly dependent on the event initiators. For rapid pressurization events, there is a rapid increase in reactor vessel and reactor coolant pressure boundary pressure and core power. The pressure and power increase is limited by the automatic recirculation pump trip (ATWS-RPT) on high reactor pressure and operation of the safety/relief valves. Reactor shutdown is accomplished by manual initiation of the standby liquid control system. In the plant safety analysis, bounding analyses are used in the analysis of ATWS. Evaluations have been performed to assure that the ATWS modifications are adequate for ABB reload fuel. Therefore, this event does not have to be reanalyzed unless the bounding assumptions are exceeded due to a plant modification. Therefore, these events are not evaluated as a part of the standard ABB reload safety analysis process.

### 6.3.2 Potentially Limiting Events

In the safety analysis process, it is not required nor practical to reanalyze all of the events that are considered a part of the plant safety analysis for each plant modification. Only the potentially limiting events associated with the specific plant modification are evaluated for that modification. The approach of evaluating only potentially limiting events is an inherent part of the ABB reload safety analysis process.

To identify the potentially limiting events, each event in the plant safety analysis is evaluated to determine that, for an ABB reload application or for a change in the plant operating domain, the event analysis results can establish a core operating limit or exceed an event acceptance limit. The events that have this potential are evaluated for each reload application as a part of the process for establishing the cycle specific core operating limits.

Because of the differences between plant specific safety analyses, ABB has developed a process to determine the potentially limiting events that assure coverage of all applicable potentially limiting events for plants utilizing ABB reload safety analysis methodology. This process involves the use of generic safety analysis events supplemented by events associated with plant specific licensing commitments. This process provides assurance that all applicable plant safety analysis events are considered for each use of ABB reload application or change to plant operating domain justified by the use of ABB reload safety analysis methodology.

In this process, a set of generic safety analysis events that are common to essentially all BWR safety analyses have been identified. This set of events has been provided as Table 6-1. Based on the information provided in Sections 6.3.1.1 through 6.3.1.3, the potentially limiting events within the set of generic safety analysis events have been established. These generic potentially limiting events are identified in Table 6-2. This process establishes the minimum set of events evaluated for each application of the ABB reload safety analysis methodology.

As shown in Table 6-2, the following generic safety analysis events are evaluated each reload: the most limiting of turbine trip or generator load rejection without bypass; loss of feedwater heating; control rod withdrawal error; feedwater controller failure - maximum demand; fuel loading error; control rod drop accident, standby liquid control system capability; and overpressure protection. In addition, the pressure regulator failure - closed is evaluated for BWR/6 plants. The recirculation flow controller failure - increasing flow is evaluated as part of the process for establishing core operating limits at reduced flow and core power levels.

The fuel handling accident is evaluated for the plant for each new fuel design. [ Proprietary Information Deleted ]

The loss of coolant accident is evaluated for the initial application of ABB reload fuel and then only supplemented to establish the core operating limits associated with new fuel types. [ Proprietary Information Deleted ] Core thermal-hydraulic stability is evaluated to the extent as required by the plant specific licensing commitments.

As also shown in Table 6-2, the generic, potentially limiting events discussed above, are supplemented, as necessary, to include events that are associated with plant specific licensing commitments. [ Proprietary Information Deleted ]

## 6.4 Design Bases and Acceptance Limits

Event acceptance limits are the figures of merit for the plant safety analysis to demonstrate compliance with plant design bases. The results of the plant safety analysis for each event analyzed must demonstrate conformance to the applicable event acceptance limits. The event acceptance limits for the plant safety analysis identified cover the three categories of events: (1) Anticipated Operational Occurrences, (2) Design Based Accidents, and (3) Special Events. The event acceptance limits are based on a qualitative assessment of the relative probability of the various events with the more probable events having more restrictive limits. Further, because of the differences in event signatures, the event acceptance limits for accidents and other events are identified for each event in the category. The event design bases and acceptance limits are discussed in general below. The event acceptance limits are summarized in Table 6-3. Specifics of the event design bases and acceptance limits along with the analysis methodology for each generic event evaluated in the ABB reload safety analysis methodology are discussed in detail with the respective events in Section 7, 8, and 9.

### 6.4.1 Anticipated Operational Occurrences

For anticipated operational occurrences, there are four basic event acceptance limits: (1) radioactive effluents; (2) specified acceptable fuel design limits (SAFDLs); (3) peak reactor vessel pressure; and (4) suppression pool temperature.

Radioactive Effluents

The limits for radioactive effluents are those contained in 10CFR20 (Reference 42). By demonstrating that the specified acceptable fuel design limits are not exceeded during Anticipated Operational Occurrences, conformance to this limit is demonstrated in the safety analysis. This conclusion holds because there are only four types of Anticipated Operational Occurrences that can lead to radioactive releases except through the normal operational release paths. These types of release are: (1) momentary pressure relief (e.g., turbine trip or generator load rejection with bypass); (2) power isolation (e.g., MSIV closure while operating at power); (3) inadvertent opening of a safety/relief valve while at full power; and (4) MSIV closure with control rod inserted while the reactor is being cooled down. The radiological consequences of the events are minimal because there are no calculated fuel failures during these events and the reactor coolant activity is contained within the reactor vessel and primary containment. As a result, the offsite doses are negligible, and radiological evaluations are considered unnecessary. Therefore, no additional radiological evaluations are required for Anticipated Operational Occurrences as long as the SAFDL event acceptance limit is satisfied.

Specified Acceptable Fuel Design Limits

SAFDLs are used as an event acceptance limit for Anticipated Operational Occurrences to demonstrate that there are no calculated fuel failures. [ Proprietary Information Deleted ]

Peak Reactor Vessel Pressure

The peak reactor vessel pressure limit is used as an event acceptance limit for Anticipated Operational Occurrences conditions are not exceeded. The ASME Code (Reference 49) upset limit of 110% of the reactor pressure vessel design pressure is used for this limit. The overpressure protection event analysis, evaluated in the ABB reload safety analysis process, bounds all AOO events with regard to this acceptance limit.

Suppression Pool Temperature

The suppression pool temperature is used as an event acceptance limit for Anticipated Operational Occurrences to assure that the suppression pool is available to function as a heat sink for events involving operation of the safety/relief valves. The heat capacity temperature limit identified in the plant specific emergency operating procedures is used for this limit. [ Proprietary Information Deleted ]

### 6.4.2 Design Bases Accidents

As described previously, the event acceptance limits for accidents are dependent on the specific event being analyzed. The specific accidents considered in the safety analysis include: (1) pipe breaks outside of primary containment; (2) loss of coolant accident; (3) control rod drop accident; (4) fuel handling accident; (5) fuel loading errors; (6) recirculation pump failure; and (7) instrument line breaks.

Pipe Breaks Outside of Primary Containment

For pipe breaks outside of containment, the figures of merit are the onsite and offsite radiological effects. The event acceptance limit for offsite radiological effects is the guideline dose values presented in 10CFR100, and the event acceptance limits for onsite radiological effects is the limits identified in General Design Criterion (GDC) 19 (Reference 42, 10CFR50 Appendix A).

Loss of Coolant Accident

For the loss of coolant accident, there are three basic event acceptance limits: (1) the onsite and offsite radiological effects; (2) the ECCS acceptance criteria of 10CFR50.46 (Reference 42); and (3) the primary containment design limits.

The event acceptance limit for offsite radiological effects is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological effects is the limits identified in the GDC 19.

There are five event acceptance limits associated with the ECCS acceptance criteria: (1) the calculated maximum fuel element cladding temperature is not to exceed 2200 °F; (2) the calculated local oxidation of the cladding is not to exceed 0.17 times the local cladding thickness before oxidation; (3) the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam is to not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react; (4) calculated changes in core geometry are such that the core remains amenable to cooling; and (5) after any calculated successful operation of the emergency core cooling system, the calculated core temperature shall be maintained for the extended period of time required by the long-lived radioactivity remaining in the core.

The event acceptance limit for the primary containment design limits is the ASME Code upset limit of a peak containment pressure, typically 10% greater than the containment design pressure. [ Proprietary Information Deleted ]

Control Rod Drop Accident

For the control rod drop accident, there are two basic event acceptance limits: (1) the onsite and offsite radiological effects and (2) the peak fuel enthalpy limit.

The event acceptance limit for offsite radiological effects is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological effects is the limits identified in the GDC 19.

The peak fuel enthalpy limit is a calculated peak average pellet enthalpy of 280 cal/gm to assure the integrity of the reactor coolant pressure boundary. For radiological evaluation purposes, all fuel rods evaluated to exceed a peak average pellet enthalpy of 170 cal/gm are assumed to fail.

Fuel Handling Accident

For the fuel handling accident, the figures of merit are the onsite and offsite radiological effects. The event acceptance limit for offsite radiological effects is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological effects is the limits identified in GDC 19.

Fuel Loading Error

For fuel loading errors, the figure of merit is the MCPR safety limit. The specific value for this limit is core and fuel design dependent and is identified in the plant reload application. This limit is used to preclude long term operation with the potential for fuel in transition boiling.

Recirculation Pump Failure Accident

For the recirculation pump failure accident, the figures of merit are the onsite and offsite radiological effects. The event acceptance limit for offsite radiological effects is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological effects is the limits identified in GDC 19.

Instrument Line Break

For the instrument line break, the figures of merit are the onsite and offsite radiological effects. The event acceptance limit for offsite radiological effects is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological effects is the limits identified in GDC 19.

### 6.4.3 Special Events

As described above the event acceptance limits for special events are dependent on the specific event being analyzed. The specific accidents considered in the safety analysis include: (1) core thermal-hydraulic stability; (2) overpressure protection; (3) shutdown without control rods; and (4) anticipated transients without scram.

Core Thermal-Hydraulic Stability

The event acceptance limits for thermal-hydraulic stability are the same as the specified acceptable fuel design limits identified for anticipated operational occurrences. Compliance with the event acceptance limit for stability is demonstrated by plant specific licensing commitment.

Overpressure Protection

For the overpressure protection, the ASME peak reactor vessel pressure limit is used as an event acceptance limits to demonstrate that the reactor coolant pressure boundary design conditions are not exceeded. The ASME Code upset limit of 110% of the reactor pressure vessel design pressure is used for this limit.

Shutdown Without Control Rods

For the shutdown without control rods, the figure of merit is a reactor criticality less than one ($k_{eff} < 1.0$) at the most reactive temperature. This value provides assurance that the reactor will be subcritical at the most reactive temperature.

Anticipated Transients Without Scram

For ATWS, there are five basic event acceptance limits: (1) reactor coolant pressure boundary pressure limit; (2) containment pressure limit; (3) coolable geometry; (4) offsite radiological effects; and (5) equipment availability.

The event acceptance limit for the reactor coolant pressure boundary pressure limit is the ASME Code emergency limit of a peak reactor vessel pressure of 120% of the reactor pressure vessel design pressure in gauge pressure.

The event acceptance limit for containment pressure is the ASME Code upset limit of a peak containment pressure 10% greater than the containment design pressure.

The event acceptance limit for the maintenance of a coolable geometry is a calculated peak fuel cladding temperature of 2200 °F.

The event acceptance limit for offsite radiological effects is the guideline dose values of 10CFR100. It can be demonstrated that the guideline dose values of 10CFR100 are satisfied by demonstrating that the first three event acceptance limits for ATWS are met. This approach limits the fuel rod failures to less than 100% perforations, which provides assurance that the dose will be less than that calculated for the loss of coolant accident.

The event acceptance limit for equipment availability is to provide a high degree of assurance that it functions in the environment predicted to occur as a result of the ATWS event.

## 6.5 Plant Allowable Operating Domain

One of the primary objectives of the reload safety analysis process is to demonstrate the capability of the plant to operate safely within the allowable operating domain as defined, in part, by the power/flow map for the specific plant being evaluated. For the ABB reload safety analysis process, the allowable operating domain is defined by the current plant safety analysis. The allowable operating domain is provided as an analysis input by the plant licensee. Any changes to the allowable operating domain desired by the plant licensee are treated as a plant modification in the reload safety analysis process.

The allowable operating domain considered in the reload safety analysis process may include both operating flexibility improvements and MCPR margin improvements. Operating flexibility options include: (1) extensions to the originally licensed power/flow map such as load line limit analyses (LLLA), extended load line limit analyses (ELLLA), increased core flow operation (ICF), or maximum extended operating domain (MEOD); (2) single loop operation; (3) feedwater temperature reduction; (4) average power range monitor - rod block monitor technical specification (ARTS) program; and (5) end of cycle coastdown. Margin improvement options include: (1) end of cycle recirculation pump trip (EOC RPT); (2) average power range monitor simulated thermal power scram; (3) exposure dependent limits; and (4) improved scram time.

In the ABB reload safety analysis process for an ABB reload application, the analysis of the allowable operating domain is performed consistent with the analysis requirements established by the current safety analysis. This results in evaluations being performed for all potentially limiting conditions within the allowable operating domain, consistent with those identified to establish the current plant licensing basis. For extensions to the allowable operating domain, the extension is treated as a plant modification and all potentially limiting events for new operating domain are evaluated at their most limiting allowable operating condition. These evaluations then become the basis for the evaluation of future reloads.

## 6.6 Reload Safety Analysis Methodology

The reload safety analysis methodology is used to perform the required safety analysis evaluations associated with an ABB plant reload application or plant operational modification. A detailed view of the ABB reload safety analysis process including the reload design and safety analysis methods, reload safety analyses, primary input data and interfaces, and reload operating limits is shown on Figure 6-3. Evaluations using the reload safety analysis methodology result in establishing the operating limit MCPR; demonstrating the acceptability of operating limit LHGR; and demonstrating conformance to the event acceptance limits for reactor vessel pressure, standby liquid control system capability, control rod drop accident, thermal-hydraulic stability, and loss of coolant accident.

An overview of the reload safety analysis methodology used for a plant specific safety evaluations, is given below. Specific details of the ABB reload safety analysis methodology for the event identified in Table 6-3, are given in Section 7, 8, and 9.

### 6.6.1 Methods and Analyses

The primary methods used in the overall reload safety analysis process include: (1) the lattice physics nuclear design methods; (2) the 3D nuclear simulator nuclear design methods; (3) the steady state thermal hydraulic performance methods; (4) the BWR system and limiting channel dynamic analysis methods; (5) the fuel design methods; (6) the ECCS evaluation methods; and (7) the critical power margin evaluation methods. The reload safety analysis methodology center around using the above methods for analysis of: (1) fuel assembly and core design, (2) static and quasi steady-state transient events, (3) dynamic transient events, and (4) LOCA.

<u>Fuel Assembly and Core Design</u>

The reload design and safety analysis process begins with the use of the lattice physics nuclear design methods to develop the two-dimensional nuclear libraries which are required as input to the three-dimensional BWR simulator. The reload design and safety analysis process is based on a reference fuel cycle and fuel design, which satisfies the plant licensee's energy utilization plan. The fuel design inputs to the reload fuel design and safety analysis process are developed using the fuel design methods consistent with the fuel performance parameter requirements. To perform the required analyses, the lattice physics nuclear design methods require fuel assembly design information and cross section library data. The lattice physics methods also provide the local peaking patterns used in the critical power margin evaluation and used as input to the ECCS evaluation.

The 3D nuclear simulator is used to define the core state and 3D nuclear parameters used as input to the BWR system dynamic analysis methods. In addition to the inputs from the lattice physics methods, the 3D nuclear simulator requires the reference reload core design, the core operating domain, and the steady-state thermal-hydraulic parameters. It should be noted that the 3D nuclear simulator is used as a part of the nuclear design process to develop the reference core loading pattern and demonstrate that the nuclear design requirements (e.g., shutdown margin) are satisfied.

The required thermal hydraulic parameters are developed using the steady-state thermal-hydraulic performance methods and are derived from fuel assembly specific pressure drop data as a function of power and flow, based on the number and type of fuel assemblies to be used in the reference fuel cycles. Other inputs to the steady-state thermal-hydraulic performance methods include the radial power distribution and the axial power shape. With the critical power margin correlation as input, the 3D nuclear simulator is used to predict the anticipated MCPR throughout the operating cycle.

Static and Quasi Steady-State Transient Events

The 3D nuclear simulator is used in the analysis of static and quasi steady-state transients. The 3D nuclear simulator is used in the analysis of the misplaced fuel assembly error and quasi steady-state transients to determine the change in critical power ratio (ΔCPR) for these events. The misplaced fuel assembly error and transient analyses with the 3D nuclear simulator also determine the change in LHGR for these events. In addition, the 3D nuclear simulator is used to demonstrate conformance to the event acceptance limits associated with the standby liquid control system capability analysis.

Dynamic Transient Events

The dynamic analysis methods are used to determine the peak transient pressure, the transient change in power, the transient heat flux, and thermal-hydraulic parameter changes required in the limiting channel analysis to determine the ΔCPR for highly dynamic events. As part of the reload safety analysis methodology, the 3D nuclear parameters may be collapsed to one dimension or point kinetics for use by the dynamic analysis methods. The dynamic analysis methods require the plant configuration and system performance parameters as inputs through the dynamic analysis base plant model. The dynamic analysis models provide many key functions in the reload safety analysis process. As described below, they are inherent in the process for establishing the operating limit MCPR and operating limit LHGR that are used as core operating limits. In addition, the transient peak pressure determined from the overpressure protection analysis is used to demonstrate conformance to the reactor pressure vessel limit, which is based on the reactor pressure vessel design pressure. Also, the dynamic analysis methods are used to perform the stability and control rod drop accident analyses to demonstrate conformance to the appropriate event acceptance limits.

The limiting channel dynamic analysis is performed to determine the ΔCPR in the limiting channels of each type in the core. The limiting channel analysis is based on the transient parameter changes provided by the dynamic analysis models. The limiting channel dynamic analysis requires the critical power margin evaluation and the assembly design description for each fuel type as input, in addition to the transient parameter changes during the event.

LOCA

The results of the loss of coolant accident analysis are required as a part of the process to establish the adequacy of the operating limit LHGR. The results of the loss of coolant accident analysis are required to demonstrate compliance to the ECCS acceptance limits. The loss of coolant accident analysis is performed using the ECCS evaluation methods. The ECCS evaluation methods require plant configuration and system performance parameters, fuel design parameters, local peaking patterns, and the critical power margin correlation as input. The loss of coolant accident analysis is performed based on an assumed operating limit LHGR for each fuel type in the core. By demonstrating compliance to the ECCS acceptance limits, the operating limit LHGR is validated from the perspective of the loss of coolant accident analysis requirements.

### 6.6.2 Operating Limits

The MCPR calculated during the transient is compared to the safety limit. The MCPR safety limit is established using the critical power evaluation methods and includes consideration of the operating domain and manufacturing uncertainties, and a conservative core power distribution as inputs. The operating limit MCPR is established such that the transient ΔCPR will not exceed the safety limit MCPR. In establishing the operating limit MCPR, the ΔCPR for the dynamic anticipated operational occurrences, quasi steady-state anticipated operational occurrences, and the fuel loading errors are included in the evaluation. Thus, the operating limit MCPR is specified to maintain an adequate margin to boiling transition, considering all of the events in the safety analysis process that are required to demonstrate compliance to the SAFDLs.

To establish the LHGR and MAPLHGR operating limits, both anticipated operational occurrences and the loss of coolant accident analysis are considered. The results of the evaluation of anticipated operational occurrences are used to demonstrate conformance to the thermal-mechanical performance limits, and the results of the evaluation of the loss of coolant accident are used to demonstrate conformance to the ECCS acceptance limits. The initial or operating limit LHGR assumed in these analyses is validated through these analyses as being acceptable by demonstrating compliance to the applicable limits.

[ Proprietary Information Deleted ]

### 6.6.3 Input Data

There are two basic types of inputs required for the ABB reload safety analysis process: (1) plant configuration and system performance inputs and (2) fuel and core design inputs. The plant configuration and system performance inputs are used to define as-built plant design and operational requirements. The plant configuration and system performance inputs include: (1) the energy utilization plan for the operating cycle; (2) the end of current cycle projections; (3) the reference fuel cycle; (4) the allowable operating domain; (5) the desired allowances for equipment out-of-service; (6) margin improvement options; (7) instrument setpoints; and (8) equipment performance characteristics, such as system flow rates, control rod scram times, valve opening and closing times, instrument response times, control system characteristics, etc. The fuel and core design inputs are used to define the plant change due to the reload. The fuel and core design inputs include: (1) the reference reload core design; (2) fuel thermal-mechanical design parameters and limits; (3) the fuel nuclear design parameters; and (4) the fuel thermal-hydraulic performance characteristics.

The plant configuration and system performance inputs to the plant safety analysis are developed by the plant licensee and provided to ABB based on instructions from ABB. Once they are received by ABB, they are maintained in accordance with applicable parts of the ABB Quality Assurance Program. The key analysis inputs for reload safety analysis are identified on controlled documents. These documents are used by ABB in performing the safety analysis to demonstrate the ABB reload application or modified plant operational strategies is acceptable.

The fuel and design inputs to the reload safety analysis process for the ABB reload application are developed by ABB using approved fuel design methods. The fuel design inputs are developed consistent with the input requirements for the particular analysis being performed and are based on the operating cycle requirements established by the plant licensee. Fuel design inputs for fuel designed by other vendors is provided to ABB by the plant licensee. The fuel design information for fuel provided by other vendors is treated in the same manner as the plant configuration and system performance inputs.

### 6.6.4 Reload Safety Evaluation Confirmation

The reload safety analysis is performed based on an assumed Reference Core design (discussed in Section 4.3.1). The actual reload core configuration and initial conditions generally deviates slightly from the Reference Core used in the Reload Safety Analysis. A verification is performed of the as-loaded reload core to confirm that the reload safety evaluation are still applicable. Guidelines are established for each reload analysis, to determine when a re-examination and potentially re-analysis of the event is required.

Any deviation from the Reference Core outside the guidelines is explicitly treated by repeating affected parts of the Reload Safety Analysis calculations to confirm that the conclusions based on the Reference Core are valid or require modification.

## TABLE 6-1

## GENERIC BWR SAFETY ANALYSIS EVENTS

<u>Anticipated Operational Occurrences</u>

Increase in Reactor Vessel Pressure
- Pressure Regulator Failure - Closed
- Generator Load Rejection with Bypass
- Generator Load Rejection without Bypass
- Turbine Trip with Bypass
- Turbine Trip without Bypass
- Closure of One MSIV
- Closure of All MSIVs
- Loss of Condenser Vacuum

Decrease in Reactor Core Coolant Temperature
- Loss of Feedwater Heating
- Inadvertent RHR Shutdown Cooling Operation
- Inadvertent HPCI Start

Reactor Core Positive Reactivity Insertion
- Control Rod Withdrawal Error (All Power Levels)
- Control Rod Misoperation
- Incorrect Fuel Assembly Insertion

Decrease in Reactor Vessel Coolant Inventory
- Inadvertent Safety Relief Valve Opening
- Pressure Regulator Failure - Open
- Loss of AC Power
- Loss of Feedwater Flow

Decrease in Reactor Core Coolant Flow
- Trip of One Recirculation Pump
- Trip of Two Recirculation Pumps
- Recirculation Flow Control Failure - Decreasing Flow

Increase in Reactor Core Coolant Flow
- Recirculation Flow Controller Failure - Increasing Flow
- Startup of an Idle Recirculation Loop

Increase in Reactor Core Coolant Temperature
- Failure of RHR Shutdown Cooling

Increase in Reactor Vessel Coolant Inventory
- Feedwater Controller Failure - Maximum Demand

**TABLE 6-1 (CONTINUED)**

**GENERIC BWR SAFETY ANALYSIS EVENTS**

Accidents

Pipe Breaks Outside of Primary Containment

Loss of Coolant Accident

Control Rod Drop Accident

Fuel Handling Accident

Fuel Loading Error

Recirculation Pump Failure Accident

Instrument Line Break

Special Events

Core Thermal-Hydraulic Stability

Reactor Overpressure Protection

Shutdown Without Control Rods

Anticipated Transients without Scram

**TABLE 6-2**

**POTENTIALLY LIMITING EVENTS EVALUATED IN RELOAD SAFETY ANALYSIS**

## Anticipated Operational Occurrences

*Generic Analyses*

Turbine Trip or Generator Load Rejection without Bypass

Pressure Regulator Failure - Closed (BWR/6 Only)

Loss of Feedwater Heating

Control Rod Withdrawal Error

Recirculation Flow Controller Failure - Increasing Flow

Feedwater Controller Failure - Maximum Demand

*Plant Specific Analyses*

## Design Base Accidents

*Generic Analyses*

Loss of Coolant Accident

Control Rod Drop Accident

Fuel Handling Accident

Fuel Loading Error

*Plant Specific Analyses*

## Special Events

*Generic Analyses*

Core Thermal-Hydraulic Stability

Reactor Overpressure Protection

Shutdown Without Control Rods (Standby Liquid Control System Capability)

*Plant Specific Analyses*

## TABLE 6-3

## DESIGN BASES EVENT ACCEPTANCE LIMITS

Anticipated Operational Occurrences

- Radioactive Effluents ≤ 10CFR20 Limits
- Specified Acceptable Fuel Design Limits Satisfied
  - MCPR ≥ MCPR Safety Limit (Core Design Dependent)
  - LHGR ≤ Overpower Limit (Fuel Design Dependent)
  - Average Fuel Pellet Enthalpy ≤ 170 cal/gm
- Peak Reactor Vessel Pressure ≤ 110% Vessel Design Pressure
- Suppression Pool ≤ Heat Capacity Temperature Limit

Accidents

- Pipe Breaks Outside of Primary Containment
  - Offsite Dose ≤ Guideline Values of 10CFR100
  - Operator Dose ≤ GDC-19 Limits
- Loss Coolant Accident
  - Dose ≤ Guideline Values of 10CFR100
  - 10CFR50.46 Limits Satisfied
    - Peak Clad Temperature ≤ 2200 °F
    - Max. Clad Oxidation ≤ 0.17 times Clad Thickness
    - Core Wide Metal Water Reaction ≤ 0.01
    - Maintenance of a Coolable Geometry
    - Demonstration of Long Term Cooling Capability
  - Containment Pressure ≤ Containment Design Limit
- Control Rod Drop Accident
  - Offsite Dose ≤ Guideline Values of 10CFR100
  - Operator Dose ≤ GDC-19 Limits
  - Peak Fuel Enthalpy ≤ 280 cal/gm
- Fuel Handling Accident
  - Offsite Dose ≤ Guideline Values of 10CFR100
  - Operator Dose ≤ GDC-19 Limits
- Fuel Loading Error
  - MCPR ≥ MCPR Safety Limit
- Recirculation Pump Failure Accident
  - Offsite Dose ≤ Guideline Values of 10CFR100
  - Operator Dose ≤ GDC-19 Limits
- Instrument Line Break
  - Offsite Dose ≤ Guideline Values of 10CFR100
  - Operator Dose ≤ GDC-19 Limits

## TABLE 6-3 (CONTINUED)

## DESIGN BASES EVENT ACCEPTANCE LIMITS

Special Events

Core Thermal-Hydraulic Stability

Specified Acceptable Fuel Design Limits Satisfied
MCPR ≥ MCPR Safety Limit
LHGR ≤ Overpower Limit
Average Fuel Pellet Enthalpy ≤ 170 cal/gm

Shutdown without Control Rods

$k_{eff} < 1.0$

Overpressure Protection

Peak Reactor Vessel Pressure ≤ 110% Vessel Design Pressure

ATWS

Peak Reactor Vessel Pressure ≤ 120% Vessel Design Pressure
Containment Pressure ≤ Containment Design Limit
Peak Clad Temperature ≤ 2200 °F
Dose ≤ Guideline Values of 10CFR100
Demonstrated Equipment Availability

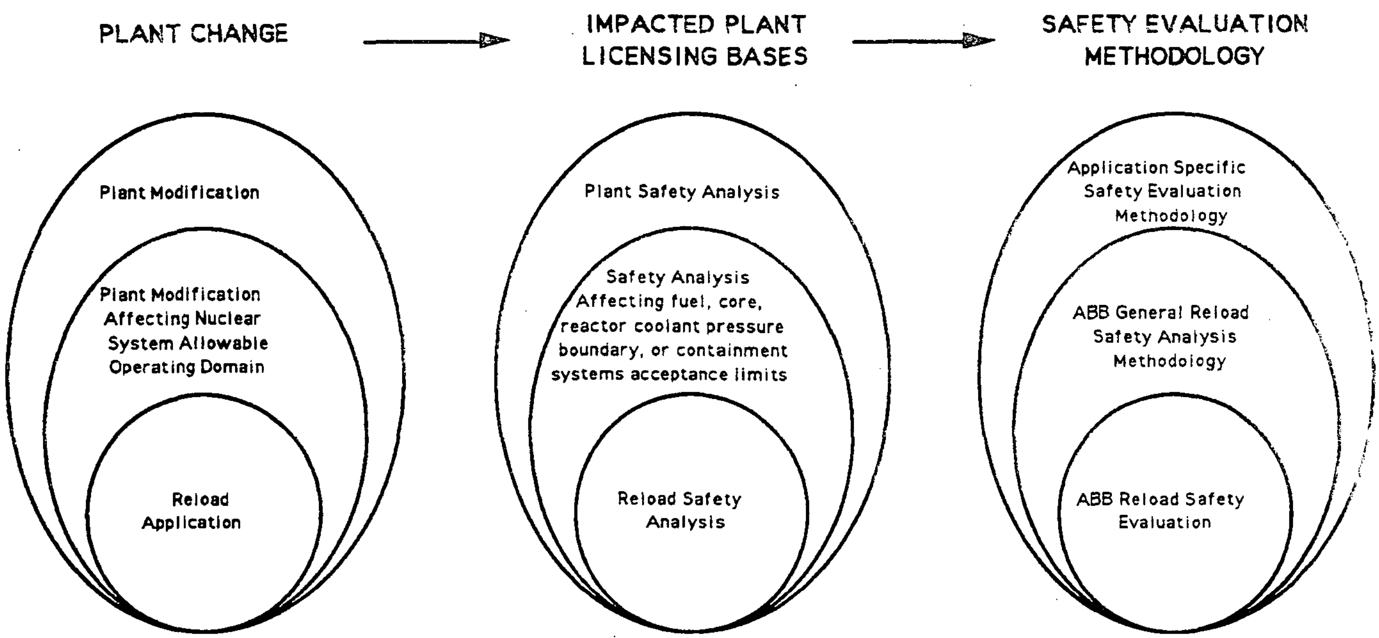



Figure 6-1 Safety Evaluations Process for Plant Modifications

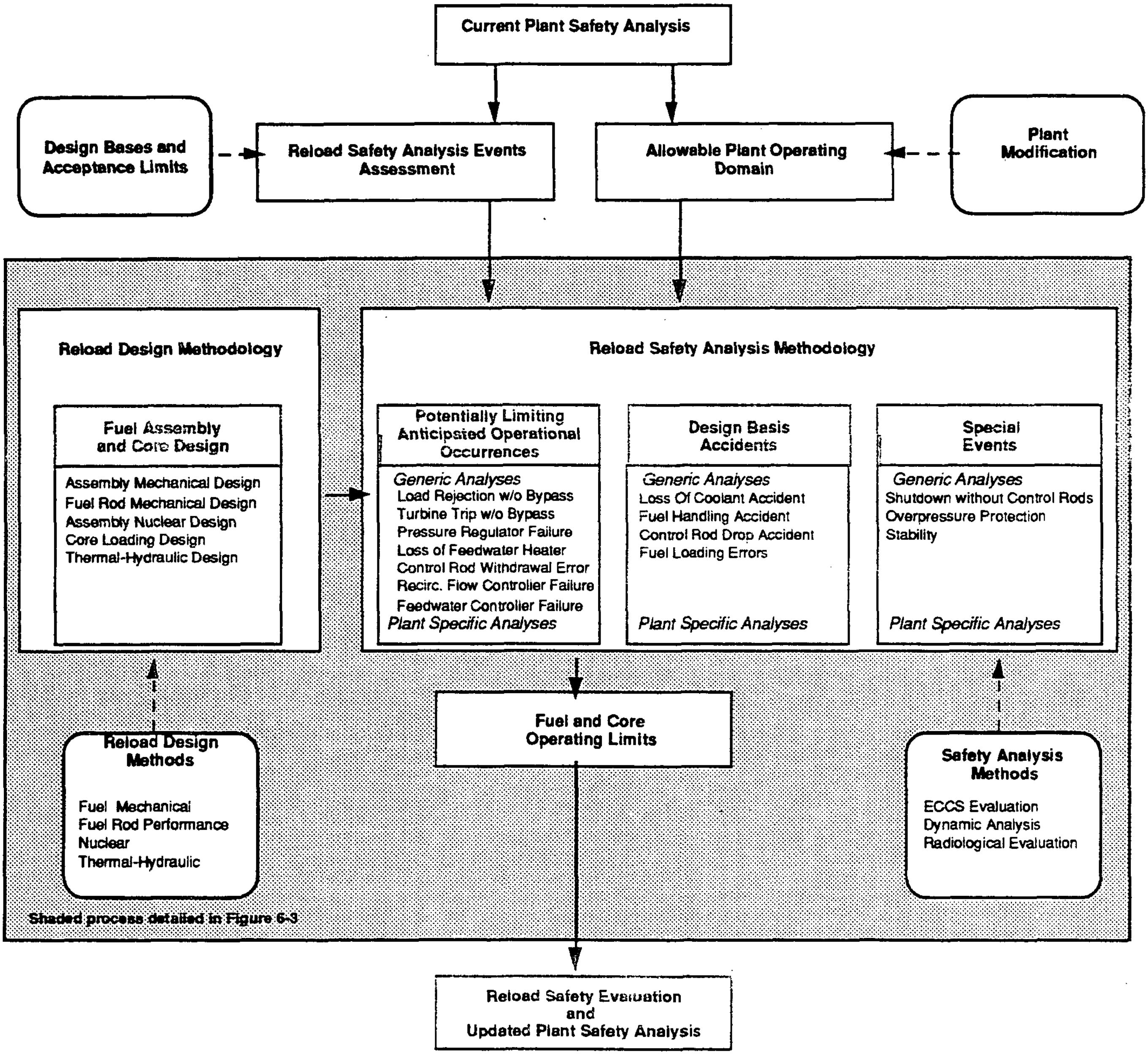



Figure 6-2 Overall Reload Safety Analysis Process

**FIGURE 6-3**

Proprietary Information Deleted

# 7 ANTICIPATED OPERATIONAL OCCURRENCES (AOO)

Anticipated Operational Occurrences (AOOs) are those conditions of normal operation which are expected to occur one or more times during the life of the plant. In the ABB reload safety analysis process the potentially limiting anticipated operational occurrences are systematically determined and evaluated using the ABB reload safety analysis methodology. The evaluation determines the plant operating limits within the allowable operating domain for the specific reload application. The ABB safety analysis methodology for evaluating the potentially limiting AOOs is described in this section.

## 7.1 Summary and Conclusions

*Summary*

This section describes, for an ABB reload application, the process of establishing the plant operating limits defined by the safety analysis of the limiting anticipated operational occurrences.

The reload safety analysis of the AOOs establishes Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) operating limits. The operating limits set by AOO events are determined by evaluating all potentially limiting AOO events. A bounding group of AOO events and state points are identified for the entire plant allowable operating domain. These bounding events are evaluated using fast or slow transient analysis methodology based on the characteristics of the event and analysis methods used. For the bounding events, the analysis uncertainty is determined either by confirming that the analysis assumptions bound acceptable probability levels or by quantifying the analysis uncertainty required to meet acceptable probability levels. Finally, the operating limits throughout the plant allowable operating domain based on AOO events are defined.

The ABB reload safety analysis process generic event assessment established the following potentially limiting AOO events:

- Generator Load Rejection Without Bypass
- Turbine Trip without Bypass
- Feedwater Controller Failure – Maximum Demand
- Pressure Regulator Failure – Closed (BWR/6 only)
- Recirculation Flow Controller Failure - Increasing Flow
- Rod Withdrawal Error

- Loss of Feedwater Heating

The safety analysis methodologies for these specific AOO events are described in this section. The general approach, as illustrated for these events, can also be used in evaluating other fast and slow transients which may result from plant-specific licensing commitments.

*Conclusions*

The ABB safety analysis methodology for evaluating slow and fast AOO transients can be applied for the potentially limiting events evaluated for all BWRs and for other slow and fast AOO transients determined to be potentially limiting based on the specific plant licensing safety analysis.

The plant specific methodologies for the Generator Load Rejection and Turbine Trip Without Bypass, Feedwater Controller Failure, Pressure Regulator Failure, Recirculation Flow Controller Failure, Rod Withdrawal Error, and Loss of Feedwater Heating can be used in reload applications and plant modifications to establish fuel and core plant operating limits.

## 7.2 Design Bases and Acceptance Limits

The reload safety evaluation shall be such that the results compared against stated design bases ensure compliance to all regulatory criteria, including the General Design Criteria of 10CFR50 Appendix A, as they are applicable to fuel systems and the effect of fuel designs on reactor systems.

For anticipated operational occurrences, there are four basic event acceptance limits: (1) radioactive effluents; (2) specified acceptable fuel design limits (SAFDLs); (3) peak reactor vessel pressure; and (4) suppression pool temperature. As explained in Section 6.4.1, only the SAFDL acceptance limits require re-evaluation for a plant reload application. The SAFDL design criteria for AOOs consist of a reload design cladding integrity criterion and fuel design cladding integrity criteria.

### 7.2.1 Reload Design Cladding Integrity

*Basis*

The minimum value of the Critical Power Ratio (CPR), denoted MCPR, is established such that at least 99.9% of the fuel rods in the core would not be expected to experience boiling transition during normal operation or anticipated operational occurrences.

*Discussion*

The acceptance limit for this design criteria is that the Operating Limit MCPR (OLMCPR) be such that the safety limit MCPR (SLMCPR), will not be violated during an AOO. The SLMCPR is defined for the reload design to ensure that 99.9% of the fuel rods in the core are expected not to experience boiling transition. This requirement provides assurance that the fuel can be operated for its specified lifetime with an acceptably low probability of failure due to boiling transition. A further discussion of this design acceptance limit with regard to both reload design and safety analysis is provided in Section 5.2.1.

### 7.2.2 Fuel Design Cladding Integrity

*Basis*

The fuel centerline temperature and the cladding strain must be below fuel type specific limits to preclude fuel melting and excessive cladding strain.

*Discussion*

[ Proprietary Information Deleted ]

## 7.3 AOO Methodology

The overall ABB reload safety analysis process and methodology was described in Section 6. The process categorized events into Anticipated Operational Occurrences, Accidents, and Special Events, and determined the events requiring evaluation for each reload application or plant operational modification. The ABB reload analysis methodology identified the reload methods and analyses, the interfaces between different disciplines, and the process of determining the plant operating limits. In this section, the reload analysis methodologies for evaluation of anticipated operational occurrences and determination of associated operating limits is described further.

### 7.3.1 AOO Events and Analysis Method

*Methodology*

Table 6-2 of Section 6 listed the potentially limiting AOO events evaluated in the ABB reload safety analysis methodology as determined on a generic basis. These AOO events, grouped by analysis methods, are:

*Fast Transients*

- Generator Load Rejection Without Bypass

- Turbine Trip Without Bypass
- Feedwater Controller Failure – Maximum Demand
- Pressure Regulator Failure – Closed (BWR/6 only)

*Slow Transients*

- Recirculation Flow Controller Failure - Increasing Flow
- Rod Withdrawal Error
- Loss of Feedwater Heating

These events are grouped into fast and slow transients based on the dynamic characteristics of the transient. "Fast transients" are those events of relatively short duration such that the impact of the spatial and temporal dynamics on the system nuclear and thermal-hydraulics is important to the overall plant response. "Slow transients" are defined as those transients for which the dynamic changes during the transient are sufficiently slow that the assumption that steady state conditions are achieved at each time step is either realistic or conservative. The fast and slow transient analysis methodologies are described in Section 7.4 and 7.5, respectively, for the AOO events listed above.

Other potentially limiting AOO events may be included in a specific plant safety analysis as a result of specific plant licensing commitments. These plant-specific AOO events, if present, are confirmed potentially limiting for a reload application, and then added, if appropriate, to the above list of generic events. Analysis of other, plant-specific AOO events uses the same general approach illustrated in detail for the generic AOO events.

### 7.3.2 Limiting Plant States and Events

Each potentially limiting AOO event is evaluated for the limiting plant condition(s) throughout the plant allowable operating domain. A single operating state or operating boundary (i.e., maximum flow, end-of-cycle exposure, maximum power) may conservatively, but not restrictively, bound all other possible states. The event analysis is performed for these limiting plant operating states.

The event analysis is performed for the Reference Core reload design to determine the event specific operating limits. The events that establish the plant operating limits throughout the plant allowable operating domain are identified. These are the limiting AOO events of the reload safety evaluation.

### 7.3.3 Analyses Calculational Uncertainty

For the limiting AOO events, an assessment of the transient analysis uncertainty is performed to confirm that there is an acceptably high probability that the predicted event consequences will not occur. All potentially limiting AOO events are analyzed with conservative assumptions covering uncertainties in the analysis code, plant model inputs, and plant operating state inputs. [ Proprietary Information Deleted ]

#### 7.3.3.1 Treatment of Analysis Uncertainty

In the safety analysis used to establish a plant operating limit, it is desired that there is a high probability with a high level of confidence that the underlying design bases will not be violated. [ Proprietary Information Deleted ]

Approach A

*Methodology*

[ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

Approach B

*Methodology*

[ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

Approach C

*Methodology*

[ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

Approach D

*Methodology*

[ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

#### 7.3.3.2 Slow Transient Analysis Uncertainty

For the MCPR operating limit there are two components to the evaluation uncertainty. There is a steady state uncertainty associated with the prediction of the number of rods in the reload core which may reach boiling transition under certain steady state conditions, in the unlikely event that such conditions are reached. This uncertainty is reflected in the safety limit MCPR (SLMCPR).

[ Proprietary Information Deleted ]

#### 7.3.3.3 Fast Pressurization Transient Analysis Uncertainty

The generic list of potentially limiting fast transient events are all fast pressurization events, hence a MCPR uncertainty associated with fast transient analysis is an uncertainty of fast pressurization events analysis. [ Proprietary Information Deleted ]

### 7.3.4 Fuel and Core Operating Limits

Fuel and core operating limits are established for the plant reload application to maintain compliance with the plant safety analysis throughout the allowable plant operating domain. Operating limits generally established by the Anticipated Operational Occurrences are the operating limit MCPR and LHGR. The MCPR and LHGR limits ensure that the AOO core and fuel design bases and acceptance limits are met.

MCPR and LHGR operating limits are established for part or all of the plant allowable operating domain. The operating limits are dependent on the specific plant allowable operating domain and flexibility options (see Appendix C). Typical plant parameters and associated flexibility options that are used to partition the allowable operating domain into a range of differing operating limits are listed in Table 7-2.

#### 7.3.4.1 MCPR Operating Limit

*Methodology*

For each potentially limiting AOO event an MCPR operating limit is evaluated for full power operation by safety analyses bounding full power operation. Full power operation is plant operation at rated power throughout the range of allowable core flows and cycle burnups. The plant Operating Limit MCPR for full power operation is the limiting value of all AOO events and the misplaced assembly accident (see Section 8.5). [ Proprietary Information Deleted ]

In some instances it is desirable, from the standpoint of effective plant operation, to partition the Operating Limit MCPR into limits applicable to exposure periods in the reload cycle or to a functional mode or operating range of plant equipment (i.e., list in Table 7-2). [ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

#### 7.3.4.2 LHGR Operating Limit

*Methodology*

The plant Linear Heat Generating Rate (LHGR) operating limit is specified for each fuel type present in a given reload cycle. The plant LHGR operating limit is the most restrictive of:

[ Proprietary Information Deleted ]

*Discussion*

[ Proprietary Information Deleted ]

### 7.4 Fast Transient Methodology

The generic, fast transient analysis methodology are described here for:

- Generator Load Rejection Without Bypass
- Turbine Trip Without Bypass
- Feedwater Controller Failure – Maximum Demand
- Pressure Regulator Failure – Closed (BWR/6 only)

### 7.4.1 Analysis Codes

*Methodology*

A transient analysis code system approved by the NRC is utilized for fast transient analyses. The transient analysis code system is used to analyze the fast transient events determined in Section 7.3.1.

*Discussion*

The current ABB fast transient analysis code used for this purpose is discussed in Appendix A. The code description and verification are described in References 23 and 39. The code system described in References 23 and 39 is referred to in this section as the "dynamic analysis model". This section provides the methodology with which the transient analysis code system is applied to fast transient analysis.

### 7.4.2 Analysis Calculational Procedure

The procedure for evaluating the ΔCPR during a fast transient involves at least two transient calculations. The first step addresses the transient response of the plant. The core is modeled as a single assembly representing a core-wide average. The plant model includes the core, the primary loop, the steam lines, and the recirculation pumps including the jet pumps.

The second step is a single channel calculation referred to as the "slave" or the "hot channel" calculation in which the impact of the transient on thermal limits is evaluated. It is assumed that the single channel does not influence the thermal-hydraulic or neutronic core response. This methodology is generally utilized in the U.S. and Europe for BWR fast transient applications.

#### 7.4.2.1 Fast Transient Code Models

Since the fast transients are analyzed using a one-dimensional dynamic analysis model, nuclear and thermal-hydraulic data simulating the three-dimensional situation are required. The ABB licensing methodology utilizes core radial averaging procedures to collapse three-dimensional geometric, thermal-hydraulic and neutronic data generated with the approved nuclear and thermal-hydraulic design codes to obtain appropriate one-dimensional data. The three-dimensional nodal simulator is also used to establish cross section dependence on coolant density, fuel temperature, and control rod fraction for the preparation of appropriate polynomial forms for the one-dimensional model. The methodology used in the collapsing process is general in that it is not limited to ABB fuel types, but is applicable to non-ABB fuel as well.

In addition, effective fuel rod performance information and thermal-hydraulic data for the dynamic analysis model are obtained from the same fuel rod performance and thermal-hydraulic codes used for the fuel rod performance and thermal-hydraulic evaluations discussed in Sections 3 and 5, respectively.

Specific fuel rod, nuclear, and thermal-hydraulic input to the transient analyses are described in Sections 3, 4, and 5, respectively. The application of these data to establishing the dynamic analysis model is summarized in Figure 7-1.

[ Proprietary Information Deleted ]

As discussed in Section 4, delayed neutron fractions, inverse neutron velocities, and energy deposition factors are also calculated with the nuclear design codes for application in the fast transient analyses.

The radial heat conduction model in the fuel rod determines the fuel rod time constant which influences the peak and integrated core power and maximum heat flux in fast pressurization events. [ Proprietary Information Deleted ]

The one dimensional dynamic analysis core hydraulic model is derived from the three dimensional model used for the nuclear and thermal-hydraulic evaluations described in Sections 4 and 5. [ Proprietary Information Deleted ]

#### 7.4.2.2 Plant Response Calculations

The plant response to the AOO is simulated with the one-dimensional dynamic analysis model at each state point requiring evaluation. The initial conditions for the transient analysis are established from the Reference Core three-dimensional core simulator calculations performed for the nuclear design evaluation. For each fast transient a sufficient range of plant state points are evaluated to establish the minimum margin to thermal limits, using the methodology discussed in Section 7.3.2.

The plant nuclear and thermal-hydraulic initial operating conditions including the plant heat balance, core power distribution and fuel rod characteristics, are analyzed with conservative assumptions. The plant transient event boundary conditions including reactor protection actions and control system functions are set at the bounding conditions for plant operating. The treatment of all uncertainties in the fast transient evaluation is discussed in detail in Section 7.3.3.3.

The plant response calculation provides inlet and outlet boundary conditions, and time-dependent power and axial power shape conditions for the hot channel calculations.

#### 7.4.2.3 Hot Channel Calculations

The hot channel calculations are utilized to evaluate the impact on thermal limits during the transient. Specifically, the change in CPR is evaluated for the specific fuel assembly type determined to be the hot assembly.

[ Proprietary Information Deleted ]

### 7.4.3 Generator Load Rejection Without Bypass

#### 7.4.3.1 Event Description

The generator load rejection without bypass event is the postulated complete loss of electrical load to the turbine generator coupled with the assumed failure of the turbine bypass system. Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close rapidly to prevent overspeed of the turbine generator rotor due to the loss of load.

The rapid closure of the turbine control valves causes a sudden reduction of steam flow which results in a nuclear system pressure increase. Neutron flux increases rapidly because of the core void reduction caused by the pressure increase. Turbine control valve fast closure initiates the reactor scram trip signal and the prompt Recirculation Pump Trip (RPT) for the BWR designs that have this feature, which results in a rapid reactor shutdown. The reactor pressure vessel pressure increase is limited by the action of relief valves. The neutron flux increase is limited by the scram and the prompt RPT (on applicable plants) consistent with the inherent effects of void and Doppler reactivity feedback. The peak fuel surface heat flux increases initially due to the neutron flux increase then decreases following reactor shutdown. Long term reactor water makeup is provided by the feedwater system or high pressure makeup systems. Heat rejection is through the relief valves to the suppression pool.

No restart is assumed and the reactor is to be cooled down. The operator is expected to take the following actions as appropriate, consistent with the normal and emergency plant operating procedures:

- Control the reactor pressure.
- Ascertain that the control rods are in.

- Monitor and maintain reactor water level.
- Cool down the reactor consistent with plant procedures.

### 7.4.3.2 Analysis Methodology

The following plant operational conditions and assumptions form the principal bases for the analysis of the generator load rejection without bypass event:

[ Proprietary Information Deleted ]

## 7.4.4 Turbine Trip without Bypass

### 7.4.4.1 Event Description

A variety of turbine or nuclear system malfunctions can initiate a turbine trip which is the rapid closure of all turbine stop valves (TSV). The rapid closure of the turbine stop valves causes a sudden reduction of steam flow which results in a nuclear system pressure increase. Neutron flux increases rapidly because of the core void reduction caused by the pressure increase. Turbine stop valve closure initiates the reactor scram trip signal and the prompt Recirculation Pump Trip (RPT) for the BWR designs that have this feature, which results in a rapid reactor shutdown. The reactor pressure vessel pressure increase is limited by the action of relief valves. The neutron flux increase is limited by the scram and the prompt RPT (on applicable plants) consistent with the inherent effects of void and Doppler reactivity feedback. The peak fuel surface heat flux increases initially due to the neutron flux increase then decreases following reactor shutdown. Long term reactor water makeup is provided by the feedwater system or high pressure makeup systems. Heat rejection is through the relief valves to the suppression pool.

No restart is assumed and the reactor is to be cooled down. The operator is expected to take the following actions as appropriate, consistent with the normal and emergency plant operating procedures:

- Control the reactor pressure.
- Ascertain that the control rods are in.
- Monitor and maintain reactor water level.
- Cool down the reactor consistent with plant procedures.

The turbine trip without bypass is similar to the generator load rejection and frequently bounded by that event. Differences between the two events are due the initialization of the transients and the increase in recirculation system pump speed due to the loss of electrical load for the load rejection case. The turbine trip event starts with the closure of the initially fully opened turbine stop valves, whereas the load rejection starts with closure of the partly closed turbine control valves. The turbine stop valves have a slightly faster stroke time, which tends to compensate for differences in initial opening positions. The load rejection without bypass also has slightly longer signal delays from the start of the event until scram initiation.

#### 7.4.4.2 Analysis Methodology

The following plant operational conditions and assumptions form the principal bases for the analysis of the turbine trip without bypass event:

[ Proprietary Information Deleted ]

### 7.4.5 Feedwater Controller Failure – Maximum Demand

#### 7.4.5.1 Event Description

The feedwater controller failure (FWCF) - maximum demand is a postulated single failure of a control device which causes an increase in coolant inventory and core inlet subcooling by increasing the feedwater flow. It is assumed that the feedwater controller is forced to its upper limit resulting in a maximum flow demand.

The increase in feedwater flow causes a slow increase in power because of the increased saturation temperature due to the water level increase and the increased core inlet subcooling. The increased power results in a somewhat higher steam flow. The pressure regulator is assumed to be operating to initially control reactor pressure. During this portion of the transient, the water level continues to increase. Generally, turbine and feedwater system trips are initiated on high water level due to the water level increase, unless the high power reactor trip setpoint is first reached.

The turbine trip initiates the rapid closure of all turbine stop valves. The rapid closure of the turbine stop valves causes a sudden reduction of steam flow which results in a nuclear system pressure increase. Neutron flux increases rapidly because of the core void reduction caused by the pressure increase. Turbine stop valve closure or high water level (for BWR/6 plants) initiates the reactor scram trip signal and the prompt Recirculation Pump Trip (RPT) for the BWR designs that have this feature, which results in a rapid reactor shutdown. The reactor vessel pressure increase is limited by the action of the turbine bypass valves and the relief valves. The neutron flux increase is limited by the scram and the prompt RPT (on applicable plants) consistent with the inherent effects of void and Doppler reactivity feedback. The peak fuel surface heat flux increases initially due to the neutron flux increase then decreases following reactor shutdown. Long term reactor water makeup is provided by the high pressure makeup systems. Long term heat rejection is through the turbine bypass valves.

No restart is assumed and the reactor is to be cooled down. The operator is expected to take the following actions as appropriate, consistent with the normal and emergency plant operating procedures:

- Observe that the high water level feedwater pump trip and a scram have terminated the event.
- Switch the feedwater controller from automatic to manual control in order to try to regain a correct output signal.
- Ascertain that the control rods are in.
- Monitor and maintain reactor water level.
- Cool down the reactor consistent with plant procedures.

[ Proprietary Information Deleted ]

### 7.4.5.2 Analysis Methodology

The following plant operational conditions and assumptions form the principal bases for the analysis of the feedwater controller failure - maximum demand event:

[ Proprietary Information Deleted ]

## 7.4.6 Pressure Regulator Failure – Closed (BWR/6 only)

### 7.4.6.1 Event Description

For BWR/6 plants, a potentially limiting event is the postulated failure of the pressure regulator to a zero steam flow demand. Should this occur, it would cause closure of the turbine control valves at their normal closing speed as well as inhibit the turbine bypass valve from opening. Closure of the turbine control valves causes a reduction of steam flow which results in a nuclear system pressure and power increase. Reactor scram is initiated on high neutron flux. The relief valves open to relieve excess steam and limit the nuclear system pressure. In some cases, RPT may be initiated on high reactor pressure. The peak fuel surface heat flux increases initially due to the neutron flux increase, then decreases following reactor shutdown. Long term reactor water makeup is provided by the feedwater system or high pressure makeup systems. Heat rejection is through the relief valves to the suppression pool.

No restart is assumed and the reactor is to be cooled down. The operator is expected to take the following actions as appropriate, consistent with the normal and emergency plant operating procedures:

- Control the reactor pressure.
- Ascertain that the control rods are in.
- Monitor and maintain reactor water level.
- Cool down the reactor consistent with plant procedures.

### 7.4.6.2 Analysis Methodology

The following plant operational conditions and assumptions form the principal bases for the analysis of the pressure regulator failure - closed event:

[ Proprietary Information Deleted ]

## 7.5 Slow Transient Methodology

"Slow transients" are defined as those transients for which the power increase during the transient is sufficiently slow that the assumption that steady-state conditions are achieved at each time step is either realistic or conservative. These transients are sufficiently slow that the impact of kinetic phenomena such as delayed neutron effects are negligible.

The following AOOs are classified as slow transients:

- Recirculation Flow Controller Failure - Increasing Flow
- Rod Withdrawal Error
- Loss of Feedwater Heating

### 7.5.1 Analysis Codes

A nuclear design code system accepted by the NRC is utilized for the analyses of the slow transients (See Appendix A for a list of approved codes.) The two-dimensional lattice physics code is used to calculate the nuclear data (e.g. cross sections, local peaking factors, MCPR subchannel factors, detector constants, etc.) required for the three-dimensional nodal core simulator input. The use of the three-dimensional core simulator for these transients provides specific representation of the axial and radial power distribution changes during the transient.

### 7.5.2 Analysis Calculational Procedure

[ Proprietary Information Deleted ]

### 7.5.3 Recirculation Flow Controller Failure - Increasing Flow

#### 7.5.3.1 Event Description

The recirculation loop flow controller failure is assumed to fail in a manner which results in an increase in recirculation loop flow. Increasing recirculation flow results in an increase in core flow. The increase in core flow causes an increase in core power level as well as a shift of the power toward the top of the core by reducing the void fraction in the top of the core.

The rate and magnitude of the power increase are dependent on the rate and magnitude of the flow increase. If the flow increase is at a relatively slow rate or a relatively small increase, the operator would be expected to control the power increase through normal operational procedures. Conversely, if the flow increase is relatively rapid or of sufficient magnitude, the neutron flux could exceed the high flux scram set point and a scram would be initiated. [ Proprietary Information Deleted ]

A representative sequence of events for this transient are:

(1) The Recirculation Flow Controller fails, increasing flow demand,

(2) Gradual recirculation loop flow increases and subsequent core flow increases,

(3) Turbine control valves and possibly bypass valves open to control reactor pressure, and

(4) Core power increases until a steady state core power level is achieved at maximum recirculation flow.

#### 7.5.3.2 Analysis Methodology

[ Proprietary Information Deleted ]

### 7.5.4 Rod Withdrawal Error

#### 7.5.4.1 Event Description

The control rod withdrawal error event (RWE) is initiated by an operator erroneously selecting and continuously withdrawing a control rod or a control rod bank at its maximum withdrawal rate. Both the core average power and local power in the vicinity of the erroneously withdrawn control rod or control rod bank increases due to the positive reactivity insertion. The core average power and the local power increase until the control rod or rod bank reaches its fully withdrawn position or the rod block monitor (RBM) for BWR/3 through BWR/5 plants, or rod withdrawal limiter (RWL) for BWR/6 plants, acts to inhibit further control rod withdrawal. The BWR/2 plants utilize a quarter core RBM. During the event, the core power increases until the control rod withdrawal is terminated. The turbine control valves will open to compensate for the increased steam flow until a new steady state condition is reached.

#### 7.5.4.2 Analysis Methodology

The differences in rod control systems for BWR/3 through BWR/5 plants and BWR/2 and BWR/6 plants require modification of the methodology for the different plant types. Therefore, the methodology is initially described for the BWR/3 through BWR/5 plants, and required modifications for BWR/2 and BWR/6 plants are subsequently described.

BWR/3-5 Plants

The number of possible control rod withdrawal error events is very large due to the number of control rods in the core and the wide range of exposures and power levels during an operating cycle. In order to encompass all of the possible control rod withdrawal errors which could credibly occur, a limiting analysis is defined such that a conservative assessment of the consequences is provided. Therefore, the postulated error is a continuous withdrawal of the control rod which is expected to cause the maximum change in CPR. Specifically, the following initial conditions are assumed:

[ Proprietary Information Deleted ]

(3) The control rod selected for withdrawal is initially fully inserted. This rod is designated as the "error rod".

(4) Candidate error rods selected from the Reference Core control rod sequence are considered. All error rods with a potential for being limiting are evaluated.

[ Proprietary Information Deleted ]

In addition, the following conservative assumptions are imposed on the licensing analysis during the transient:

[ Proprietary Information Deleted ]

(4) The operator ignores all warnings during the transient, including RBM system alarms which must be reset in order to continue rod withdrawal. Therefore, the error rod is assumed to be withdrawn until its motion is terminated by the RBM.

(5) Failures are assumed to have occurred in the local power range monitor (LPRM) strings that provide input to the RBM system (i.e., the four LPRM strings nearest to the control rod being withdrawn). The assumed failures are selected based on the plant design basis for failed LPRMs.

(6) Unless the failure mode has been explicitly eliminated for a given plant, one of the two RBM instrument channels is assumed to be bypassed and out of service. The A and C elevation LPRM chambers input to one channel while the B and D elevation LPRM chambers input to the other. The channel with the greatest response is assumed to be bypassed.

The Rod Withdrawal Error is evaluated with the three dimensional core simulator described in Appendix A. The full core is modeled to describe detector strings and error rods as accurately as possible.

[ Proprietary Information Deleted ]

BWR/6 Plants

The licensing analysis methodology for a BWR/6 plant is the same as that for BWR/2 through BWR/5 plants consistent with use of a Rod Withdrawal Limiter (RWL) system rather than an RBM system.

The BWR/6 RWL system can be summarized as follows:

(1) The RWL system allows control rod withdrawal of two notches at powers higher than 70% power and four notches at powers between 40 and 70%.

(2) Multiple control rods can be withdrawn simultaneously as groups, and

(3) The rod withdrawal error can occur from any initial position and can be more limiting when withdrawn from an intermediate position. Therefore, the limiting initial configuration can not be assumed to be the fully inserted group and all intermediate control rod positions for the error rod must be investigated.

Consequently, the same calculational model is used for the BWR/6 case as the BWR/3-5 case with the constraints for the RWL system utilized in place of the RBM system constraints and calculated responses. Furthermore, the change in thermal margin is calculated assuming that the RWE is initiated from each step allowed by the RWL rather than assuming that the transient is initiated from the completely inserted position of the error group.

BWR/2 Plants

The analysis process for the BWR/2 plants is the same as the BWR/3-5 plants except that the rod block is based on the response of the LPRMs from the quarter core configuration rather than the LPRM strings surrounding the control rod being withdrawn.

### 7.5.5 Loss of Feedwater Heating

### 7.5.5.1 Event Description

Loss of feedwater heating (LOFH) results in a core power increase and power distribution shift due to an increase in the core inlet subcooling. Examples of evolutions resulting in LOFH are as follows:

(1) A steam extraction line to a feedwater heater is closed.

(2) Feedwater flow bypasses one or more feedwater heaters.

The first case produces a gradual cooling of the feedwater. The second case causes an interruption of heating of the feedwater. In either case cooler feedwater is mixed with the recirculation flow. Since the recirculation flow rate is substantially greater than the feedwater flow rate, the rapid decrease in feedwater temperature causes a gradual increase in core inlet subcooling. The power increases at a moderate rate and the power shifts towards the bottom of the core.

If the power exceeds the normal full power flow control line, the operator would be expected to insert control rods to return the power and flow to their normal range. Without this action the neutron flux could exceed the scram set point and a scram would occur. If the scram set point is not reached, the reactor would settle at a new steady state condition until operator action is taken to bring it back into the normal operating range of the power/flow map.

In either case the power increase results in a decrease in the MCPR and in an increase in the MLHGR.

The sequence of events can be summarized as follows:

(1) The maximum feedwater temperature reduction credible for the plant is assumed to occur instantaneously.

(2) The reduced temperature feedwater starts to increase the core power level and steam flow,

(3) The turbine control valves open to control the pressure,

(4) The APRM or thermal power alarm setpoint is reached, and the operator may take action to remain within the correct operating range,

(5) If the core power does not reach the scram setpoint, a new steady state operating condition is achieved,

(6) If core power reaches the scram setpoint, the APRMs will initiate a reactor scram which terminates the power increase.

### 7.5.5.2 Analysis Methodology

The following initial core conditions are assumed:

(1)The event is initiated from the core power and flow conditions providing the greatest challenge to thermal limits. The plant licensing basis, as augmented by ABB sensitivity studies as required, are utilized to establish or confirm these conditions.

(2) A control rod pattern is established for the initial core state which simultaneously places bundles as close to MCPR and LHGR thermal limits as practical.

(3) Equilibrium xenon is established for the initial core condition.

The transient is simulated in the following manner:

[ Proprietary Information Deleted ]

The Loss of Feedwater Heating event is evaluated with the three dimensional core simulator described in Appendix A. [ Proprietary Information Deleted ]

## TABLE 7-1

## FAST PRESSURIZATION TRANSIENT IMPORTANT INPUT PARAMETERS

| PARAMETER |
|---|
| *NEUTRONIC MODEL* |
| Void feedback gain |
| Scram reactivity |
| Doppler feedback gain |
| Prompt moderator heating |
| *THERMAL-HYDRAULIC MODEL* (core average and hot channel models) |
| Core two-phase friction factor |
| Core inlet pressure drop moved to outlet |
| Active core nodes |
| Initial core bypass flow |
| Transient CPR performance |
| *RECIRCULATION SYSTEM MODEL* |
| Recirc. loop inertia |
| Jet pump fluid inertia |
| Jet pump M ratio |
| Jet pump N ratio |
| Separator outlet inertia |
| Separator inertia |
| Separator pressure drop |
| Inertia of Downcomer & Lower Plenum |
| *VESSEL and STEAMLINE MODELS* |
| Steam dome volume |
| Upper downcomer volume |
| Steamline length |
| Steamline flow area |
| Steamline inertia |
| Steamline pressure drop |
| Steamline specific heat ratio |
| Steamline nodes |
| *INITIAL OPERATING CONDITIONS* |
| Power/ heat balance |
| Control rod pattern |
| Core axial burnup distribution |
| Fuel rod gas gap heat transfer coefficient |
| *TRANSIENT CONDITIONS* |
| Control Rod Scram Speed |
| Reactor Protection System Actuations |
| Reactor Control System Actions |

## TABLE 7-2

## EXAMPLE OF OPERATING LIMIT DEPENDENCIES WITHIN PLANT ALLOWABLE OPERATING DOMAIN

| Parameter | Flexibility Options |
|---|---|
| Reactor Power | Normal Planned Operation<br>Equipment Out of Service |
| Core Flow | Normal Planned Operation<br>Extended Load Limit Line<br>Maximum Extended Operating Domain<br>Increased Core Flow<br>Equipment Out of Service |
| Core Average Burnup | Normal Planned Operation<br>Extended Cycle Operation |
| Number of Recirculation Loops in Operation | Single Loop Operation |
| Feedwater Temperature | Partial Feedwater Heating<br>Final Feedwater Temperature Reduction |
| Reactor Scram Time | Technical Specification Scram Speed<br>Plant Measured Scram Speed |
| Recirculation Pump Trip Operability | Inoperable Recirculation Pump Trip |

**FIGURE 7-1 THROUGH FIGURE 7-2**

Proprietary Information Deleted

# 8 ACCIDENT ANALYSIS

Accidents are defined as those postulated events that affect one or more of the radioactive material barriers. These events are not expected to occur during the plant lifetime, but are used to establish the design basis for certain systems. In the ABB reload fuel safety analysis process, the postulated accidents that require re-evaluation for the introduction of ABB reload fuel or changes in allowable plant operating domain have been systematically identified. It is these potentially limiting accidents that are evaluated for plant specific reloads to demonstrate that the applicable design bases are satisfied and the plant operating limits within the allowable operating domain are acceptable. The ABB safety analysis methodology for evaluating the potentially limiting accidents is described in this section.

## 8.1 Summary and Conclusions

*Summary*

This section describes, for an ABB reload application, the process for evaluating postulated accidents and confirming the adequacy or the plant operating limits defined by the plant safety analysis. Based on an assessment of the consequences of the spectrum of postulated accidents considered in plant safety analyses, there are four groups of accidents that generically require re-evaluation in the reload fuel safety analysis process. These accidents are:

- Loss of Coolant Accident
- Control Rod Drop Accident
- Fuel Handling Accident
- Misplaced Bundle Accident - Rotated or Mislocated

The specific safety analysis methodology for each of these specific types of accidents are described in this section.

*Conclusions*

Appropriate design bases and evaluation methodologies are established for the specific accidents evaluated in reload fuel safety analysis process. These evaluation methodologies can be used as part of the process to establish the acceptability of the core operating limits for ABB reload fuel.

## 8.2 Loss of Coolant Accident

The Loss of Coolant Accident (LOCA) has been selected to bound the consequence of events that release radioactivity directly to the primary containment as a result of pipe breaks inside the primary containment. The reactor coolant pressure boundary contains a number of different sizes, lengths, and locations of piping. Failure of this piping results in loss of coolant from the reactor and discharge of the coolant directly to the primary containment.

The postulated LOCA consists of a piping break in the reactor coolant pressure boundary which exceeds the capability of the reactor coolant makeup system. The pipe breaks to be considered encompass all sizes and locations up to and including a the rapid circumferential failure of the largest reactor recirculation system piping. By evaluating the entire spectrum of postulated break sizes, the most severe challenge to the emergency core cooling System (ECCS) and primary containment can be determined. The plant maximum average planar linear heat generation rate (MAPLHGR) operating limit is establish to ensure, in part, compliance with the LOCA design bases.

The LOCA analysis design bases, event description, and methodology are described here.

### 8.2.1 Design Bases

The Loss of Coolant Accident is a postulated accident, prescribed in the Code of Federal Regulations Title 10 Part 50.46 (Reference 42), to determine the design acceptance criteria for the plant Emergency Core Cooling System. Title 10CFR50.46 prescribes five specific design acceptance criterion for the plant:

(1) Peak Cladding Temperature

(2) Local Oxidation

(3) Total Hydrogen Generation

(4) Coolable Geometry

(5) Long Term Cooling

The design basis acceptance criteria are described below.

#### 8.2.1.1 Peak Cladding Temperature

*Basis*

The Code of Federal Regulations (10CFR50.46) requires that "The calculated maximum fuel rod cladding temperature shall not exceed 2200°F."

*Discussion*

The loss of coolant accident analysis is performed for each new fuel type to demonstrate compliance to the above requirement. Fuel type specific operating limits are established in the plant technical specifications to ensure that this design acceptance criteria is not violated. The plant maximum average planar linear heat generation rate (MAPLHGR) operating limit or LHGR operating limit ensures compliance with this design bases.

#### 8.2.1.2 Local Oxidation

*Basis*

The Code of Federal Regulations (10CFR50.46) requires that "The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness before oxidation."

*Discussion*

The maximum local cladding oxidation limit, along with the fuel rod clad temperature limit discussed above, together ensure that the cladding remains sufficiently intact to retain the fuel pellets within the fuel rods both during the blowdown and reflood phase of the LOCA. When these criteria are satisfied, the extent of clad swelling and rupture are limited and sufficient ductility remains to prevent fracture during reflood.

#### 8.2.1.3 Total Hydrogen Generation

*Basis*

It is required to demonstrate that "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react."

*Discussion*

Restricting the amount of hydrogen generated to below that established in this design acceptance criteria conservatively ensures that the concentration of this gas is maintain below the flammability limit. For most fuel designs, the peak cladding temperature and local maximum oxidation acceptance limits, restrict the potential total core hydrogen generation significantly below the 0.01 limit.

### 8.2.1.4 Coolable Geometry

*Basis*

It is required that the ,"Calculated changes in core geometry shall be such that the core remains amenable to cooling."

*Discussion*

In order for coolant to reach all areas of the core, the changes in core geometry due to clad swelling and rupture cannot result in blockage of flow to any portion of the core.

In their review of the acceptance criteria for ECCS (Reference 52) the United States Atomic Energy Commission, predecessor to the Nuclear Regulatory Commission, concluded that compliance with the first two design criterion, in themselves ensures compliance with this fourth design criteria. Specifically, it was concluded that maintaining the peak cladding temperature below 2200°F and maintaining less than 17 percent local cladding oxidation will ensure that sufficient ductility of the cladding remains during the quenching process. Therefore, the core fuel structure will remain intact and amenable to long-term cooling.

Hence, in the ABB reload safety analysis methodology this criterion is met by demonstrating compliance to the Peak Cladding Temperature and Local Oxidation design acceptance criteria.

### 8.2.1.5 Long Term Cooling

*Basis*

The Code of Federal Regulations (10CFR50.46) requires that "After any calculated successful operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

*Discussion*

Following quenching of the fuel cladding, it is necessary to maintain the cladding temperature sufficiently low to assure that the cladding continues to maintain its function. The criterion of maintaining the core coolable for an extended period of time following a postulated LOCA is achieved by ensuring a continuous source of ECCS water. Once the reactor vessel has been reflooded all fuel cladding temperatures would return to near saturation temperatures. The core will remain at these cool temperatures as long as a continuous supply of ECCS is available. Compliance with this criterion has been demonstrated during the original review of the plant ECCS design. Since the ECCS design and performance does not change with fuel reloads, compliance is still maintained in subsequent reload cycles. Hence this criterion is be addressed for ABB reload applications.

### 8.2.2 Event Description

The LOCA event described here is for the limiting break in a modern BWR with two external recirculation loops that drive the internal jet pumps. The limiting break is generally a large double-ended guillotine break of the recirculation line at the suction side of the recirculation pump. Other break locations and plant designs have slightly varying transient characteristics similar to that outlined here.

Following the postulated pipe rupture, rapid discharge of coolant occurs through both sides of the break, with greater flow from the vessel side. Rapid depressurization of the reactor vessel occurs after a short period of slower pressure decrease. Pump side flow is restricted by the reduced flow area of the jet pump nozzle and friction losses in the recirculation loop and pump. Loss of all AC power is assumed to occur in conjunction with the break, resulting in coastdown of the recirculation pumps. The reactor scrams on low steam dome pressure or low reactor vessel water level followed by closure of the primary containment isolation valves. Main steam isolation valves (MSIVs) close at the time of reactor scram. Following reactor shutdown the pressure begins to fall rapidly. After several seconds the liquid level in the downcomer falls to the jet pump suction elevation. Uncovery of the jet pump suction pipes has a marked effect on the discharge flow through the break. The broken loop flow decreases very rapidly, and the steam content increases, resulting in a sudden decrease in break mass flow rate.

Flashing in the jet pumps and subsequently in the lower plenum occurs as the pressure continues to decrease. This results in a short term level rise in the core and downcomer. Following this level swell, the continued inventory decrease results in falling liquid level in the downcomer which initiates high pressure emergency core cooling systems, followed soon afterwards by the low pressure core cooling systems . Core water level will drop exposing the fuel rods to a steam environment. Downflow of injected coolant from the upper plenum into the core provides convective cooling of the fuel rods. The fuel rod convective cooling and radiative heat transfer to cooler surfaces compete with the generation of decay heat. The relative rate of heat generation and removal dictates the resultant fuel cladding temperature transient. The fuel temperature transient is terminated by emergency core cooling refilling the reactor vessel and reflooding the core. The peak cladding temperature can occur during reactor blowdown, refill, or at core reflood depending on the effectiveness of fuel heat removal relative to the fuel initial stored energy and continued heat generation.

The reactor is initially shut down by the increase in void fraction due to the depressurization which is followed by the automatic insertion of the control rods. The event is terminated by the closure of the containment isolation valves, actuation of the ECCS and operation of the other required safety systems.

### 8.2.3 Analysis Methodology

A LOCA analysis evaluation is performed for each new reload fuel type introduced in a reload application. Appropriate analyses are performed to establish the core operating limits for the new fuel. If no new fuel types are introduced, an evaluation of the loss of coolant accident is not required by the ABB reload safety analysis process.

#### 8.2.3.1 ECCS Evaluation Model

*Methodology*

LOCA analysis is performed with an approved ECCS Evaluation Model including the analysis code, plant model sensitivities, and plant evaluation methodology.

*Discussion*

The approved ECCS Evaluation Model described in Appendix A.4.3 is used to perform reload safety evaluations of new fuel designs introduced in a plant specific reload application.

#### 8.2.3.2 Limiting LOCA Design Basis Event

*Methodology*

The potentially limiting design basis LOCA events for the specific plant in question are identified based on the break spectrum analysis in the plant safety analysis. The peak cladding temperature is calculated for the potentially limiting events and the design basis break for the specific plant identified.

*Discussion*

The potentially limiting design basis LOCA events are characterized by a break sizes, break locations, and worst single failures. [ Proprietary Information Deleted ]

### 8.2.3.3 Design Basis Event Analysis

*Methodology*

For the design basis break, the plant system response to the postulated LOCA event, is calculated. The limiting fuel assembly thermal-hydraulic and limiting fuel rod response are calculated based on the plant system response. For each new fuel design, the maximum average planar LHGR is determined that ensures compliance with the LOCA design acceptance criteria.

*Discussion*

The ABB ECCS Evaluation Model contains sufficient conservatism to assure that the LOCA design acceptance criteria are met with a significant safety margin. [ Proprietary Information Deleted ]

### 8.2.3.4 Total Hydrogen Generation

*Methodology*

The methodology used to conservatively calculate the total amount of hydrogen generated during a postulated LOCA consists by the following steps:

[ Proprietary Information Deleted ]

*Discussion*

In the total hydrogen generation analysis, the uncertainty in core-wide bundle power distribution will be bounded [ Proprietary Information Deleted ] As commonly acknowledged, the small number of high-power bundles contributes the largest portion of the total cladding oxidation during a LOCA. [ Proprietary Information Deleted ]

### 8.2.3.5 MAPLHGR Operating Limit

Fuel type specific operating limits are included in the plant technical specifications to ensure that ECCS design acceptance criteria are not violated. The fuel type specific operating limit established to meet ECCS LOCA requirements is the maximum average planar linear heat generation rate (MAPLHGR).

Methodology

The plant Maximum Average Planar Linear Heat Generating Rate (MAPLHGR) operating limit is specified for each fuel type present in reload cycle. The plant MAPLHGR operating limit is the most restrictive of:

(1) The MAPLHGR of established to comply with LOCA ECCS design acceptance criteria,

(2) Any other plant-specific fuel MAPLHGR operational restrictions.

Discussion

[ Proprietary Information Deleted ]

## 8.3 Control Rod Drop Accident

The Control Rod Drop Accident Methodology has been provided in detail in Reference 33. The design bases are provided in Section 3 of Reference 33. The analysis methodology, including examples of the calculation to determine the limiting control rod and the energy deposition into the fuel, is described in Section 4 of Reference 33.

## 8.4 Fuel Handling Accident

### 8.4.1 Design Bases

The amount of the radioactive material that is released to the environment as a result of the refueling accident must be less than the limits specified in 10CFR100. The onsite radiological effect of the fuel handling accident is also limited by the criteria identified in GDC 19.

### 8.4.2 Event Description

The refueling accident is postulated to provide an upper bound on the release of radioactive materials outside of the drywell. For BWR/2s through BWR/5s, the refueling accident can occur within secondary containment in the spent fuel pool or in the core if the vessel head is off for refueling. For BWR/6s, the refueling accident can occur within containment or within the auxiliary building in the spent fuel pool.

The dropping of a fuel assembly could be caused by breakage of the fuel assembly handle, the fuel grapple or the grapple cable, or improper grappling. Energy from the dropped assembly is transmitted to the impacted fuel assemblies during two or more impacts. A portion of the energy is absorbed by the dropped assembly, and a portion is absorbed by the impacted assemblies. Energy absorption by the fuel rod cladding can cause cladding failure and the release of fission products to the reactor coolant.

The dropping of a fuel assembly can result in the release of fission products directly to the atmosphere of the building in which the accident is postulated to occur. A high radiation signal in the ventilation exhaust system radiation monitors will automatically close the building isolation valves and initiate standby gas treatment.

### 8.4.3 Analysis Methodology

Based on the design of ABB reload fuel assemblies, the introduction of ABB fuel into the core should not increase the potential of fission product release to the environment or the dose to control room personnel as a result of a fuel handling accident. This conclusion is based on the structural characteristics of ABB reload fuel. ABB reload fuel is lighter than other fuel designs evaluated in current safety analyses and more resistant to failure mechanisms associated with fuel handling accidents.

To assess potential fuel handling accidents for ABB reload fuel, the fuel handling accident analysis can be divided into two parts: 1) determining the quantity and type of fission products which are released into the reactor coolant and 2) determining the quantity and type of fission products which are released from reactor coolant to the containment and out into the environment.

The ABB reload methodology involves a comparison of the postulated accident consequences for the new fuel assembly type being evaluated (referred to below as the "new assembly") with the postulated accident for the "reference assembly" evaluated in the existing plant safety analysis. The existing plant safety analysis is bounding for the new fuel assembly being evaluated if it can be conservatively demonstrated that the total fission product release into the reactor coolant as a result of a fuel handling accident involving the new assembly is less than the release for the reference assembly evaluated in the existing plant analysis. In this case, calculation of releases to the environment and resulting exposure to the public and onsite personnel are not necessary.

To determine if the existing analysis is bounding, the following issues are addressed:

(1) The weight of the new fuel assembly relative to the reference assembly,

(2) The number of failed rods in the existing analysis based on the reference assembly relative to the number of rods which will fail in a new fuel assembly,

(3) The gaseous fission product inventory in the new assembly failed rods relative to that assumed in the existing safety analysis based on the reference assembly.

Fuel Bundle Weight

The weight of the dropped fuel assembly is an important parameter in determining the number of fuel rods damaged in the fuel assemblies struck by the dropped assembly. If the new fuel assembly is heavier than the reference assembly, the number of failed fuel rods may increase if the heavier new assembly is dropped on reference fuel assemblies. In this case, the original analysis will require reevaluation and the number of failed fuel rods in any of the reference assemblies must be determined when a new assembly is dropped on it.

If the maximum weight of the new assembly is less than or equal to the assembly assumed to be dropped in the existing analysis, it is sufficient to determine the number of fuel rods that fail in a new assembly as a result of being struck by heaviest reference assembly dropping on it. Any other combination of dropped and impacted assemblies is bounded by this analysis and the original analysis.

Number of Damaged Fuel Rods

The complex nature of the impact and the resulting fuel damage to the fuel assemblies makes a rigorous prediction of the number of failed fuel rods complex. Typically, a simplified energy approach is used in conjunction with a number of conservative assumptions to estimate the number of rods damaged during the event. The assembly is assumed to drop from the position which maximizes the drop distance and, therefore, maximizes the kinetic energy of the dropped assembly when it impacts the target assemblies. The dissipation of energy during the fuel assembly's fall through water is assumed to be negligible. Therefore, the entire kinetic energy is assumed to be absorbed by the assemblies involved in the event.

The dropped assembly is assumed to impact the core at a small angle relative to the vertical direction, possibly inducing a bending mode of failure. It is assumed that each rod resists the imposed bending load by a couple consisting of two equal and opposite concentrated forces. The energy absorbed in the bending mode before failure is relatively small. Therefore all the rods in the dropped assembly are assumed to fail.

[ Proprietary Information Deleted ]

Since the assembly handle is struck by the falling fuel assembly, it is necessary to distinguish between assembly designs for which a load on the handle is directly transmitted to the fuel rod cladding and one for which a load on the handle is transmitted to the channel.

[ Proprietary Information Deleted ]

[ Proprietary Information Deleted ]

It is possible that the falling assembly will impact more than one assembly in the core, possibly as many as four assemblies in the first impact. Depending on the design of the bundle and the handle, the available energy is conservatively transferred to impacted assemblies in a conservative manner which maximizes the number of failed fuel rods.

[ Proprietary Information Deleted ]

## 8.5 Misplaced Assembly Accident

The misplaced fuel assembly accident, also sometimes referred to as a fuel loading error, can consist of a fuel assembly mislocated in a incorrect location or a fuel assembly in the proper location rotated into a misoriented position.

### 8.5.1 Mislocated Fuel Assembly

#### 8.5.1.1 Design Basis

*Basis*

This event is considered to be an accident in the ABB reload safety analysis process. The SLMCPR is used as the event acceptance limit for this accident.

*Discussion*

[ Proprietary Information Deleted ]

#### 8.5.1.2 Event Description

This accident is the postulated placement of a fuel assembly in a location other than that assumed in the Reference Core. This causes a discrepancy between the Reference Core configuration and the actual core configuration. An erroneous thermal-hydraulic and nuclear behavior is assumed for the mislocated assembly. Furthermore, differences in nuclear and thermal-hydraulic performance characteristics between the mislocated assembly and the assembly intended for that location can cause monitoring errors in the core supervision system.

It is assumed that the loading error is not detected and that the plant operates for the entire cycle with the misloaded bundle in accordance with the core operating limits for the Reference Core. The accident is extremely improbable since a fuel assembly must be loaded into the wrong location, the fuel assembly intended for that location must be placed in an improper location or not loaded in the core, and the error must be overlooked during the core verification.

[ Proprietary Information Deleted ]

### 8.5.1.3 Analysis Methodology

The mislocated assembly analysis is performed under the following assumptions:

[ Proprietary Information Deleted ]

## 8.5.2 Rotated Fuel Assembly Accident

### 8.5.2.1 Design Bases

*Basis*

This event is considered to be an accident in the ABB reload safety analysis process. The SLMCPR is used as the event acceptance limit for this accident.

*Discussion*

[ Proprietary Information Deleted ]

### 8.5.2.2 Event Description

This accident is the postulated rotation of a fuel assembly relative to the orientation assumed in the Reference Core. The postulated rotation modifies the orientation of the fuel pins relative to the interassembly gaps and changes the interassembly gap widths. The interassembly gap widths are changed due to the interference of the channel spring clip with the upper core grid. Rotations of 90° and 180° relative to the correct orientation are considered. A rotation of 270° is equivalent to the 90° rotation due to the symmetry of BWR fuel assemblies.

As a result of the accident, the power distribution within the assembly is changed with a corresponding change in CPR. Since the core supervision system assumes correct assembly orientation, the predicted margin to the SLMCPR could be incorrect.

It is assumed that the misorientation is not detected and that the plant operates for the entire cycle with the misoriented assembly in accordance with the core operating limits for the Reference Core.

The severity of the event depends on the lattice design. A C-lattice core has symmetric interassembly gaps for a correctly installed assembly. Therefore, the deviation from the Reference Core is due to the change in gap sizes associated with the interference of the channel spring clip with the upper core grid. The impact of the rotation for the D-lattice case may be somewhat greater due to the asymmetric interassembly gap widths for the nominal orientation and the nuclear design of the bundle. The enrichment distribution in D-lattice assemblies tend to be less symmetric than for C-lattice assemblies to compensate for the asymmetric nominal gap widths.

The most severe challenge to the SLMCPR can occur at any time during the cycle. It is assumed that at any time during the cycle a control rod configuration could be selected which would place a fuel assembly in the Reference Core on the MCPR operating limit and cause an assembly in the core containing the misoriented assembly to exceed those limits.

Since this event is considered to be an accident, no other AOOs or equipment failures are assumed to occur during the cycle with the misoriented bundle.

### 8.5.2.3 Analysis Methodology

[ Proprietary Information Deleted ]

# 9 SPECIAL EVENTS ANALYSIS

Special events are evaluated to demonstrate plant capabilities required by regulatory requirements and guidance, industry codes and standards, and licensing commitments. The Special Events considered in the plant safety analysis are dependent on the goals of the analysis. The ABB safety analysis methodology for evaluating Special Events is described in this section.

Generically, three Special Events are analyzed for a ABB reload application. The Special Events are:

- Core Thermal-Hydraulic Stability,
- Reactor Overpressurization Protection, and
- Standby Liquid Control System Capacity.

In addition, ABB reload safety analysis methodology has the capability to evaluate:

- Anticipated Transients Without Scram events.

This analysis capability may be required for the evaluation of specific modifications necessary to demonstrate acceptable plant capability.

## 9.1 Summary and Conclusions

<u>Summary</u>

This section describes the process of establishing the plant operating limits defined by the safety analysis of the limiting Special Events for an ABB reload application. Four Special Events are addressed in the ABB reload safety analysis methodology.

The ABB reload safety analysis methodology includes the capability to analyze Core Thermal-Hydraulic Stability, as required by the plant specific reload safety analysis process. NRC approved stability analysis codes and analysis methodology are used to perform reload safety evaluations and plant modification evaluations, as required. ABB also has advanced stability tools and reload safety licensing analysis methodology, for supporting future implementations of licensing commitments related to core thermal-hydraulic stability (e.g., BWROG solutions to the "Long Term Stability Issue").

The ABB methodology performs Reactor ASME Overpressure Protection analysis to confirm for each reload application that the safety/relief overpressure protection system performance requirements are maintained. The methodology confirms for the most limiting event, MSIV closure, the maximum pressure vessel system pressure does not exceed the plant-specific design acceptance limit.

The Standby Liquid Control System (SLCS) evaluation confirms that the liquid poison reactivity control system performance requirements are satisfied for each reload application. The ABB methodology confirms for the plant technical specification requirements, plant shutdown can be attained with only the standby liquid control system.

In accordance with Federal Code of Regulations (Reference 42, 10CFR50.62), the capability to mitigate postulated Anticipated Transients Without Scram events has been demonstrated. Safety evaluations have confirmed this conclusion to be valid for reload core design. As discussed in Section 6.3.1.3, it is not necessary to evaluate ATWS events for the use of ABB reload fuel. However, the potential does exist for performing ATWS evaluations for certain types of plant modifications. The ABB safety analysis methodology does have the capability for evaluating ATWS events, if required in the evaluation of plant modifications.

*Conclusions*

Appropriate design bases and evaluation methodologies are established for the specific licensing base Special Events examined in reload application.

## 9.2 Core Thermal-Hydraulic Stability

ABB has analysis codes and methodologies to perform core thermal-hydraulic stability evaluations for plant specific reload applications and plant modifications as required. ABB has both frequency domain and time domain codes used for stability analysis (see Table 9-1). These stability analysis tools can be used for reload safety evaluations of the plant in question, based on the application methodology adopted by the utility licensee (e.g., see Table 9-2).

The following sections describe the core thermal-hydraulic stability analysis design bases, the ABB stability analysis methodology, and the reload plant application methodology.

### 9.2.1 Design Bases

*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 42). Design requirements are put on the reload fuel assemblies to also ensure compliance with the GDC-12. The corresponding fuel bundle and loading pattern design basis is discussed in Section 4.2.5.

### 9.2.2 Stability Analysis Methodology

*Methodology*

An NRC approved analysis code is used for core and channel stability margin calculations.

*Discussion*

The ABB stability analysis tools are summarized in Table 9-1. These stability tools are used, as appropriate, in supporting reload fuel and core design, plant reload applications, and plant modifications. Approved stability analysis methodology will be used in the reload safety analysis process.

The ABB frequency domain thermal-hydraulic stability analysis code, is documented in Reference 24. [ Proprietary Information Deleted ]

The ABB advanced frequency domain and 3D time domain codes are described in Reference 44. Reference 44 provides a description of the codes and qualification for core and channel stability performance evaluations. Three dimensional transient stability analysis methods are used in the ABB advanced stability methodology. Licensing Topical Report CENPD-295-P-A (Reference 45) provides a description of general stability analysis methodology using the advanced stability codes.

### 9.2.3 Plant Reload Application Methodology

*Methodology*

The reload stability evaluation performed for a specific plant reload application will be consistent with plant-specific licensing commitments. The reload stability evaluation will use approved stability methods and reload safety evaluation methodology.

*Discussion*

Each plant licensee has a stability licensing base which bounds or is confirmed for subsequent reload applications. The plant licensing base may change as plant modifications, such as modifications supporting stability detection and suppression, are implemented. ABB shall use an NRC approved reload evaluation methodology consistent with the specific plant licensing base. Examples of plant reload application methodologies are shown in Table 9-2.

## 9.3 Overpressurization Protection

The overpressurization protection analysis is a Special Event conservatively analyzed to address the adequacy of the plant's pressure relief system. The system design is based upon ASME Code requirements (Reference 49) and NRC regulations.

### 9.3.1 Design Bases

*Basis*

The plant overpressure protection system capability shall be confirmed adequate for the cycle specific reload. The specific plant licensing basis ASME code overpressure protection design limit shall not be exceeded.

*Discussion*

Potentially limiting plant overpressurization events are analyzed to confirm that the reactor pressure limit is not exceeded. The maximum pressure acceptance limit shall be that limit established in the plant licensing basis. For most BWRs, a conservative upset condition limit of 110% of design pressure is used in the code overpressure protection analysis.

### 9.3.2 Overpressurization Protection Methodology

*Methodology*

The most severe pressurization event is analyzed for each reload cycle to confirm the adequacy of the plant's pressure relief system. The most severe pressurization event used in the overpressure protection analysis is the MSIV closure with failure of direct scram signal. The evaluation procedure for this event is:

[ Proprietary Information Deleted ]

The overpressurization MSIV closure event is analyzed with the NRC approved dynamic analysis methods (i.e., see Appendix A.4.1). The plant model developed for rapid pressurization events analysis is also used for calculating the ASME overpressurization event.

[ Proprietary Information Deleted ]

*Discussion*

The overpressurization MSIV closure event could be treated as an emergency condition consistent with the current version of the ASME code (Reference 49), with acceptable results compared to the ASME emergency condition limits (i.e., the reactor pressure acceptance limit of 120% of design pressure). However, the current approach is to maintain a margin of conservatism in the methodology by treating this event as an upset condition. Under this classification the ASME upset acceptance limit is used (i.e., the reactor pressure is not to exceed 110% of design pressure.) Because of the conservatism in this approach, and conservatism assumed in the event conditions, no other failures are assumed.

## 9.4 Standby Liquid Control System Capability

### 9.4.1 Design Bases

*Basis*

The Standby Liquid Control System (SLCS) shall be capable of shutting the reactor down from the most reactive reactor operating state at any time in cycle life.

The acceptance limit is a calculated reactivity demonstrating that the reactor is shutdown for the most reactive moderator temperature at any time during the cycle for the boron concentration selected for the plant SLCS.

*Discussion*

Two independent reactivity control systems are provided in BWRs, namely control rods and soluble boron in the coolant from the Standby Liquid Control System. The control rod system is the mechanical system that can compensate by itself for the reactivity effects of the fuel and water temperature and density changes accompanying power level changes over the complete range from full-load to no-load, cold, xenon-free conditions. The control rod system alone provides the minimum shutdown margin under all operating conditions and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits assuming that the highest worth control rod is stuck out upon trip. This capability is available at all times in core life at all operating states. Confirmation of minimum shutdown margin by the control rod system is verified as discussed in Section 4.3.

The Standby Liquid Control System provides an alternate means of attaining and maintaining the reactor in the shutdown state by injecting boron into the reactor vessel. At any time in core life, the

SLCS must be capable of bringing the reactor to a shutdown condition from any operating state, assuming no movement of the control rods. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical poison system, satisfying General Design Criteria 26 and 27 of 10 CFR 50, Appendix A (Reference 42).

### 9.4.2 SLCS Evaluation Methodology

Standby Liquid Control System performance is evaluated to demonstrate independent shutdown ability for each cycle. The analysis of the SLCS shutdown capability is done using an NRC approved three-dimensional core simulator code (see Appendix A). The evaluation is performed for the reload safety analysis Reference Core design. The minimum SLCS shutdown capability is established at the point in the cycle that produces the largest reactivity defect from the operating reactor state to the cold (most-reactive) xenon-free condition, assuming no movement of the control blades during the SLCS shutdown procedure.

[ Proprietary Information Deleted ]

These calculations are performed to confirm that the reactor will be shutdown with the minimum boron concentration defined in the plant technical specifications with no movement of control rod positions from their initial state. The core must be shutdown at any temperature between hot operating and cold, shutdown conditions. [ Proprietary Information Deleted ]

The moderator cross sections with the appropriate boron concentrations are calculated with the same NRC-approved lattice physics code used to generate the nuclear data for the Reference Core calculations (see Appendix A). Branch calculations from the main line lattice physics code depletion calculations supporting the three-dimensional nodal simulator Reference Core model are performed with the appropriate boron concentration. These cross sections are utilized in the three-dimensional nodal simulator to evaluate the impact of the borated moderator on core reactivity.

## 9.5 Anticipated Transients Without Scram (ATWS)

Anticipated Transients Without Scrams (ATWS) are anticipated operational occurrences followed by a failure of the reactor trip portion of the reactor protection system. BWR plants require alternative reactivity insertion systems and features to mitigate the consequences of this postulated event as addressed in 10 CFR 50.62 (Reference 42). ATWS evaluations are not required for reloads. ATWS evaluations are performed only for plant modifications that have the potential to challenge the event acceptance limits.

### 9.5.1 Design Bases

*Bases*

The BWR plant design bases for a postulated ATWS event are:

(1) Fuel Integrity: The core and fuel must maintain a coolable geometry.

(2) Containment Integrity: The containment pressure must not exceed the design limit.

(3) Primary System Integrity: The reactor system transient pressure must be limited such that the maximum primary stress within the reactor coolant pressure boundary does not exceed Service Level C of the ASME Boiler and Pressure Vessel Code Article NB-3000 of Section III.

(4) Long-Term Shutdown Cooling: Subsequent to the ATWS event, the capability must exist (a) to bring the reactor to a safe condition without depending on control rod insertion, and (b) to achieve and maintain a cold shutdown condition.

These criteria are used to demonstrate plant compliance with the ATWS Rule of 10 CFR 50.62.

Acceptance limits used to demonstrate compliance with the design bases are:

- Maximum Cladding Temperature less than 2200 °F
- Containment Pressure less than Containment Design Pressure
- Peak Reactor Vessel Pressure less than 120% of Vessel Design Pressure
- Radiation Dose less than guideline values of 10 CFR 100 (Reference 42)
- Demonstrated Equipment Availability

*Discussion*

[ Proprietary Information Deleted ]

### 9.5.2 ATWS Evaluation Methodology

An ATWS evaluation is performed for each plant modification that has the potential to challenge the ATWS event acceptance criteria. The methodology for a plant modification consisting of the introduction of an ABB fuel design is described below.

*Methodology*

Each new ABB fuel design introduced into a plant is confirmed to comply with the design characteristic of the core assumed in the plant licensing basis ATWS analysis. [ Proprietary Information Deleted ] Once the ABB fuel design in confirmed not to have a significant impact in the current ATWS analysis, it is considered acceptable.

*Discussion*

ABB fuel designs are generally demonstrated to be less limiting than fuel designs assumed in the plant licensing basis ATWS analysis, by intrinsic mechanical design features which result in larger margins to fuel integrity limits (i.e., lower linear heat generation rate than for larger diameter fuel rod designs).

## TABLE 9-1

## ABB STABILITY ANALYSIS TOOLS

| Tool | Methods | Methods Qualification | Analysis Methodology |
|---|---|---|---|
| Traditional Frequency Domain Code | RPA-90-91-P-A (NUFREQ code) | RPA-90-91-P-A | RPA-90-91-P-A |
| Advanced 3D Time Domain Code | CENPD-294-P-A (RAMONA code) | CENPD-294-P-A | CENPD-295-P-A |

RPA-90-91-P-A (Reference 24)
CENPD-294-P-A (Reference 44)
CENPD-295-P-A (Reference 45)
10 CFR 50, Appendix A (Reference 42)

## TABLE 9-2

## EXAMPLES OF STABILITY LICENSING METHODOLOGIES FOR PLANT RELOAD APPLICATIONS

| Plant Reload Application | Methodology |
|---|---|
| Traditional Stability Evaluation | (1) Compliance with NRC Bulletin 88-07 and Supplement 1 (Reference 48)<br>(2) Plant Specific Licensing Commitments |
| BWROG Option IA Enhance Evaluation | Described in NEDO-32339 (Reference 55) |
| BWROG Option ID Evaluation | Described in NEDO-31960 (Reference 54) |
| BWROG Option II Evaluation | Described in NEDO-31960 (Reference 54) |
| BWROG Option III Evaluation | Described in NEDO-31960 (Reference 54) |
| ABB Advanced Reload Evaluation | Described in CENPD-295-P-A (Reference 45) |

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# APPENDIX A: DESCRIPTION OF CODES

## A.1 Mechanical Design

### A.1.1 Fuel Rod Performance Codes

#### A.1.1.1 VIK-2

VIK-II calculates the cladding stresses at the beginning of life (BOL) for a fuel rod. The code consists of subroutines which calculate different stresses on the cladding. These individual stresses are condensed and compared to the allowed stresses specified by the appropriate design criterion. Standard analytical expressions are used to calculate the stresses.

A complete description of the VIK-2 code is provided in Reference 36.

#### A.1.1.2 STAV6.2

STAV6.2 is the latest version of the code series STAV (the Swedish word for "rod") used for BWR fuel performance by ABB. STAV6 offers a state-of-the-art analytical tool for predicting steady-state fuel performance for operation of light water reactor (LWR) fuel rods including $Gd_2O_3$-$UO_2$ fuel. STAV6.2 is the primary analysis code used in fuel thermal mechanical design process.

STAV6.2 calculates the variation with burnup of all important fuel rod performance quantities including fuel and cladding temperatures, fuel densification, fuel swelling, fission product gas release, rod pressure, gas gap conductance, cladding stresses and strains due to elastic and thermal creep and plastic deformations, cladding oxidation, and cladding hydriding. Burnup-dependent radial power distributions for both $UO_2$ and $Gd_2O_3$-$UO_2$ fuel, fuel grain growth, and helium release are modeled in the code.

For example, in the reload safety analysis process, STAV6.2 is used to establish the fuel thermal mechanical performance limit. It is also used to develop the calculated fuel rod inputs to the nuclear design, thermal hydraulic, and safety analysis process.

A complete description of the STAV6.2 code is provided in Reference 36.

#### A.1.1.3 COLLAPS-II

COLLAPS-II is used by ABB for prediction of cladding ovality and cladding creep down in BWRs as a function of irradiation time.

The COLLAPS-II code models the cladding as a long, thin cylindrical tube which is subject to creep as a result of a uniform external force. The cross section of the tube is assumed to have a slight initial deviation from circularity. The standard assumptions appropriate to creep deformation analysis of shells are utilized in the COLLAPS-II code.

COLLAPS-II calculates the following quantities as a function of irradiation time:

- Cladding ovality,

- Creep down strain and total axial strain of the cladding, and

- Bending moments of the cladding.

A complete description of the COLLAPS-2 code is provided in Reference 36.

### A.1.2 Finite Element Model Analysis Codes

### A.1.2.1 ANSYS

ANSYS is a large-scale, general purpose code recognized world-wide for its many capabilities. It is used extensively in power generation and nuclear industries. The code is developed and supported by the Swanson Analysis System, Inc., Houston, Pennsylvania. The code's capabilities include:

- Static and dynamic structural analysis, with linear and nonlinear transient methods, harmonic response methods, mode-frequency method, modal seismic method, and vibration analysis.

- Buckling and stability analysis with linear and nonlinear buckling.

- Heat transfer analysis with transient capability and coupled thermal structural capabilities.

- Ability to model material nonlinearities such as, plastic deformation, creep, and swelling.

- Fracture mechanics analysis.

The ANSYS element library consists of 78 distinct element types. However, many have option keys which allow further specialization of element formulation in some manner, effectively increasing the size of the element library.

The reliability and accuracy of ANSYS software is maintained by a rigorous quality assurance program. A library of verification problems now numbering over 2000, is continuously updated to reflect the changes and new features in the program.

## A.2 Nuclear Design

A series of codes are utilized for the nuclear design and nuclear safety analysis. The two major computer codes used in the nuclear design are the PHOENIX and POLCA codes which are briefly described below. A complete description of the nuclear design and analysis codes is provided in Reference 19.

### A.2.1 Two Dimensional Lattice Design

#### A.2.1.1 PHOENIX

PHOENIX is a two-dimensional, multi-group transport theory code which is used for the calculation of eigenvalue, spatial flux and reaction rate distributions, and depletion of rod cells for BWR and PWR fuel assemblies. The code can simulate BWR cruciform control blades containing cylindrical absorber elements, PWR cluster control rods, water gaps, burnable absorber rods, burnable absorbers that are integral with the fuel, water rods, and the presence of objects in the water gaps such as neutron detectors.

PHOENIX is supported by the burnable absorber program FOBUS and by the PHOENIX library service program PHOEBE. PHOENIX is the standard ABB depletion program for BWR fuel assembly and rod cell calculations. ABB Atom also uses PHOENIX for PWR fuel assembly and rod cell calculations. Each of the fuel rods is individually treated throughout the calculations. There is no limitation on the number of different rod types that can be represented in the PHOENIX problem. The code can accommodate a variety of geometric configurations including fuel rods with different radii, plutonium fuel, burnable absorber rods, and water holes. Any number of objects, such as detectors, control blades, and control blade tips, may be specified in the water gaps. These are either treated homogeneously or, in the case of a control blade with absorbing rods, heterogeneously. In addition to rod cell and fuel assembly calculations, quadruple assembly calculations, consisting of four assemblies in a 2x2 array, can be performed. This option is used for the detailed calculation of rod-wise power distributions, reaction rates, reactivities, and detector constants for the case of different types of adjacent fuel assemblies in a mixed core. It is also used for detailed evaluations of the impact of channel bow.

PHOENIX provides the two-dimensional cross section libraries used by the three-dimensional core simulator POLCA. It also produces the local peaking patterns used as input to the critical power margin calculation and the emergency core cooling system evaluation model GOBLIN-EM system of computer codes.

## A.2.2 Three Dimensional Nodal Core Simulator

### A.2.2.1 POLCA

POLCA is a core simulator which provides realistic three-dimensional simulations of the nuclear, thermal, and hydraulic conditions in boiling water reactors.

The nodal equations are based on a specially adapted coarse-mesh diffusion approximation. A set of coupling coefficients describes the inter-nodal coupling. These coefficients are evaluated from two-group data which are stored as a number of three-dimensional tables. The table entries are burnup, void, and void history. The voids affect the neutron energy spectrum and cross sections, while the void history affects the isotopic composition per node. The neutronics equations are solved by Gauss-Seidel inner iterations with a Chebyshev iteration of the fission source. A thermal coupling correction, based on the asymptotic thermal fluxes of the direct neighbors, is made by modifying the removal cross sections prior to the iteration process.

The hydraulic calculations are performed by a special version of the CONDOR thermal-hydraulic code described in Section A.3.1.

In addition to the linear heat generation rate and CPR edits, POLCA also edits bundle, core average axial, and three-dimensional nodal distributions of power, burnup, void, xenon, and iodine concentrations. Further, inlet flow distributions, local power range monitor (LPRM) and traversing in-core probe (TIP) signals predicted by POLCA can be edited. POLCA can be used to perform criticality searches on such parameters as reactor power, recirculation pump flow, inlet subcooling, and control rod position. POLCA can be run in eighth-, quarter-, half-, or full-core configurations. Each fuel assembly is modeled radially using one node per assembly and typically 25 nodes axially, which permits the explicit modeling of the top and bottom natural uranium blanket regions.

In the safety analysis process, POLCA is used in the analysis of slow (quasi-steady state Anticipated Operational Occurrences) and fuel loading errors. It also provides input to the BWR dynamic analysis methods BISON and RAMONA.

## A.3 Thermal-Hydraulics Design

### A.3.1 CONDOR

ABB utilizes the CONDOR code for the evaluation of the steady-state thermal-hydraulic performance of BWR primary systems. This program is also used as the thermal-hydraulic module of the three-dimensional core simulator code, POLCA.

CONDOR is used for the thermal-hydraulic analysis of a single fuel assembly, a reactor core, or a complete light-water reactor system. It calculates the steady-state variation of pressure, enthalpy, temperature, and flow along the entire coolant flow path through the system. It also calculates 3D core distributions of pressure, enthalpy, temperature, flow, heat flux, steam quality, void fraction, and minimum critical power ratio (MCPR).

A complete description of the code is provided in Reference 20.

## A.4 Safety Analysis

### A.4.1 One Dimensional Time Domain Dynamic Analysis

#### A.4.1.1 BISON

Fast and moderate-speed core-wide transients are analyzed with the BISON transient analysis system of codes. As described in Section A.2.2, slow and localized transients are modeled with the POLCA three-dimensional steady-state core simulator.

BISON has a one-dimensional thermal-hydraulic model for the coolant loop of the reactor vessel, which can accommodate internal, external and jet pumps. The coolant loop is divided into regions, i.e., downcomers, external recirculation loop, jet pumps, a core coolant and a bypass channel, riser and steam separator, which are further divided into subregions.

A complete description of BISON is provided in References 23 and 39.

### A.4.2 Three Dimensional Time Domain Dynamic Analysis

#### A.4.2.1 RAMONA-3

RAMONA-3 is a systems transient code for prediction of the dynamic behavior of a BWR. It is specifically designed to simulate normal and abnormal operational plant transients, as well as accidents such as the ATWS transients, Control Rod Drop Accident and time domain stability analyses. RAMONA-3 also has been used to simulate a rod withdrawal error during startup and can be used in other transient applications requiring complete three-dimensional representation.

Because of its unique feature of combining full 3-D modeling of the reactor core and transient plant response, it is particularly suited for transients showing large local effects in the core.

A detailed description of the modeling characteristics in RAMONA-3 for neutron kinetics, thermal conduction, and thermal-hydraulics are given in Reference 44.

## A.4.3 ECCS Evaluation

### A.4.3.1 GOBLIN Series

The GOBLIN-EM system of computer codes uses one-dimensional assumptions and solution techniques to calculate the BWR transient response to both large and small break loss of coolant accidents. The code system is composed of three major computer programs – GOBLIN-EM, DRAGON and CHACHA-3C. The functions of the individual codes are:

> GOBLIN-EM performs the thermal-hydraulic calculations for the entire reactor primary system including interactions with the various safety systems.
>
> DRAGON performs the thermal-hydraulic calculations for a specified fuel assembly in the reactor core. The GOBLIN code provides DRAGON with the necessary boundary conditions.
>
> CHACHA-3C calculates the detailed temperature distribution at a given axial cross section of the assembly analyzed by DRAGON. Its input, boundary conditions, are supplied by GOBLIN-EM and DRAGON.

A detailed description of these codes is provided in References 21 and 40.

## A.4.4 Frequency Domain Stability Analysis

### A.4.4.1 NUFREQ-NPW

NUFREQ-NPW is a frequency domain computer program developed for homogeneous and mixed BWR core nuclear coupled stability analysis. It is based on drift flux thermal-hydraulics, arbitrary non-uniform axial and radial power profiles, distributed local losses, detailed nuclear fuel and heater rod dynamics, and point or multi-dimensional neutron kinetics. Different assembly types can be represented (e.g., 8x8, 9x9, SVEA-96). Further, this code allows for excitation by several external perturbations, including dome pressure, for direct comparisons of evaluated transfer functions against measured plant data.

All major BWR systems are modeled in NUFREQ-NPW, including the bypass, upper plenum, riser, steam separators, feedwater, recirculation, and jet pump dynamics. The code provides the system frequency response transfer functions which are evaluated for absolute and relative stability performance.

A detailed description of NUFREQ-NPW is provided in Reference 57.

#### A.4.4.2 MAZDA-NF

MAZDA-NF is a frequency domain stability program developed specifically for detailed channel stability analysis including parallel channel flow communication. The mathematical models incorporated in MAZDA-NF can accommodate phase slip, arbitrary axial power distribution, distributed local losses, channel-to-channel radial power skews, discrete or continuous flow communication between channels, adiabatic two-phase flow dynamics, external single phase loop dynamics, and nuclear reactivity feedback.

The system geometry and the operating conditions are specified as input. For the case of parallel channels the code evaluates flow splits, pressure drops and enthalpy/void distributions in the system. The relative stability indicators are evaluated from the frequency response solution of the two-phase conservation equations in two-dimensional form.

A detailed description of MAZDA-NF is provided in Reference 56.

## A.5 Statistical Analysis

### A.5.1 Industry Accepted Codes

#### A.5.1.1 SIGMA

The SIGMA code is used to combine Gaussian, uniform and arbitrary probability distributions into a resultant distribution using a "Monte Carlo" technique. The code first generates data populations conforming to input probability distributions of each independent variable. Next, the data populations are sampled randomly in order to generate the dependent variable probability distribution through use of a user supplied functional relationship. The theoretical bases of this code involves a Monte Carlo simulation incorporating variance reduction using stratified sampling techniques.

The NRC approved methodology which incorporates SIGMA is described in Reference 61.

## A.5.2 Utility Provided Codes

There are some codes used by ABB to perform statistical analysis that are approved by NRC for use by the utility. The utility can provide these codes to ABB for use on reload design analyses for their plant(s). An example of this type code is the statistical analysis code STARS (Statistical Transient Analysis by Response Surface). STARS is a PC-DOS computer code designed to apply the EPRI statistical combination of uncertainties (SCU) methodology to a variety of plant performance and safety analyses. Since it is highly unlikely that all of the event analysis inputs would be simultaneously at their most adverse or design limit values, it is logical to treat the most sensitive parameter(s) in a statistical manner. The SCU methodology provides a mathematically rigorous and computationally efficient way of reducing the sources of unnecessary conservatism in plant analyses.

A complete description of the STARS code is provided in Reference 58. The NRC approved methodologies which include the use of the STARS code are described in References 59 and 60.

# APPENDIX B: PLANT AND CYCLE SPECIFIC RELOAD SAFETY ANALYSIS SUMMARY REPORT (RSASR)

## 1 INTRODUCTION

This appendix (See Exhibit A for format) describes the cycle specific Reload Safety Analysis Summary Report (RSASR), which summarizes the results of the analyses performed in support of an ABB supplied reload application. This appendix is intended to be used in conjunction with the main body of this report, which describes the analyses performed by ABB in support of the reload and identifies the methodology used for those analyses. Sections in this Appendix are in the same order as the sections in the main body.

It is possible that, in some cases, work related to particular activities is performed by the utility in conjunction with, or independently of, ABB. In this case results are still reported, but appropriate references are cited.

## 2 CONDITIONS FOR DESIGN SUMMARY

This section of the RSASR references and summarizes the Conditions for Design Document. The summary includes a table of the most important/critical inputs and operating flexibility options.

## 3 FUEL MECHANICAL DESIGN SUMMARY

This section of the RSASR references and summarizes the Fuel Assembly Mechanical Design Report. The section includes a brief discussion regarding any new design features, relative to the previous RSASR (e.g., debris filter).

The ABB methodology for fuel assembly and fuel rod mechanical evaluation, the design criteria, and design methodology are discussed in Section 3 of the main body of this report.

## 4 NUCLEAR DESIGN SUMMARY

This section of the RSASR references and summarizes applicable parts of the Reload Design Report. The summary includes the following:

<u>Reference Core Loading</u>

1. A table of bundle types showing number and cycle loaded in the Reference Core which meets the required energy output and cycle length.
2. Core map showing the assumed core loading, by fuel type, in the Reference Core. This core loading is used to develop control rod sequences and expected core power, burnup, and void history

distributions to support the cycle reload safety analyses. Please see Sections 4 and 6 of the main body of this report for a discussion of these items.

3. Reference Core assumed BOC and EOC core average exposures.

4. Assumed previous cycle nominal core average exposure.

Reference Core Calculated Reactivity Characteristics

1. Beginning of Cycle, Cold $K_{effective}$ for the following:

   a. All Rods In

   b. All Rods Out

   c. Strongest Rod Out

2. Maximum increase in core cold reactivity during the cycle (Reactivity Defect), R

These values are used to verify that Shutdown Margin is acceptable and within Technical Specifications. Please see Section 4 of the main body of this report for a discussion concerning this item.

## 5 THERMAL-HYDRAULIC DESIGN SUMMARY

This section of the RSASR references and summarizes applicable parts of the Reload Design Report. The summary includes the following:

Bypass Flow Fraction

As discussed in Section 5.2 of Section 5 of the main body of this report, at rated power and flow conditions the total interassembly bypass flow will be maintained within the design range of the plant (typically 8 to 12% of total core flow).

Safety Limit Minimum Critical Power Ratio (SLMCPR)

The Safety Limit MCPR (SLMCPR) is established such that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The methodology for establishing SLMCPR values is provided in Section 5.3 of the main body of this report.

## 6 TRANSIENT ANALYSIS RESULTS SUMMARY

This section of the RSASR references and summarizes applicable parts of the Reload Safety Analysis Report. The summary includes the following:

Anticipated Operational Occurrences (AOOs) at Full Power

As described in Sections 6 and 7 of the main body of this report, plant and cycle specific analyses are performed to determine the impact of the most limiting AOOs on the MCPR. Events investigated include:

Fast Transients - Turbine Trip Without Bypass, Generator Load Reject Without Bypass, Feedwater Controller Failure - Max. Demand, Pressure Regulator Failure - Closed (BWR/6 only)

Slow Transients - Loss of Feedwater Heating, Control Rod Withdrawal Error (RWE), Recirculation Flow Controller Failure - Increasing Flow

Other Events on a plant specific basis (if necessary), based on specific plant licensing commitments.

Results for the limiting AOOs are summarized in this section of the RSASR, including ΔCPR values associated with the events.

Operating Limit Minimum Critical Power Ratio (OLMCPR)

The Operating Limit Minimum MCPR (OLMCPR) is set such that the SLMCPR is not violated during steady state operation or during any AOO. This is accomplished by combining the highest calculated ΔCPR (from the AOOs or Misplaced Assembly Accident) to the SLMCPR. Plant operation within the resulting OLMCPR ensures that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition during steady state reactor operation, as well as during AOOs.

The treatment of MCPR for ABB and non-ABB fuel to assure that the OLMCPR is satisfied during reactor operation and during the reload design phase are discussed in Section 5.3 of Section 5 of the main body of this report.

Operating Limit LHGR

The operating limit LHGR is set such that thermal/mechanical fuel limits are met during steady state operation and AOOs. In addition, the LHGR limit may also include other plant specific operating restrictions.

In combination with the MAPLHGR limits, the operating limit LHGR helps ensure that compliance to the SAFDLs is maintained under all design basis conditions.

Off-Rated Operation

This section of the RSASR presents the necessary figures and tables, with respect to MCPR and LHGR, to support operation at other than rated conditions (e.g., MCPR(F) and MCPR(P) curves for ARTS). These tables and curves are plant specific, depending on the type of plant, as well as its licensed allowable operating domain.

# 7 ACCIDENT ANALYSIS RESULTS SUMMARY

This section of the RSASR references and summarizes applicable parts of the Reload Safety Analysis Report. The summary includes the following:

Loss of Coolant Accident

MAPLHGRs are presented in tabular form as a function of average planar exposure. (Note: MAPLHGRs for re-insert assemblies are presented in their respective RSASRs).

Also presented in this section of the RSASR are necessary figures and tables, with respect to MAPLHGR, to support operation at other than rated conditions. These tables and curves are plant specific, depending on the type of plant, as well as its licensed allowable operating domain. This includes any necessary single loop multiplier on MAPLHGR values for those plants licensed for single loop operation.

Control Rod Drop Accident

Results of the plant and cycle specific analysis are summarized in this section of the RSASR, including the worth of the limiting dropped rod and the deposited energy of the dropped rod.

A discussion concerning the design basis and methodology for this event is presented in Section 8.3 of the main body of this report.

Misplaced Assembly Accident (Fuel Loading Error)

The calculated ΔCPRs, relative to the Reference Core steady state case, are presented for the Mislocated and Rotated Fuel Assembly Accidents.

The reason for using an AOO acceptance criteria, as well as the methodology for the event, is presented in Section 8.5 of the main body of this report.

# 8 SPECIAL EVENTS ANALYSIS RESULTS SUMMARY

This section of the RSASR references and summarizes applicable parts of the Reload Safety Analysis Report. The summary includes the following:

Core Thermal-Hydraulic Stability

Summary results are presented for any analysis done to support Licensee commitments in this area. Due to the wide variation in Licensee commitments and approaches to satisfying GDC-12, a generic approach is not currently feasible.

Stability design bases and a discussion concerning methodology are contained in Section 9.2 of the main body of this report.

Overpressurization Protection

Results of the plant and cycle specific analysis are summarized in this section of the RSASR, including the peak calculated vessel pressure and the ASME pressure limit.

Design bases and a discussion concerning methodology are contained in Section 9.3 of the main body of this report.

Shutdown Without Control Rods (SLCS Capability)

Core reactivity is given, assuming the following:

- No movement of control rods
- Most reactive moderator temperature
- Xenon free final conditions
- Standby Liquid Control System has injected the Technical Specification minimum boron (typically 660 ppm) in the reactor.

Design bases and a discussion concerning methodology are contained in Section 9.4 of the main body of this report.

# Exhibit A

# RELOAD SAFETY ANALYSIS SUMMARY REPORT (RSASR)

# FOR

# {PLANT NAME}

# CYCLE {N}

{LEGAL NOTICE}

**NOTE**

**Throughout this Exhibit, the use of { } indicates that plant, fuel, and/or cycle specific information is to be entered here in the actual document.**

## I. CONDITIONS FOR DESIGN SUMMARY

### A. Important Analytical Inputs

| Parameter | Value |
|---|---|
| Rated Thermal Power (MWth) | {. . . } |
| 100% Core Flow (Mlb/hr) | {. . . } |
| {. . . } | {. . . } |
| {. . . } | {. . . } |
| {. . . } | {. . . } |
| {. . . } | {. . . } |
| {. . . } | {. . . } |
| Power/Flow Operating Domain | Figure { } |

### B. Operating Flexibility Options

{List (e.g., Single Loop Operation, Final Feedwater Temperature Reduction, etc.)}

### C. References

1. Applicable Conditions for Design Document.
2. Other as needed.

## II. FUEL MECHANICAL DESIGN SUMMARY

### A. Reload Fuel Bundle Design - Figure { }

{Figure providing general, brief description of the reload bundle, channel, etc.}

### B. New Design Feature(s)

{Brief description of any new mechanical design features relative to previous cycle, e.g., debris filter}

### C. References

1. Applicable Fuel Mechanical Design Report.
2. Other as needed.

# III. NUCLEAR DESIGN SUMMARY

## A. Reference Core Loading

| Bundle Type | Number in the Core | Cycle Loaded |
|---|---|---|
| { Description } | { } | { } |
| { Description } | { } | { } |
| { Description } | { } | { } |
| { Description } | { } | { } |

| | |
|---|---|
| Reference Core loading pattern | Figure { } |
| Reference Core assumed BOC core average exposure | { } MWd/MTU |
| Reference Core assumed EOC core average exposure | { } MWd/MTU |
| Assumed previous cycle nominal core average exposure | { } MWd/MTU |

## B. Reference Core Calculated Reactivity Characteristics

| | |
|---|---|
| BOC, $K_{eff}$ - All Rods In | { } |
| BOC, $K_{eff}$ - All Rods Out | { } |
| BOC, $K_{eff}$ - Strongest Rod Out | { } |
| R - Reactivity Defect | { } $\Delta k/k$ |

## C. References

1. Applicable Reload Design Report.

2. Other as needed.

# IV. THERMAL HYDRAULIC DESIGN SUMMARY

## A. Bypass Flow Fraction: { }

## B. Safety Limit Minimum Critical Power Ratio (SLMCPR)

| Fuel Type | SLMCPR | |
|---|---|---|
| {. . . } | {two loop } | {single loop} |
| {. . . } | {two loop } | {single loop} |
| " | {two loop } | {single loop} |
| " | {two loop } | {single loop} |
| {. . . } | {two loop } | {single loop} |

## C. References

1. Applicable Reload Design Report.
2. Other as needed.

# V. TRANSIENT ANALYSIS RESULTS SUMMARY

## A. Anticipated Operational Occurrences (AOOs) at Full Power

Exposure Range: { . . . }

| Event | ΔCPR | Figure |
|---|---|---|
| {Limiting Vessel Pressure Increase Event} | { . . } | {. . . } |
| Loss of Feedwater Heating ({ } °F) | { . . } | N/A |
| FW Controller Failure - Max. Demand | { . . } | {. . . } |
| Control Rod Withdrawal Error (RWE)<br>Rod Block Setpoint Selected: { } | { . . } | N/A |

## B. Operating Limit Minimum Critical Power Ratio (OLMCPR)

Exposure Range: { }

Limiting Event: {State Limiting Event}

| Fuel Type | OLMCPR | |
|---|---|---|
| {. . . } | {two loop } | {single loop} |
| " | " | |
| {. . . } | {two loop } | {single loop} |

## C. Operating Limit LHGR

| Fuel Type | Figure |
|---|---|
| {. . . } | { } |
| " | " |
| {. . . } | { } |

## D. Off-Rated Operation

{Any appropriate MCPR and LHGR dependent power and flow Figures, $K_f$ curve, etc.}

## E. References

1. Applicable Reload Safety Analysis Report.
2. Other as needed.

## VI. ACCIDENT ANALYSIS RESULTS SUMMARY

### A. Loss of Coolant Accident

Limiting Break: {Short description of limiting break for the plant}

Results: {Reload Fuel Bundle Type}

| Average Planar Exposure | Analyzed MAPLHGR |
|---|---|
| {. . .} MWD/MT | {. . .} kW/ft |
| {. . .} | {. . .} kW/ft |
| " | " |
| " | " |
| " | " |
| " | " |
| " | " |
| " | " |
| " | " |
| " | " |
| {. . .} | {. . .} |

Reduced Power and Flow Dependent Operation:

{If applicable, power and flow dependent Figures}

Single Loop Multiplier: { }

### B. Control Rod Drop Accident

Dropped Rod Worth: { } $\Delta k/k$

Peak Deposited Fuel Enthalpy: { } cal/gm

### C. Misplaced Assembly Accident (Fuel Loading Error)

| Event | $\Delta$CPR |
|---|---|
| Mislocated Fuel Assembly | { } |
| Rotated Fuel Assembly | { } |

### D. References

1. Applicable LOCA Report.
2. Applicable Reload Safety Analysis Report.
3. Other as needed.

# VII. SPECIAL EVENTS ANALYSIS RESULTS SUMMARY

## A. Core Thermal-Hydraulic Stability

{Format and content as required to meet Licensee commitments}

## B. ASME Overpressurization Protection

MSIV Closure (Flux Scram):

| | |
|---|---|
| ASME Pressure Limit (110% of design pressure): | { } psig |
| Peak Vessel Pressure: | { } psig |
| Plant Response: | Figure { } |

## C. Shutdown Without Control Rods (SLCS Capability)

| | |
|---|---|
| Min. Technical Specification Boron Concentration: | { } ppm |
| Core Reactivity at min. Tech Spec concentration, limiting moderator temperature, Xenon free: | { } $\Delta k/k$ |

## D. References

1. Applicable Reload Safety Analysis Report.
2. Others as needed.

# APPENDIX C: RELOAD LICENSING PLANT OPERATING FLEXIBILITY OPTIONS

In order to support utility needs and provide operational flexibility, additional reload safety analyses may be performed. Optional analyses generally are performed to support changes to the plant technical specifications which justify increased operational flexibility.

Some of these features are directly considered and included in the plant specific reload analysis. These features usually do not place any additional restrictions on the operating limits. Other features require additional calculations supporting operating limits when the operational state is being implemented. The following sections provide some examples of the BWR reload analyses which may be requested to provide increased operational flexibility. It should be noted that non-fuel related analyses are not discussed in the following sections, even though they may be needed to fully implement the option.

## C.1 Extension of Load Line Limits

The original design basis of most plants allowed operation on the power/flow map bounded by 100%. However, significant operational benefits are possible through use of an expanded region of the BWR power flow map. In order to justify operation in these extended regions of the power flow map, additional analyses are performed. The analyses which may be performed are:

Extended Load Line Limit Analysis (ELLLA)

Maximum Extended Load Line Limit Analysis (MELLLA)

The analysis of these operating domain extension options is performed using the same methodology and assumptions basis as the standard reload analysis, except that the extreme points on the allowable operating domain are different. To justify such operation, the following events are reviewed and evaluated to assure that core operating limits cover the limiting points on the extended operating domain:

- Stability
- Loss of Coolant Accident
- Pressurization Transients
- Rod Withdrawal Error

## C.2 Increased Core Flow (ICF)

Safety analyses justify plant operation over a range of core power and flow as shown on the core power/flow map. Generally, all flows up to

100% of rated are considered. Design margins in the recirculation system generally make possible a flow in excess of 100%. Additional flexibility in plant operation is achieved if core flow in excess of 100% is permitted. Therefore, some utilities choose to provide justification for operation with core flow in excess of 100%. Additional analyses, evaluations, and operation limit specifications are necessary to justify operation with increased core flow.

Limiting anticipated operational occurrences (AOOs) and the overpressure protection analysis are evaluated for the higher core flow. These transients include:

- Turbine Trip or Generator Load Rejection Without Bypass (TTWOB or GLRWOB)
- Feedwater Controller Failure with maximum demand (FWCF)
- Feedwater Temperature Reduction
- Rod Withdrawal Error
- Overpressurization Analysis (MSIV closure with flux scram)

The analyses of these transients are performed using the standard methodology. These analyses will demonstrate that the presence of ABB fuel in the core does not adversely affect operation with increased core flow and to ensure that the fuel related design limits will not be exceeded.

## C.3 Maximum Extended Operating Domain (MEOD)

The modified operating envelope termed MEOD permits extension of operation into additional power/flow areas, provides improved power ascension capability to full power, and additional flow range at rated power. It also includes an increased flow region to compensate for reactivity reduction due to exposure during an operating cycle. Overall, MEOD can be utilized to increase operating flexibility and plant capacity factor. It is a combination of the MELLLA option described in Section C.1 and the ICF option described in Section C.2.

The extended load line region boundary of MEOD is typically limited to 75% core flow at 100% power and the corresponding power/flow constant rod line.

The increased core flow region is bounded approximately by the 105% core flow line and is limited by plant recirculation system capability, acceptable flow-induced vibration, and loading impact on the vessel internal components.

Evaluations performed for MEOD conditions include normal operation and AOOs, LOCA analysis, containment responses, stability, flow induced vibration, and the effects of increased flow induced loads on reactor internal components and fuel channels. The results of the potentially limiting analyses are re-evaluated each cycle to establish the acceptability of the core operating limits.

The analysis of this operating domain extension option is performed using the same methodology and assumptions basis as the standard reload analysis except that the difference in the extreme points or the allowable operating domain are different.

## C.4 Single Loop Operation

Single Loop Operation (SLO) addresses continued plant operation for extended periods with one recirculation loop out of service. Single loop operation at reduced power is highly desirable in the event a recirculation pump or other component maintenance renders one loop inoperative. Due to increased nuclear instrumentation and core flow measurement uncertainties, a more conservative Safety Limit CPR is used for SLO conditions.

To justify SLO, the items listed below must be reviewed and evaluated. These analyses are performed using the standard methodology and bases, except for required model changes to reflect SLO plant conditions.

- MCPR Fuel Cladding Integrity Safety Limit
- MCPR Operating Limit
- Plant Response to a LOCA (MAPLHGR Limit)
- Channel and Fuel Rod Fatigue Loading
- Thermal-Hydraulic Stability

## C.5 End-of-Cycle Coastdown Operation

Typically, the nuclear plant is run throughout its designed cycle lifetime in a full power base load mode of operation or with a combination of base load and load follow operation. When the plant cycle reaches its full power, end-of-cycle (EOC), all control rods out (ARO) condition, the plant then progresses to an orderly shutdown in preparation for the next refueling outage. However, it may be advantageous for the utility to operate the plant beyond this point at reduced power.

In the coastdown mode of operation, the control rods are held in the ARO position and the plant proceeds to coastdown to a lower power while maintaining rated core flow. This is an acceptable mode of operation since it results in increased pressure and thermal margins relative to the existing cycle safety analysis.

[ Proprietary Information Deleted ] If extended coastdown operation is required, this mode of operation will be analyzed using the standard ABB methodology and bases.

## C.6 Safety/Relief Valve(s) Out-Of-Service

The plant safety analyses typically assumes all of the safety/relief valves (SRVs) are operable. If one or more SRVs become inoperable, continued operation for any extended period is not allowed. However, if a proper evaluation and analysis is performed for the limiting transient events, appropriate accident events, and appropriate special events, it can be demonstrated that continued plant operation with one or more SRVs out of service is acceptable.

These analyses demonstrate that this mode of operation maintains complete compliance with all design criteria for AOOs and accidents, including fuel thermal limits, ASME code overpressure requirements, and LOCA peak clad temperatures. These analyses use standard bases and methodology, except one or more SRVs are assumed to be out of service or unable to perform their function.

## C.7 Turbine Bypass Valve Out-Of-Service

Typically, the accident analyses described in the FSAR, and the limiting events evaluated for reload fuel, take credit for turbine steam bypass operation. For most design basis transients, failure of turbine bypass causes the event to be no more severe than the Generator Load Rejection Without Bypass (GLRWOB) or Turbine Trip Without Bypass. (TTWOB) However, one limiting event, feedwater controller failure, would become more severe if turbine bypass failed to function.

In the absence of specific analyses, the Technical Specifications would typically require operation at a substantially reduced power level if turbine bypass is out of service. However, plant- and core-specific analyses can be performed assuming no turbine bypass following a feedwater controller failure (maximum demand) to define the plant operating limits minimum critical power ratio (OLMCPR). In this case, if a failure occurred or maintenance is required in the turbine bypass system, plant operation could continue at a high power level with a more restrictive OLMCPR.

For a utility that has selected the technical specification option for high power operation with turbine bypass inoperable, with ABB fuel in the core, ABB will perform the above analysis to demonstrate that the presence of ABB fuel does not adversely affect operation with turbine bypass inoperable. These analyses use standard bases and methodology except for the assumption of the unavailability of the turbine bypass system.

## C.8 Main Steam Isolation Valve Out-Of-Service

When a main steamline isolation valve (MSIV) is determined to be inoperable, its closure results in one of the four steamlines being out of service. In this situation, the plant may continue operation at approximately 75 percent of rated power and steam flow. However, if a proper safety analysis is performed consisting of the limiting transient events, appropriate accident events and appropriate special events, it is possible to justify continued plant operation with one MSIV out of service at a power level greater than 75 percent.

Reload related transients and special events considered for reanalysis at the higher power level (initially the licensed power level) would typically include:

- Generator Load Rejection without Bypass or Turbine Trip without Bypass
- Loss of Feedwater Heater
- MSIV Closure without Direct Scram

The first two are analyzed to determine the effect on operating limit MCPR, while the third is to address the ASME code requirement for RCS overpressure protection. In addition, the plant LOCA behavior with an isolated steamline would be evaluated.

## C.9 End-Of-Cycle Recirculation Pump Trip Out-Of-Service

For plants which have selected the option which allows the end of cycle (EOC) recirculation pump trip (RPT) to be out of service, the Technical Specifications have been revised to include an additional curve which reflects the MCPR operating limit for this condition. Continued plant operation at rated power conditions is allowed provided that within 1 hour, the MCPR is determined to be equal to or greater than the MCPR limit from this curve times the $K_f$ shown on an accompanying Technical Specification curve.

To show compliance with this requirement, limiting transients such as Generator Load Rejection without Bypass, Turbine Trip without Bypass and Feedwater Controller Failure, all with the EOC-RPT inoperable, are analyzed.

Overpressure protection analysis is not impacted by this option since the MSIV closure without direct scram does not use the EOC-RPT.

## C.10 Degraded Emergency Core Cooling Flow (e.g., LPCI Out Of Service)

Power plant Technical Specifications generally require the plant to be shut down if a low pressure coolant injection (LPCI) system is out of service for a significant period of time (typically 7 days). A utility can potentially continue plant operation under this condition for an extended period of time (typically 30 days), under more restrictive Technical Specification limits. A LPCI system out of service impacts only the loss of coolant accident (LOCA) safety analysis. Hence the LOCA Technical Specification requirements of maximum average planar linear heat generation rate (MAPLHGR) must be reviewed and if necessary revised to reflect the degraded emergency core coolant system (ECCS) capability.

## C.11 Reduced Feedwater Temperature (i.e., Partial Feedwater Heating)

There are two types of conditions under which the utility might wish to operate with lower-than-design feedwater temperature: (a) feedwater heater out-of-service (or some portion of the steam cycle temporarily out-of-service such that normal feedwater heating cannot be maintained); and (b) intentionally valving out feedwater heaters for operation at core burnups beyond standard end-of-cycle. In the absence of specific evaluations, plant operation at feedwater temperatures significantly below normal are typically precluded by the plant license.

Analyses are performed in order to justify operation at a reduced feedwater temperature. Usually, the analyses are performed for EOC operation with the last stage feedwater heaters valved out of service. However, for operation throughout the cycle, an additional feedwater temperature reduction can be justified by additional analyses at the appropriate operating conditions.

The selection of limiting transients analyzed is dependent on the time in the cycle at which the operating flexibility is desired. The transients analyzed include:

- Generator Load Rejection Without Bypass (GLRWOB)
- Feedwater Controller Failure with Maximum Demand (FWCF)
- Feedwater Temperature Reduction
- Rod Withdrawal Error
- Overpressurization Analysis (MSIV closure without direct scram)

These analyses will be performed using the standard methodology and assumption bases.

## C.12 Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS)

Some utilities have elected to implement the ARTS program to enable more effective use of existing margins for operating flexibility. This program provides the following major improvements:

1) Power dependent MCPR and MAPLHGR limits and flow dependent MAPLHGR limits are incorporated into the Technical Specifications and the APRM setdown requirement is removed.

2) The Rod Block Monitor System hardware and design are improved, and analysis of the rod withdrawal event is improved by use of a generic statistical analysis which allows bypassing the RBM under certain circumstances when large margins are available.

In addition to the Rod Withdrawal Error re-analysis, various safety analyses are required to establish or verify the revised set-points and operational limits which accompany ARTS.

1) Flow-dependent APRM rod block line

2) $MCPR_F$ curve(s)

3) $MAPFAC_F$ curves

4) $K_p$ curve

5) $MAPFAC_P$

These analyses are performed using the standard methodology and assumptions bases which include the simulation of the hardware modifications.

## C.13 Extended Operating State Dependence On Operating Limit MCPR

For BWRs, the end-of-cycle all-rods-out condition gives the worst scram response. If multiple transient analyses are performed at mid-cycle exposure points, the mid-cycle limits will yield significantly more MCPR margin. Except for the division of the cycle into a series of intervals, the mid-cycle analyses do not alter the reload analysis methodology. [ Proprietary Information Deleted ]

# APPENDIX D: RELOAD METHODOLOGY SAMPLE APPLICATIONS

## D.1 DESCRIPTION OF RELOAD EXAMPLES

Illustration of the ABB reload fuel methodology is given throughout this document by presenting sample applications. Several different application examples are used in this appendix. Example(s) are chosen to best communicate the general methodology being described for each discipline. The following sections summarize the four reload application examples used throughout the appendix.

### D.1.1 Description of Reload Example 1

[ Proprietary Information Deleted ]

### D.1.2 Description of Reload Example 2

[ Proprietary Information Deleted ]

### D.1.3 Description of Reload Example 3

[ Proprietary Information Deleted ]

### D.1.4 Description of Reload Example 4

[ Proprietary Information Deleted ]

**TABLE D.1-1 THROUGH TABLE D.1-4**

Proprietary Information Deleted

## D.2 MECHANICAL DESIGN

### D.2.1 SVEA-96 Mechanical Design Description

A complete description of the SVEA-96 design is provided in Reference 37. This section contains a description of the SVEA-96 design appropriate for a C-lattice plant with an active fuel length of 3810 mm. This mechanical design would be suitable for the sample applications described in Section D.1.

The primary objective of the SVEA design is integrity and reliability of the fuel rod and assembly. To this end, numerous features have been adopted with the goal of achieving zero fuel rod failures during reactor operation. [ Proprietary Information Deleted ]

#### D.2.1.1 Assembly Description

The SVEA-96 fuel assembly was designed for U.S. domestic BWRs. The SVEA-96 fuel assembly consists of three basic components:

- The fuel bundle,
- The fuel channel, and
- The handle.

Figure D.2.1-1 shows the SVEA-96 assembly.

The fuel bundle consists of 96 fuel rods arranged in four 5x5 minus 1 (5x5-1) subbundles. The channel has a cruciform internal structure with a square center channel that forms gaps for non-boiling water during normal operation. The subbundles are inserted into the channel from the top and [ Proprietary Information Deleted ] This design principle has been used in various ABB BWR fuel assembly designs for many years, and eliminates the leakage flow path at the bottom end of the channel. This design feature also avoids stresses in the tie rods during normal fuel handling operations. The fuel assembly is lifted with a handle connected to the top end of the channel.

The subbundles are freestanding inside the channel. There is sufficient space for subbundle growth at the top of the assembly to avoid restriction due to differential growth between the fuel bundles and the channel.

The bottom of the transition piece, or "nose piece," seats in the fuel support piece. The top ends of fuel assemblies are supported laterally against the adjacent assemblies through the interaction of leaf springs on two sides and the upper core grid on the other two sides. Compatibility evaluations and operating experience have confirmed the mechanical compatibility of the SVEA-96 assembly with U.S. and European reactors and several existing fuel types.

Handle with Spring

Figure D.2.1-6 shows the SVEA-96 handle and leaf spring design. The handle and leaf spring configuration are fitted to the top end of the channel. [ Proprietary Information Deleted ]

The handle is equipped with a double leaf spring which maintains contact with the corresponding springs on adjacent assemblies and firmly presses the fuel assembly into the corner of the upper core grid.

Lattice and Fuel Rod Types

Each subbundle is a 5x5-1 lattice. The fuel assembly has [ Proprietary Information Deleted ]

### D.2.1.2 Fuel Subbundle Description

The fuel subbundle designs are shown in Figure D.2.1-4. Each subbundle is a separate unit with top and bottom tie plates. [ Proprietary Information Deleted ]

The tie rods are connected to the top and bottom tie plates with threaded end plugs extending through the plates and secured by nuts. [ Proprietary Information Deleted ]

Top and Bottom Tie Plates

The top tie plates are [ Proprietary Information Deleted ]

ABB has accumulated extensive in-reactor experience with these basic tie plate designs. [ Proprietary Information Deleted ]

Standard Fuel Rods, Tie Rods, and Spacer Capture Rods

A standard fuel rod is shown in Figure D.2.1-7. The tie rod is shown in Figure D.2.1-8.

The fuel consists of $UO_2$ or, in case of Burnable Absorber (BA) rods, $UO_2$-$Gd_2O_3$ ceramic pellets. The pellets are contained in Zircaloy-2 cladding tubes which are plugged and welded at the ends to encapsulate the uranium fuel. [ Proprietary Information Deleted ]

The two tie rods are identical to the standard rods with the exception of the top and bottom end plugs. These rods are structural members of the fuel assembly, and establish the overall subbundle length. [ Proprietary Information Deleted ]

The spacer capture fuel rods are shown in Figure D.2.1-9. A spacer capture rod is [ Proprietary Information Deleted ]

Pellets

The pellet for SVEA-96 is designed to [ Proprietary Information Deleted ] A sketch of the enriched fuel pellet is shown in Figure D.2.1-10.

The pellet sintering process is designed to minimize in-pile fuel pellet densification. [ Proprietary Information Deleted ]

Fuel pellets with burnable absorber (BA) consist of mixed $Gd_2O_3$ and uranium oxide powders. [ Proprietary Information Deleted ]

Spacers

The spacer is shown in Figure D.2.1-11. The spacer grid is a [ Proprietary Information Deleted ]

The spacer grid is designed for [ Proprietary Information Deleted ] and to withstand all dynamic loads encountered during reactor operation. The spacers provide lateral support for the fuel rods and minimize rod vibrations and axial loads that could lead to rod bowing. The spacers must also maintain sufficient space between fuel rods and between the rods and the channel to assure that thermal-hydraulic conditions are not compromised during reactor operations.

The spacer design is well proven. The basic design was used originally for the ABB 8x8 assemblies. It is currently used for the SVEA-100 and SVEA-96 designs. Extensive in-reactor experience has not revealed any evidence of stress corrosion cracking, and has demonstrated that the spacers satisfactorily provide their intended function to high burnups. Mechanical testing has confirmed that the spacer functions as designed under loading associated with accident conditions.

### D.2.1.3 SVEA-96 Fuel Channel

The Zircaloy-4 channel consists of a square outer channel with a double-walled internal cross structure which forms channels for non-boiling water. Cross sections are shown in Figure D.2.1-2 and Figure D.2.1-5. The inner, cross channel (or "watercross") has a square central water channel and smaller water channels in each of the four wings. [ Proprietary Information Deleted ] The outer channel and the watercross structure form four subchannels for the subbundles.

[ Proprietary Information Deleted ]

In addition to providing channels for non-boiling water, the integral watercross design results in improved dimensional stability leading to reduced bow and bulge of the channels.

Screws in each of the four sides of the assembly secure the outer channel and the transition piece to the bottom support plate. The transition piece fits into the fuel support piece. [ Proprietary Information Deleted ] The channel and inlet transition piece are designed for compatibility with the reactor internals as well as other fuel types in the core. The outer envelope of the SVEA-96 channel and transition piece provide ample clearance for control rods and in-core instrumentation. The dimensional stability of the SVEA channel assures that ample clearances are maintained with burnup. The length of the assembly is compatible with the relative positions of the fuel support piece and upper core grid.

## D.2.2 Methodology for Mechanical Design Input to Reload Design and Safety Analysis

### D.2.2.1 Mechanical Design Input to Reload Design and Safety Analysis

Dimensions, materials properties, and material compositions required for the Reload Design and Safety Analyses are provided as part of the interface between the mechanical design and the nuclear and thermal-hydraulic design and transient and accident analyses. An example of this mechanical design input can be found in Section 5 of Reference 37.

Fuel rod operating limits which assure that thermal-mechanical fuel rod design criteria are satisfied are also provided as part of the mechanical design input to the Reload Design and Safety Analyses. [ Proprietary Information Deleted ]

### D.2.2.2 Fuel Rod Thermal Performance Input to Transient Analyses

[ Proprietary Information Deleted ]

### D.2.2.3 Fuel Rod Thermal Performance Input to LOCA Analyses

[ Proprietary Information Deleted ]

### D.2.2.4 Fuel Rod Thermal Performance Input to CRDA Analyses

An example of a calculation of pellet cladding gap heat transfer coefficient is shown in Section A.3 of Part II of Reference 33 and is not, therefore, repeated in this document.

**FIGURE D.2.1-1 THROUGH FIGURE D.2.2-1**

Proprietary Information Deleted

## D.3 NUCLEAR DESIGN

This section contains an example of the application of the nuclear design methodology discussed in Section 4.3. Reload Example 1 is used for this sample application.

### D.3.1 Reference Reload Core Description

The cycle in Reload Example 1 is referred to as Cycle A in this document. The composition of Cycle A is shown in Table D.1-1. [ Proprietary Information Deleted ]

### D.3.2 Sample SVEA-96 Bundle Description and Characteristics

#### D.3.2.1 General SVEA-96 Nuclear Characteristics

This section summarizes the impact which some of the SVEA-96 characteristics have on the nuclear design. It is intended as a general discussion to familiarize the reader with the SVEA-96 nuclear design features. Section D.3.2.1 provides specific results for the SC SVEA-96 bundle.

##### D.3.2.1.1 Watercross

The watercross delivers more non-boiling water to the central parts of the assembly than traditional designs utilizing water rods. The resulting improvement in neutron moderation tends to increase the reactivity of the bundle under hot, operating conditions for the same average enrichment. [ Proprietary Information Deleted ]

##### D.3.2.1.2 Increased Number of Fuel Rods

The relatively large number of fuel rods in the SVEA-96 assembly translates into lower linear heat generation rates (LHGRs) than assembly designs with fewer fuel rods for the same bundle power. [ Proprietary Information Deleted ]

##### D.3.2.1.3 Axial Grading of $Gd_2O_3$ and $U^{235}$

Axially graded $Gd_2O_3$ was pioneered by ABB. This concept has been utilized by ABB extensively since 1976 in 8x8 applications and has been applied, when necessary, in the SVEA design. [ Proprietary Information Deleted ]

##### D.3.2.1.4 Nuclear Compatibility with Other Fuel Types

[ Proprietary Information Deleted ]

Therefore, numerous reload and LFA applications of SVEA-96 to-date have demonstrated that the inherent flexibility of the SVEA-96 nuclear characteristics.

### D.3.2.2 SVEA-96 Bundle Type SC

Since the type SC SVEA bundle represents the dominant fuel type in Cycle A, its description and characteristics are provided as an example in this section.

[ Proprietary Information Deleted ]

## D.3.3 Reference Reload Core Three-Dimensional Results

[ Proprietary Information Deleted ]

## D.3.4 Sample Nuclear Design Input to Reload Design and Safety Analyses

### D.3.4.1 Nuclear Design Input to Mechanical Design

An example of fuel rod power histories computed with the nuclear design code system for the fuel rod mechanical analysis is provided in Reference 37 and is, therefore, not repeated here.

### D.3.4.2 Nuclear Design Input to Thermal-Hydraulic Design

[ Proprietary Information Deleted ]

### D.3.4.3 Nuclear Design Input to Transient Analyses

[ Proprietary Information Deleted ]

### D.3.4.4 Nuclear Design Input to LOCA Analyses

[ Proprietary Information Deleted ]

### D.3.4.5 Nuclear Design Input to CRDA Analyses

A complete example of a CRDA evaluation including the nuclear input is provided in Reference 33 and is, therefore, not repeated here.

### D.3.4.6 Nuclear Design Input to Refueling Accident Analyses

A generic evaluation demonstrating that the consequences of the Fuel Handling Accident are less limiting for the SVEA-96 assembly than for an 8x8-2 fuel assembly is provided in Section D.7. Therefore, fission product data specific to Cycle A and the SC bundle were not required in this case.

## FIGURE D.3.1-1 THROUGH FIGURE D.3.3-2

Proprietary Information Deleted

# D.4 THERMAL-HYDRAULIC DESIGN

## D.4.1 SVEA-96 Thermal Hydraulic Characteristics

The current ABB reload fuel design for U.S. BWRs is the SVEA-96 fuel assembly. This is the "Reload" fuel assembly referred to in Sections 5.2 through 5.4. Therefore, this appendix gives a description of the thermal-hydraulic characteristics of the SVEA-96 fuel assembly. The mechanical design of the assembly is discussed in Reference 37 and summarized in Section 3 and Appendix D.2.

The important thermal-hydraulic design features of the SVEA-96 assembly can be summarized as follows:

[ Proprietary Information Deleted ]

### D.4.1.1 General Description

The primary parameters effecting the thermal-hydraulic performance of the SVEA-96 assemblies are:

[ Proprietary Information Deleted ]

## D.4.2 Hydraulic Modeling of Resident Fuel

[ Proprietary Information Deleted ]

## D.4.3 MCPR Modeling of Resident Fuel

[ Proprietary Information Deleted ]

## D.4.4 Sample SLMCPR Evaluation

[ Proprietary Information Deleted ]

## D.4.5 Hydraulic Compatibility in Mixed Cores

[ Proprietary Information Deleted ]

### D.4.5.1 Sample Application - Reload Example 1 in Appendix D.1

[ Proprietary Information Deleted ]

### D.4.5.2 Sample Application - Reload Example 4 in Appendix D.1

[ Proprietary Information Deleted ]

### D.4.6 Bypass, Water Rod, and Watercross Flow

[ Proprietary Information Deleted ]

### D.4.7 Input Data to Reload Design and Safety Analysis

[ Proprietary Information Deleted ]

**TABLE D.4.2-1 THROUGH TABLE D.4.5.2-1**

Proprietary Information Deleted

**FIGURE D.4.2-1 THROUGH FIGURE D.4.6-2**

Proprietary Information Deleted

## D.5 RELOAD SAFETY ANALYSIS

This section establishes the bases for the reload safety evaluations performed for the sample application. The safety analysis methodology used in this sample application is described in Section 6.

### D.5.1 Plant Licensing Basis for Reload

The plant licensing basis for a specific reload is derived from safety analysis for the plant being evaluated. From an overall perspective, the plant safety analysis contains an analysis of the overall plant design and performance to determine the margin of safety during normal plant operation and transient conditions expected during the plant lifetime (anticipated operational occurrences) and demonstrates the adequacy of the plant design for the prevention of accidents and the mitigation of their consequences, should they occur. The plant safety analysis also contains the results of other analyses evaluated to demonstrate the plant capability to respond to selected events, performed in response to regulatory requirements and guidance and to specific licensing commitments. In addition, the plant specific safety analysis may contain specific licensing commitments that are impacted by the proposed reload fuel application.

[ Proprietary Information Deleted ]

### D.5.2 Plant Operating Flexibility Options

As described in Appendix C, there are a number of plant flexibility options that may be implemented on a specific plant to enhance the plant operational flexibility. These plant flexibility options can impact the analyses performed for a plant specific reload. [ Proprietary Information Deleted ]

# D.6 TRANSIENT ANALYSES

## D.6.1 Transient Evaluations

### D.6.1.1 Generator Load Rejection

A sample application of the methodology for evaluation of the Generator Load Rejection is provided below for Reload Example 1 in Appendix D.1.

Event Description

The GLR Without Bypass (referred to as the GLRWOB or GLRNB) transient event is the postulated complete loss of electrical load to the turbine generator coupled with the assumed failure of the turbine bypass system. Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which results in significant loss of electrical load on the generator. The turbine control valves are required to close rapidly to prevent excessive overspeed of the turbine generator. The rapid closure of the turbine control valves causes a sudden reduction in steam flow that results in a system pressure increase. Neutron flux increases rapidly because of the core void reduction caused by the pressure increase. Turbine control valve fast closure initiates a scram trip signal and a Recirculation Pump Trip (RPT), which results in a rapid reactor shutdown. The reactor vessel pressure increase is limited by the action of the relief valves. The neutron flux increase is limited by the scram and the RPT. The peak fuel surface heat flux increases initially due to the neutron flux increase then decreases following reactor shutdown. Long term reactor water makeup is provided by the feedwater system or high pressure makeup systems. Heat rejection is through the relief valves to the suppression pool.

Event Conditions

[ Proprietary Information Deleted ]

Typical Results

[ Proprietary Information Deleted ]

### D.6.1.2 Feedwater Controller Failure

Sample applications of the methodology for evaluation of the Feedwater Controller Failure is provided below for Reload Example 1 in Appendix D.1.

Event Description

The Feedwater Controller Failure (FWCF) transient event is initiated by the failure of a control device which results in the feedwater controller being forced to its upper limit which creates the maximum feedwater system flow demand. The increased feedwater flow mixes with the recirculation flow and results in a gradual increase in core inlet subcooling. The increased feedwater flow also results in an increase in reactor vessel water level. The gradual increase in core inlet subcooling causes a relatively slow power increase and a shift in power distribution towards the bottom of the core. As a result of the power increase, the vessel steam flow increases which creates a slight increase in system pressure due to the larger steam line pressure drops as the pressure regulator system controls the turbine inlet pressure. The power increase continues until the reactor vessel high water level trip setpoint (L8) is reached.

High reactor vessel water level initiates closure of the main turbine stop valves (turbine trip) and a trip of the feedwater system. Closure of the turbine stop valves initiates a reactor scram, a bypass valve opening signal, and RPT. Following the turbine trip, the neutron flux increase is limited by the reactor scram and the RPT. The peak neutron flux and surface heat flux are reached following the turbine trip. The relief valves are opened in the pressure relief mode and close sequentially as the pressure is reduced by the action of the relief valves and the turbine bypass valves.

This event may be used to determine the power dependence of the operating limits; therefore, additional lower power cases may be performed.

Event Conditions

[ Proprietary Information Deleted ]

Typical Results

[ Proprietary Information Deleted ]

### D.6.1.3 Turbine Trip

A sample application of the methodology for evaluation of the Turbine Trip (TT) is provided below for Reload Example 1 in Appendix D.1.

Event Description

A variety of turbine or nuclear system malfunctions can initiate a turbine trip (closure of the turbine stop valves). Some examples are: turbine vibrations, low condenser vacuum, high levels in the moisture separator or heater drain tank, operator lock-out, and reactor high water level. The sudden reduction in steam flow caused by the closure of the turbine stop valves results in an increase in system pressure. This event can occur with or without proper operation of the turbine bypass valves. The proper functioning of the turbine bypass valves reduces the increase in system pressure. The transient without bypass is more severe since failure of the turbine bypass valves is assumed for the entire transient and the benefit of their mitigation of increasing system pressure is lost.

[ Proprietary Information Deleted ]

### D.6.1.4 Pressure Regulator Failure - Closed (BWR/6 Only)

Since Reload Example 1 in Appendix D.1 is for a BWR/5, the pressure regulator failure - closed transient is not presented in this report.

### D.6.1.5 Recirculation Flow Controller Failure - Increasing Flow

A sample application of the methodology for evaluation of the Recirculation Flow Controller Failure is provided for Reload Example 1 in Appendix D.1. [ Proprietary Information Deleted ]

### D.6.1.6 Rod Withdrawal Error

A sample application of the methodology for evaluation of the Control Rod Withdrawal Error is provided for Reload Example 1 in Appendix D.1. [ Proprietary Information Deleted ]

### D.6.1.7 Loss of Feedwater Heating

A sample application of the methodology for evaluation of the Loss of Feedwater Heating event is provided for Reload Example 1 described in Appendix D.1. [ Proprietary Information Deleted ]

### D.6.1.8 Burnup Dependence of Results

[ Proprietary Information Deleted ]

### D.6.1.9 Power and Flow Dependence of Results

A sample application of the methodology for evaluation of the power and flow dependence of the operating limit results is provided below for Reload Example 1 in Appendix D.1.

[ Proprietary Information Deleted ]

Full Power / Full Flow Results

[ Proprietary Information Deleted ]

Flow Dependent Results

[ Proprietary Information Deleted ]

Power Dependent Results

[ Proprietary Information Deleted ]

### D.6.2 MCPR Operating Limit for Reload Example 1

#### D.6.2.1 Treatment of Analysis Uncertainties

[ Proprietary Information Deleted ]

#### D.6.2.2 Reload MCPR Operating Limit

[ Proprietary Information Deleted ]

**TABLE D.6.1.1-1 THROUGH TABLE D.6.2-1**

Proprietary Information Deleted

**FIGURE D.6.1.1-1 THROUGH FIGURE D.6..2-2**

Proprietary Information Deleted

# D.7 ACCIDENT ANALYSIS

## D.7.1 Loss of Coolant Accident

An example is presented of the plant specific LOCA analysis for the SVEA-96 fuel assembly introduced in Reload Example 1.

### D.7.1.1 Limiting LOCA Design Basis Event

For the BWR/5 plant in Reload Example 1, the design basis LOCA event in the plant safety analysis is a full recirculation suction line break with failure of the low pressure core spray diesel generator. ABB Evaluation Model sensitivity studies for this 764-assembly BWR/5 plant design (Reference 22) have confirmed that this LOCA event is limiting. For the example presented below, it has been assumed that specific evaluation of the Reload Example 1 BWR/5 plant, would confirm that the same design basis event is limiting.

### D.7.1.2 Design Basis Event Analysis

The LOCA analysis example presented here used the approved ABB LOCA Evaluation Model described in Appendix A.4.3 with supplemental features currently under NRC review. [ Proprietary Information Deleted ]

The LOCA system pressure response to a design base event is shown in Figure D.7.1-1. The detailed discussion of the transient response characteristics is given in Section 3.2 of Reference 22. The hot assembly flow and mass inventory response are shown in Figures D.7.1-2 and D.7.1-3. The corresponding fuel rod heat transfer coefficients in the hot assembly peak plane prescribed by the evaluation model are shown in Figure D.7.1-4. The timing of flow regime changes is determined for the hot assembly response. The peak cladding temperature response at 20,000 MWd/MTU is shown in Figure D.7.1-5. The resultant peak cladding temperature and maximum local oxidation throughout the fuel assembly life is shown in Table D.7.1-1.

### D.7.1.3 MAPLHGR Operating Limit

The MAPLHGR limit established by the LOCA analysis is shown in Table D.7.1-1. [ Proprietary Information Deleted ]

### D.7.1.4 Total Hydrogen Generation

A sample maximum hydrogen generation calculation has been performed for the limiting LOCA transient described above. Hydrogen generation calculations were performed using conservative nuclear and fuel performance data and using data corresponding to the Reference Core of Reload Example 1.

[ Proprietary Information Deleted ]

### D.7.1.5 Assessment of Methodology Conservatism

An uncertainty evaluation was performed to quantify the inherent conservatism in the ABB LOCA evaluation methodology with models conforming to 10CFR50 Appendix K. [ Proprietary Information Deleted ]

### D.7.2 Control Rod Drop Accident

An example of the control rod drop accident analysis has been provided with the methodology description in the Licensing Topical Report CENPD-284-P-A (Reference 33).

### D.7.3 Fuel Handling Accident

The methodology for determining the impact of a fuel handling accident on a core containing ABB fuel is described in Section 8.4. A sample application of the methodology for evaluation of the Fuel Handling Accident is provided for Reload Example 1 described in Appendix D.1. The existing analysis is for a 764 assembly BWR/5 with 8x8-2 reference fuel.

[ Proprietary Information Deleted ]

### D.7.4 Mislocated Assembly

A sample application of the methodology for evaluation of the Mislocated Assembly is provided for Reload Example 1 in Appendix D.1. The feed and second-cycle (once-burned) bundles in this example are the SVEA-96 SC bundles loaded in the subject cycle ("Cycle A") and the previous cycle. Therefore, the MCPR performance during the accident is representative of SVEA-96 fuel.

The analysis was performed with the nuclear design code system described in Appendix A as follows:

[ Proprietary Information Deleted ]

### D.7.5 Misoriented Fuel Assembly

A sample application of the methodology for evaluation of the Misoriented Assembly is provided for Reload Example 1 in Appendix D.1. [ Proprietary Information Deleted ]

**TABLE D.7.1-1 THROUGH TABLE D.7.5-1**

Proprietary Information Deleted

**FIGURE D.7.1-1 THROUGH FIGURE D.7.5-1**

Proprietary Information Deleted

# D.8 SPECIAL EVENTS ANALYSIS

## D.8.1 Stability

### D.8.1.1 Sample Application for Reload Example 1

The Example 1 plant is assumed to have licensing commitments to perform stability analyses for each reload application. The plant specific reload safety evaluation includes a stability licensing evaluation in accordance with the methodology described in Reference 45. An example of the reload stability licensing evaluation is also provided in Reference 45.

### D.8.1.2 Sample Application for Reload Example 2

A reload specific stability evaluation is not always required. The Example 2 plant is assumed to be in compliance with NRC Bulletin 88-07 and Supplement 1 (Reference 48). ABB evaluated the operating limitations in NRC Bulletin 88-07 and Supplement 1 and determined that they are applicable to the ABB reload fuel design. The ABB reload fuel design was demonstrated to have improved stability performance relative to resident fuel designs in previous reload cycle applications. Hence, the current reload application does not require an explicit stability safety analysis calculation.

## D.8.2 Overpressurization Protection

### D.8.2.1 Sample Application for Reload Example 1

Sample applications of the methodology for evaluation of the Overpressurization Protection is provided below for Reload Example 1 in Appendix D.1.

Event Description

The Overpressure Protection transient event is initiated by the closure of all main steam line isolation valves (MSIVs). The closure of the MSIVs would normally result in a scram signal due to MSIV position; however, in this analysis, the direct scram is disabled due to the conservative approach to this analysis.

Closure of all MSIVs causes a rapid reduction in steam flow which results in a system pressure increase. Neutron flux increases rapidly because of the core moderator void reduction caused by the pressure increase. The pressure increase is limited by the opening of the safety/relief valves and the reactor scram that is initiated by the average power range monitor (APRM) high neutron flux signal.

Event Conditions

[ Proprietary Information Deleted ]

Typical Results

[ Proprietary Information Deleted ]

### D.8.3 Standby Liquid Control System

A sample application of the methodology for evaluation of the Standby Liquid Control System (SLCS) is provided for Reload Example 1 in Appendix D.1. [ Proprietary Information Deleted ]

### D.8.4 ATWS Evaluation

#### D.8.4.1 Evaluation for SVEA-96 Fuel

[ Proprietary Information Deleted ]

#### D.8.4.2 Sample Application - Reload Example 1

[ Proprietary Information Deleted ]

#### D.8.4.3 Sensitivity to Reload Design with SVEA-96 Fuel

[ Proprietary Information Deleted ]

## TABLE D.8.2.1-1 THROUGH TABLE D.8.3-1

Proprietary Information Deleted

## FIGURE D.8.2.1-1 THROUGH FIGURE D.8.4-2

Proprietary Information Deleted

# APPENDIX E: FAST PRESSURIZATION TRANSIENT ANALYSIS QUALIFICATION

## E.1 Introduction

An extensive qualification of the fast transient analysis code BISON, has been given in two Licensing Topical Reports (Reference 23 and 39). The information presented in this appendix supplements the referenced qualification with additional benchmark comparisons directly supporting the fast transient methodology for reload applications described in Section 7.3 and 7.4.

[ Proprietary Information Deleted ]

## E.2 Peach Bottom 2 Turbine Trip Tests

The qualification of the BISON code system discussed in Appendix A includes comparisons with the Peach Bottom 2 turbine trip tests. The Peach Bottom 2 turbine trip tests are documented in References 62 and 63 and discussed in Section 3, Volume 2 of Reference 23. Simulations of the turbine trip tests are presented in Section 3, Volume 2 of Reference 23 and Section 6.5 of Reference 39. The test simulation and sensitivity study results presented below are a direct extension of the calculations presented in Reference 39.

## E.3 Transient Axial Power Distribution

[ Proprietary Information Deleted ]

### E.3.1 Simulation Method

[ Proprietary Information Deleted ]

### E.3.2 Transient Axial Power Distribution Qualification

[ Proprietary Information Deleted ]

## E.4 BISON Fast Pressurization Transient Analysis Bias and Uncertainty

[ Proprietary Information Deleted ]

### E.4.1 Peach Bottom 2 Turbine Trip Simulation

[ Proprietary Information Deleted ]

### E.4.2 Peach Bottom 2 Turbine Trip 1 Uncertainty Evaluation

[ Proprietary Information Deleted ]

## TABLE E-1 THROUGH TABLE E-3

Proprietary Information Deleted

## FIGURE E.3-1 THROUGH FIGURE E.3-5

Proprietary Information Deleted

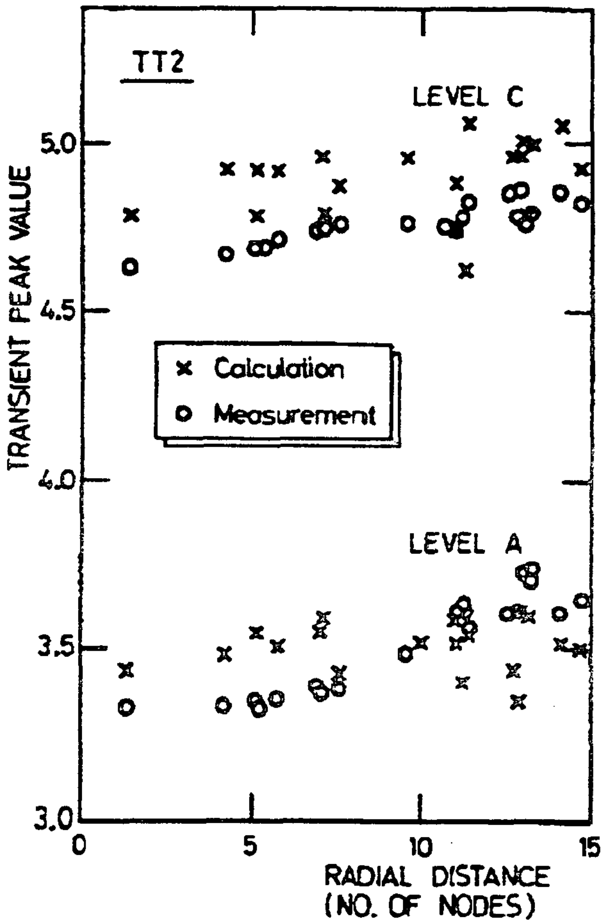


Figure E.3-6 Peach Bottom 2 Turbine Trip 2 level A and C LPRM Response and 3D Study Results from Reference E-5.

# APPENDIX F: RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

## F.1 Introduction

This appendix contains responses to the NRC Request for Additional Information regarding Reference F1 which was transmitted to ABB by the NRC letter identified in Reference F2.

The original submittal of Licensing Topical Report, CENPD-300-P (Reference F1), contained parts that were identified as Combustion Engineering, Inc., Proprietary Information. Several paragraphs of Reference F1 are now public information and are identified as such in this report.

Typographical errors have been identified in the original submittal of Licensing Topical Report, CENPD-300-P (Reference F1). The corrections have been made to the main body of this report.

## F.2 Questions and Responses

NRC Question F1

*Discuss the objectives of the Reference Safety Analysis Report and the manners in which approval of this topical report will be used in future reload.*

ABB Response to Question F1

The objective of this report, also referred to as the Reference Safety Report (RSR), is to obtain generic Nuclear Regulatory Commission (NRC) approval for the ABB reload fuel design and safety analysis process that utilizes the ABB reload fuel design and analysis codes. The RSR describes the application of the methodology that is used in the reload fuel safety analysis process and in the evaluation of plant modifications requiring updating of fuel and core related safety analyses (e.g., changes to the plant operating domain or equipment performance characteristics). The specific ABB reload fuel design and analysis code methods and methodology have been independently submitted to the NRC for review and approval and are not considered a part of the approval of this RSR. However, the RSR is based on the use of NRC approved analysis codes methods and methodology, as described in the reference licensing topical reports. Thus, the RSR is a comprehensive reference document that describes the application of the NRC approved ABB reload fuel design and analysis codes in the safety analysis process. Further, the methodology described in the RSR will be continuously improved by updating specific methodology references as they are approved for application in the safety analysis process.

It is intended that the RSR be applied consistent with the current plant license basis and the requirements of 10CFR50.59 for plant modifications, including the plant modification associated with the introduction of reload fuel and its operation in a new core configuration. If it is determined that the plant modification results in an unreviewed safety question, a license amendment request is submitted by the licensee in accordance with 10CFR50.90. When used as a reference in a license amendment request, the generic information contained in the RSR does not require additional NRC review, saving both NRC and licensee resources. Therefore, only the results of the analyses will require review and approval. If it is determined that the plant modification does not involve an unreviewed safety question, the application of the approved methodology provides additional assurance that the safety evaluation for the change is acceptable.

It is important to recognize that ABB uses the current plant license basis as an inherent part of the process for updating the plant safety analysis. By using the current plant license basis, the unique safety analysis requirements for specific plants are captured in the analysis process. Therefore, it is not necessary to identify the differences between specific plants in the application of the ABB methodology, because these differences are contained in the current plant licensing basis.

A standard reload of ABB reload fuel is a typical plant change that would be expected not to involve an unreviewed safety question. For this case, the application of the RSR methodology would be used to update the Core Operating Limits Report (COLR), that establishes the operating limits for the operating cycle. The analysis results would be included in the reload safety analysis summary report that is used as the primary basis for the safety evaluation required by 10CFR50.59.

Due to the unique objectives of an applications licensing topical report that covers the scope of the RSR, a wide spectrum of information must be captured. This range of information is necessary to cover the entire reactor fuel design and safety analysis process. The information contained in the RSR can be separated into five categories that reflect different aspects of the process: (1) the identification of the NRC approved fuel design and safety analysis codes and methods that are used in the analysis process; (2) the analysis process itself; including the transfer of information between the various approved codes and methods; (3) the identification of the potentially limiting events and the quantification of the event acceptance limits for these events; (4) the development of analysis inputs and treatment of analysis outputs to assure that the safety analysis process is adequately conservative; and (5) the development of the core operating limits that constrain plant operation. Each of these categories of information is described in more detail below.

Category 1 information is derived from NRC approved codes and methods. There are a number of different design and safety analysis codes and methods utilized in the fuel design and safety analysis methods. The codes and methods address specific analysis requirements associated with fuel design and analyses. Because of the complexity of these methods, they are the subject of individual Licensing Topical Reports. These Licensing Topical Reports are reviewed and approved for use by the NRC before they are applied in the safety analysis process. The RSR identifies the specific Licensing Topical Reports that document the methodology used in the fuel design and safety analysis process. The information included in the RSR and associated with the approved Licensing Topical Report basically summarizes the information contained in the applicable report. It is information that is generally covered in the review of the individual Licensing Topical Reports. This information is considered to be previously approved with the Safety Evaluation Reports issued on the individual Licensing Topical Reports.

Category 2 information is derived from the overall fuel design and safety analysis process employed by ABB. This information describes the analysis process and the various codes and methods used in each step of the process. Additionally, the required analysis inputs are identified along with the information that is required to be transferred between the various analysis steps. This type of information describes how the approved codes and methods are utilized in the process and what information is transferred as a part of the process. This is general process information that requires NRC approval as part of the review of the RSR.

Category 3 information is associated with the identification of the potentially limiting events and the quantification of the event acceptance limits for these events. The potentially limiting events are derived through a combination of generic ABB event analyses and an assessment of current plant licensing bases. The events identified as potentially limiting in the ABB methodology are consistent with current industry practice. In identifying the potentially limiting events, a key consideration is the event acceptance limits. Therefore, the event acceptance limits applied to each of the potentially limiting events is considered to be a part of this category of information. The approach taken by ABB in the identification of the potentially limiting events and their corresponding event acceptance limits requires NRC approval as part of the review of the RSR.

Category 4 information is an inherent part of the development of analysis inputs and treatment of analysis outputs to assure that the safety analysis process is adequately conservative. To assure that the analysis results are adequately conservative, conservatism may be introduced through the use of conservative inputs or through the use of best-estimate inputs with conservatism applied to the analysis results (e.g., an uncertainty evaluation). Further, the conservative factors applied to the analysis results may be obtained through either a deterministic process, statistical analysis process, or a combination of both processes. The processes for treating analysis inputs and outputs requires NRC approval as part of the review of the RSR.

Category 5 information is associated with the development of the core operating limits that constrain plant operation. Core operating limits are derived from the results of the fuel safety analysis process. Safety analysis results either demonstrate that there is an acceptably low probability that the applicable event acceptance limits will not be exceeded for events initiated from the core operating limits (e.g., code overpressure protection analysis) or are used as part of the process to derive the appropriate core operating limits (e.g., the use of the limiting anticipated operational occurrence change in critical power ratio in establishing the operating limit minimum critical power ratio). The process for establishing the core operating limits requires NRC approval as part of the review of the RSR.

ABB has reviewed each of the sections of the RSR with respect to each of the categories. The results of this review are summarized in Table F1-1.

NRC Question F2

*In discussion of mechanical design data obtained from ABB and other vendors, there is no mention of design uncertainties. Discuss how these uncertainties are integrated into the nuclear design and analyses and what data are available. Identify the bounding conditions and discuss how these conditions are determined (§ 3.4).*

ABB Response to Question F2

Design uncertainties are accounted for in the mechanical design evaluation as well as in the derivation of operating limits. Two basic approaches are utilized. The first is to accommodate uncertainties by utilizing bounding values based on design uncertainties to obtain conservative results. The second approach is to perform a base analysis with nominal values and to add an overall uncertainty to the result which conservatively accommodates contributing uncertainties. The contributing uncertainties include all parameters which would significantly effect the result.

Table F2-1 summarizes the application of key uncertainties in mechanical design data to various licensing analyses and provides references to more detailed discussions. Specific use of bounding values or uncertainty evaluations to obtain a conservative result are provided in the references given in Table F2-1. This issue is also addressed in the response to Question F17.

The nuclear design evaluation is performed with nominal values. The results of the nuclear design evaluation are used in conjunction with appropriate uncertainties to establish operating limits and to evaluate the performance of the nuclear design relative to those limits. The impact of mechanical design uncertainties on the nuclear design parameters are incorporated in power uncertainties in mechanical design and SLMCPR evaluations. As discussed in Section 4 of this report, uncertainties in the nuclear design are also accommodated by the use of burnup windows, the generation of conservative axial power shapes for the fast transient analysis, and conservative initial and boundary conditions on the evaluation of AOOs and accidents described in Sections 7 and 8 of this report.

The mechanical design uncertainties for ABB fuel are established from such sources as laboratory tests, materials specifications and industry literature sources, manufacturing process qualifications, and poolside and hot cell post irradiation examinations. Required uncertainties in mechanical design parameters for mechanical evaluation of non-ABB fuel in mixed cores are obtained from the utility.

NRC Question F3

*There is no mention of ABB's intent to submit testing, inspection and surveillance programs to ensure the operational acceptability of proposed reload fuel designs. Discuss where description of these programs fit into the overall reload methodology.*

ABB Response to Question F3

As discussed in Section 3.1.4 of Reference F9, all new designs and design features will be evaluated with the methodology accepted by the NRC relative to the approved design bases. The NRC is notified of the first application of new fuel designs prior to loading into a reactor. New fuel designs and design features are provided to the NRC for information as supplements to Reference F9.

Significant new design features are tested prior to full reload application. New design features are tested with out-of-reactor prototype testing, with Lead Fuel Assemblies, or with a combination of both approaches.

Furthermore, sufficient post-irradiation fuel examinations are performed to confirm that the fuel, including fuel assemblies with new design features, are operating as expected. The ABB post-irradiation surveillance program is described in Section 9 of Reference F9.

NRC Question F4

*Discuss in detail the process, in addition to Figure 4-1, to develop the Reference Core. Discussion should include information related to meeting various safety limits and plausible deviations. Furthermore, discuss and justify the frequency with which the comparison between the actual core average axial burn-up distribution near the end-of-cycle and that assumed for the Reference Core safety analysis is to be performed. (p4-11)*

ABB Response to Question F4

The objective of the development of the Reference Core loading pattern is to identify the most efficient reload fuel and core design that satisfies the plant operational requirements. The development of the Reference Core loading pattern (denoted Cycle N+1) begins with the utility supplied cycle energy requirements and a best estimate prediction of the current end-of-cycle (denoted Cycle N) exposure conditions. Based on this information, a reference fuel cycle is developed, which identifies the fuel assembly design to be used for Cycle N+1. The reference fuel cycle is developed with sufficient margin to the anticipated plant operating limits to assure that the plant operational goals and all licensing requirements can be satisfied. The amount of margin that is determined to be acceptable is based on the experience obtained from performing similar reload design and safety analyses.

The Reference Core loading pattern represents an optimization process for the reference fuel cycle. Development of the Reference Core loading pattern begins prior to the completion of Cycle N. It is based on:

- The new reload fuel assemblies purchased based on the reference fuel cycle,
- The predicted end-of-cycle conditions for fuel assembly inventory available in the core, and
- The exposure condition for and fuel assemblies in spent fuel storage pool that may be reinserted.

The initial design of the Reference Core loading pattern utilizes the experience of the core designer to locate the fuel assemblies to optimize the performance of the core. The core design analyses are performed to demonstrate that the Reference Core loading pattern will meet the energy requirements for Cycle N+1 and the shutdown requirements throughout Cycle N+1. In addition, an assessment is made of the anticipated operating margin to assure there is sufficient operational flexibility throughout the entire operating cycle. The reload fuel safety analysis is then performed for all of the potentially limiting events to establish the core operating limits necessary to satisfy all of the safety analysis requirements. If it is determined that the Reference Core satisfies the operational requirements and has sufficient operational flexibility to satisfy the core operating limits for Cycle N+1, then the Reference Core loading pattern is considered finalized. If it is determined that the Reference Core does not satisfy the operational requirements or have sufficient operational flexibility, the loading pattern is modified until all of the requirements are satisfied.

As described above, the reload safety analysis is performed based on the Reference Core loading pattern, prior to the start of plant refueling. The actual reload core configuration and assumed exposure distribution generally may deviate slightly from the Reference Core assumed in the safety analysis. Verification is performed for the as loaded reload core to confirm that the reload safety analysis is valid.

New analyses, as required, are performed to establish an acceptable set of core operating limits has been established.

The core operating limits developed from the reload safety analysis are typically based on the predicted end-of-cycle conditions (e.g., average axial power distribution based on the predicted exposure distribution for operating cycle). To develop a conservative end-of-cycle axial power distribution, control rod patterns and operating strategies are used to yield reasonable assurance that a more limiting axial exposure will not be encountered in actual plant operation. The axial power distribution is checked by the licensee throughout the operating cycle to verify that the end-of-cycle axial power distribution is within the reload safety analysis. The frequency of the axial power distribution surveillance's for a specific plant is established by the licensee, based in part on the original nuclear steam supply system supplier recommendations and utility operating experience.

In some cases, the operational strategy and control rod patterns may be modified to assure the end-of-cycle axial power distribution is acceptably within the reload safety analysis. Should the end-of-cycle power distribution still be predicted to be more limiting than that assumed in the reload safety analysis, then appropriate analyses are performed for the potentially limiting events, and, as required, the core operating limits are modified.

NRC Question F5

*Discuss the method, codes and assumptions used to determine the void history distribution (page 4-1).*

ABB Response to Question F5

As the core depletes, the void distribution influences the nodal isotopic content and reactivity coefficients. All else being equal, nodes with a higher void fraction produce more plutonium and have a more negative void coefficient of reactivity. However, other fuel parameters, such as pellet diameter, pin-to-pin pitch, water tube, water channel, or watercross design and initial enrichment may also impact burnup characteristics. These burnup effects must be accurately represented in the fuel cross-sections.

Cross-sections are input to nuclear simulator codes (i.e., POLCA and RAMONA-3) as "libraries" for each fuel design. The cross sections are generated by a lattice physics code (i.e., PHOENIX). The cross section libraries are dependent on major state parameters (i.e., exposure, void fraction and control rod presence). The effect of the void fraction experienced during depletion is also a parameter of dependence for these libraries. This parameter is referred to as the void history, VH, and is actually an exposure-weighted nodal void fraction:

$$VH_i = \frac{\int_0^{E_i} \alpha_i(E')\, dE'}{\int_0^{E_i} E'\, dE'} \tag{F5-1}$$

where $\alpha_i$ is the nodal void fraction and $E_i$ is the cumulative exposure for all cycles of residence for the node of interest (denoted "i").

Thus, void history is introduced into the reload safety analysis process through the generation and use of nuclear cross sections.

NRC Question F6

*Does ABB perform re analysis of design basis accidents to assure that they are still bounding? Discuss and justify the criteria ABB will use to determine when it is necessary to re analyze design basis accidents or re-consider accident scenario and/or assumptions to assure that original scenarios/assumptions are still bounding.*

ABB Response to Question F6

In the ABB reload safety analysis process, the following events are considered to be potentially limiting events:

(1) Anticipated Operational Occurrences

- (a) Turbine Trip or Generator Load Rejection without Bypass
- (b) Pressure Regulator Failure - Closed (BWR/6 Only)
- (c) Loss of Feedwater Heating
- (d) Control Rod Withdrawal Error
- (e) Recirculation Flow Controller Failure - Increasing Flow
- (f) Feedwater Controller Failure - Maximum Demand

(2) Accidents

- (a) Loss of Coolant Accident
- (b) Control Rod Drop Accident
- (c) Refueling Accident
- (d) Fuel Loading Error

(3) Special Events

(a) Core Thermal-Hydraulic Stability

(b) Reactor Overpressure Protection

(c) Shutdown without Control Rods

The above generic set of events have been established as potentially limiting by a review of the current license basis events for each of the BWR product lines and confirmed by a review of the event signatures consistent with the use of ABB reload fuel design and safety analysis methodology.

For each new reload fuel supply contract in which ABB has scope of supply including reload fuel safety analysis services, ABB performs a comprehensive review of the current plant specific license basis and plant performance requirements to assure that the above list of events established in the safety analysis process and reload methodology identified in this report remain applicable. Based upon the plant specific review, additional potentially limiting event may be added to the generic events list.

Each of the events is evaluated for the first ABB reload application and for each subsequent reload if an applicable generic or bounding analysis is not available. In addition, ABB reviews each reload application, consistent with the requirements of 10CFR50.59, to assure that the cycle specific application does not introduce the potential for another event to become limiting. If another event is identified as potentially limiting, it is analyzed as a part of the reload safety analysis process. For typical BWR reloads, ABB has performed sufficient analyses to demonstrate that the generic set of analyses is sufficient to establish the core operating limits or demonstrate conformance to the applicable event acceptance limits. These events cover the entire spectrum of safety analysis events that are significantly impacted by the introduction of reload fuel and a new core configuration.

Therefore, it is not necessary to analyze additional anticipated operational occurrences, accidents, or special events beyond those identified in this report, unless there is a unique license basis or plant performance requirement that leads to the need to consider additional events beyond those identified above.

NRC Question F7

*Discuss the process by which the burn-up window is selected for the Reference Core. Discuss further the conditions/circumstances in which the actual cycle exposure would fall outside of the burn-up exposure. (p.4-11)*

ABB Response to Question 7

To complete the reload safety analysis in a timely manner, it is necessary to begin the fuel design and safety analysis for Cycle N+1 process prior to the end of cycle (EOC) N. Therefore, it is necessary to define a nominal EOC exposure for Cycle N. To accommodate deviations in actual plant operation from the nominal exposure, a "burnup window" is provided. The reload safety analysis for Cycle N+1 is valid as long as the actual EOC for Cycle N falls within the burnup window.

The selection of the size of the burnup window represents a balance between efficient fuel utilization and cycle length flexibility. If a utility desires a fuel and core design that will accommodate a larger burnup window, it can be accommodated, but generally at the expense of a less optimized core design, which may incorporate additional fuel assemblies.

[ Proprietary Information Deleted ]

The nominal burnup window is sufficiently large that most operating cycles are completed within its constraints. However, there are circumstances in which the plant can shutdown for refueling outside of the burnup window. A plant can enter a refueling outage prior to attaining the minimum exposure by, for example, having equipment problems or low plant capacity factor. A plant can enter a refueling outage after exceeding the maximum exposure by having a high capacity factor or an extended power coastdown due to system demand requirements. In either case, the reload safety analysis is augmented, as required, to cover the actual Cycle N exposure at plant shutdown for refueling.

NRC Question F8

*Describe the procedure of establishing the Kcold (reference) from the plant data when the reference core is a transition core. (p4-18).*

ABB Response to Question F8

The ABB practice for establishing the hot and cold eigenvalues for a reactor into which ABB reload fuel is to be installed, is to analyze the core tracking results of previous cycles for the reactor in question. The core tracking process provides the plant and core specific benchmark for the cold eigenvalues and hot eigenvalues. Furthermore, the benchmarking provides cycle trends and uncertainty bounds of the eigenvalues.

The cold eigenvalue is calculated with the plant data from the cold critical measurements. The hot eigenvalue is calculated with the plant data at hot operating conditions. When available, comparisons between TIP measurements and the ABB three-dimensional core simulator calculations give further information on the accuracy the core simulator model and measurement data.

The hot and cold eigenvalues typically vary slightly during the cycle or from cycle to cycle. When the hot or cold eigenvalue for the next cycle is chosen, the trend and magnitude in the hot and cold eigenvalues are taken into account.

ABB has accumulated vast experience with core tracking calculations both in Europe and USA. About 20 reactors have been evaluated, some with over 15 cycles of operation. The core tracking calculations with the ABB nuclear code system (PHOENIX/POLCA) represent many transition cores, such as transition cores from 8x8 to 9x9 or 8x8 to 10x10 fuel designs.

The ABB core tracking experience shows to date that the variation of the hot or cold eigenvalue for a transition core is of the same magnitude as for an equilibrium core. variation from cycle to cycle. [ Proprietary Information Deleted ]

NRC Question F9

*Discuss the quantitative criteria used to categorize "fast" and "slow" transients. ABB described the procedures to assure that the calculational results are self-consistent for steady state calculations. Provide transient analysis results for "slow" transients to demonstrate that steady state modeling of such transients is appropriate.*

ABB Response to Question F9

Fast transients are events that are generally characterized by rapid changes in neutron flux (events with time scales of several seconds, such as, load rejection or turbine trip). These events typically result in a scram being initiated on either the event initiator (e.g., valve position switches) or high neutron flux. For fast transients, the simulation of the system response is important.

Slow transients are generally characterized by relatively slow changes in neutron flux (events with a time scales of several minutes, such as, changes in moderator temperature). In these events the reactor heat flux closely tracks the neutron flux. Protective action for these events, if required, is based on neutron flux or simulated thermal power. For slow transients, the simulation of the system response is relatively unimportant, but the simulation of the three dimensional core response (i.e., changes in power distribution) is important. In illustration, Figures F9-1 and F9-2 show the core power and heat flux time response for a typical fast and slow transient, specifically, for a generator load rejection and loss of feedwater heating event.

[ Proprietary Information Deleted ]

NRC Question F10

*In the fast transient methodology, ABB describes a two-step approach in which the core is modeled as an average core in a first step. Then in the second step results from the first step are used in the hot channel analysis. Discuss and demonstrate the conservatism of this approach of using the average core characteristics in the hot channel analysis rather than generating the hot channel conditions in the first step by modeling the core consisting of two channels: one representing the average core and the other the hot channel.*

ABB Response to Question F10

The "two step" BISON core / BISON slave channel analysis procedure is equivalent to a "one step" process using parallel, one-dimensional channels.

In BWR analysis, channel flows can be analyzed in a decoupled manner because explicit modeling of one hot channel does not change the overall core average response. Physically, this is because:

- The channel walls keep the assembly flows from mixing with flow from other assemblies and with the inter-assembly (bypass) flow, and

- The pressures in the upper and lower plena equalize the inlet and outlet pressures for all channels.

In one-dimensional transient analysis, the three-dimensional core is collapsed to a single average channel which represents the kinetic and thermal-hydraulic response of all of the fuel assemblies in the core. This one-dimensional procedure eliminates radial variations, and implicitly assumes that the neutron flux transient is uniform for all fuel channels. [ Proprietary Information Deleted ]

In theory, the hot channel can be modeled and evaluated in parallel to the core model. In such a case either the same boundary conditions for the hot channel are used or the hot channel influence on the plena inlet and outlet pressures is included in the system response calculation. Using the first option would result in negligible differences with respect to the results of the two-step process. The impact of the second option on the transient neutronic and hydraulic response is negligible, in light of the large number of channels in a BWR core.

The decision to perform the hot channel analysis in parallel (i.e., in the same computer run) or as a separate calculation (i.e., in a subsequent run) is strictly based on convenience. For all practical purposes, both hot channel approaches will use the same boundary conditions and will therefore yield essentially the same results.

NRC Question F11

*Discuss thoroughly the methodology used to determine the SLMCPR both deterministically and statistically. Justify the selection of parameters on Table 5-3 including ranges and distributions of uncertainties and computation of standard deviation. Discuss the merits and pitfalls of the use of statistically determined SLMCPR over the deterministically determined value. In addition, discuss how the values for all other input are selected. Discuss and justify the sample size of Monte Carlo analysis and the transient selected for such analysis. Furthermore, discuss sources and meaning of uncertainties (Table D.4.4-1) used in the SLMCPR calculation.*

ABB Response to Question F11

The SLMCPR accommodates the uncertainties in the steady-state MCPR values which are utilized to determine margin to boiling transition during operations. The uncertainties in the change in CPR (ΔCPR) during an Anticipated Operational Occurrence (AOO) are evaluated separately from the SLMCPR evaluation as discussed in Section 7.3 of this report. The methodology and sources of inputs used in the determination of the SLMCPR are elaborated below.

The ABB methodology described in Section 5 of this report is a Monte Carlo methodology. No ABB deterministic methodology is currently being proposed for the determination of SLMCPR for U.S. applications. In principal, however, utilization of deterministic or Monte Carlo methods are both valid provided that they adequately capture the effects of uncertainties in plant and assembly parameters which affect the magnitude of the on-line monitored MCPR that is compared to the OLMCPR. It is judged that the Monte Carlo method provides the best currently available means of combining the various uncertainties affecting the on-line monitored MCPR without making simplifying assumptions regarding the dependence of the various parameters. For example, deterministic methods of convoluting uncertainties typically combine uncertainties in various parameters assuming that they are independent and that the resulting total uncertainty in CPR is a normal distribution. Beside not being restricted by the assumption of parameter independence, the Monte Carlo methodology can be conveniently applied in a manner which is completely general with regard to the magnitude of the uncertainties and the type of uncertainty distributions. Monte Carlo methods are the predominate methods used for the determination of SLMCPR in the U.S.

The ABB Monte Carlo SLMCPR methodology is described in Section 5.3.2.2 of this report, and some clarification of the calculational flow is provided as follows. As noted in Section 5.3.2.2, [ Proprietary Information Deleted ]

All uncertainties which significantly affect the calculated steady-state MCPR are considered in the SLMCPR calculation. In principal, the choice of uncertainties and their magnitude can depend on the plant type, the core supervision system, the fuel design, and the methodology used for the evaluation of CPR. [ Proprietary Information Deleted ]

The sample size in the Monte Carlo evaluation is selected to be large enough to assure that the calculation is converged. That is, it is selected such that additional trials will not significantly change the result. [ Proprietary Information Deleted ]

NRC Question F12

*Provide complete and thorough discussion and qualification of the perturbation methodology using response functions for the CPR distribution.*

ABB Response to Question F12

The ABB methodology for establishing the SLMCPR is discussed in Section 5 of this report and the Response to Question F11. This analysis accounts for the uncertainties in plant and assembly parameters which affect the magnitude of the on-line monitored steady state MCPR.

A separate evaluation is performed to address uncertainties in the change in MCPR (ΔCPR) during an AOO. This response deals with the transient uncertainty approaches (see Response to Question F18) that uses a response surface function in the determination of the uncertainty in the ΔCPR during the transient.

In the reload safety analysis process, it is generally recognized that the use of a very conservative deterministic analysis methodology for establishing the ΔCPR for certain events can lead to the use of an overly restrictive operating limit that can result in unnecessary plant operating restrictions. In the deterministic analysis process (see Approach A in Section 7.3.3.1 of this report), conservative analysis input parameters are used to assure that the operating limits have an acceptable level of conservatism. To obtain operationally acceptable operating limits, which have an appropriate level of conservatism, some form of statistical methodology is sometimes applied. Using statistical methodology, the uncertainties in the ΔCPR are combined with the nominal value such that for an acceptable level of probability the SLMCPR will not be exceeded during the event. In order to make a reasonable statistical assessment, a large number of experiments (e.g., Monte Carlo trials) are required. [ Proprietary Information Deleted ] acceptable operating limit.

[ Proprietary Information Deleted ]

Perturbation analyses can be used on two occasions in the process to establish the response surface function.

[ Proprietary Information Deleted ]

NRC Question F13

*Discuss the impact of having two limits (one selected deterministically for non-ABB fuel and the other statistically for ABB fuel) for transition core analysis.*

In principal the utilization of deterministic or Monte Carlo methods are equally valid for the calculation of SLMCPR, provided that they adequately capture the effects of uncertainties in the key plant and assembly parameters. It is our understanding that Monte Carlo methods are the predominate methods used for the determination of SLMCPR in the U.S. Please see the response to Question F11 for further discussion.

[ Proprietary Information Deleted ]

NRC Question F14

*ABB stated that when ABB does not have direct access to the CPR correlation on the non-ABB resident fuel, ABB renormalize the GEXL correlation based upon the information provided by the licensee. Describe completely and thoroughly the renormalization process and the data base to support a conclusion that the resulting correlation is acceptable. In addition, ABB should discuss the impact of renormalization on the uncertainty associated with the use of such correlations. Discuss the methodology and justification for use of a correlation to be renormalized which has not been approved by the NRC for that fuel assembly. Does ABB attempt to benchmark its ABB methodology against the existing set of analyses performed by other organizations in determination of gap HTC's, LHGR, and other key parameters in reload analysis of mixed cores? If so, what will ABB attempt to demonstrate?*

ABB Response to Question F14

Generally whenever the reload fuel supplier is changed, utilities retain access to the NRC approved CPR correlation provided with the previous cycle's reload fuel, with the right to continue use of this correlation to monitor the thermal limits of these fuel assemblies during mixed core operation. ABB also provides the appropriate CPR correlation which has been reviewed by the NRC to the utility for the monitoring of ABB fuel assembly thermal limits for reload applications. Therefore, licensed CPR correlations provided by the fuel vendor are always used for the on-line monitoring of fuel relative to thermal limits.

[ Proprietary Information Deleted ]

In practice, resident fuel assemblies will be at the end of their second operating cycle when core conditions become most limiting for fast transients. [ Proprietary Information Deleted ]

ABB does not benchmark its methodology against analyses performed by other organizations when determining gap heat transfer coefficients, LHGR, and other key parameters in reload analysis of mixed cores. As with other vendor's analysis methods, each part of the ABB methodology is sufficiently qualified for analysis of BWR cores comprised of ABB and non-ABB fuel designs. Sufficient information is obtained from the utilities to ensure that non-ABB fuel is accurately or conservatively modeled in the ABB methodology.

For example, the ABB nuclear codes are sufficiently general to describe the range of fuel lattices in current BWR fuel designs. Qualification of the ABB nuclear methods are made by comparison of the predictions to measured power distribution data, such as those in Reference F6. LHGR values predicted by the ABB codes are generally in good agreement with those of other state-of-the-art codes (e.g., PRESTO and SIMULATE). In addition, gap heat transfer coefficients ($h_{gap}$) calculated with ABB methods are based on an NRC approved fuel performance code comprised of models based on the ABB and available public BWR fuel data.

NRC Question F15

*We are unable to find documentation of the qualification of the BISON mixture level computation against data. If this has been already provided in another topical report, provide the reference. Otherwise, provide such documentation.*

ABB Response to Question F15

The BISON water level calculation has been qualified against plant operational transient data. The results of the qualification against a Barsebäck 2 transient are summarized here.

The reactor vessel water level is determined in a plant by measuring the pressure difference between the downcomer and a liquid filled reference leg. The indicated reactor level is the collapsed level in the downcomer and is inferred from the measured instrument pressure drop and assumed vessel conditions including: downcomer and reference leg water densities and dynamic pressure conditions at the pressure taps.

[ Proprietary Information Deleted ]

Figure F15-1 compares the water level calculated by BISON against the water level from a measured scram at the Barsebäck 2 plant, which occurred August 26, 1992. Figure F15-1 shows a good comparison to the water level data presented. [ Proprietary Information Deleted ]

BISON correctly predicts the general level trend, but overpredicts the measured water level throughout the event. Even though this overprediction is within the qualification uncertainties, the impact on licensing analysis is still negligible. In licensing analysis, the water level predictions do not measurably affect any rapid pressurization transient results. The level prediction potentially could impact the feedwater controller failure transient results, where overpredictions could cause an earlier high water level (Level 8) turbine trip. However, level overpredictions of the observed magnitude are not expected for the FWCF analysis since the primary reactor system parameters prior to the high water level trip are changing very slowly in that transient. [ Proprietary Information Deleted ]

NRC Question F16

*Clarify the value used for the peak reactor vessel pressure (110% or 120%) in the ABB reload safety analysis.*

ABB Response to Question F16

In the ABB reload safety analysis process, the value used for the peak reactor vessel pressure acceptance limit is the value contained in the license basis analysis for the plant in question. ABB will not use a value other than the plant license basis without specific approval by the NRC.

In a plant licensing basis, the most limiting event analyzed with respect to peak reactor vessel pressure is generally the overpressure protection analysis that demonstrates compliance with the ASME Code overpressure limits. Section 9.3.1 of this report, states that for overpressure protection the plant specific licensing basis limit shall not be exceeded. For this event, the acceptance limit that is typically used is the Code "upset" limit of 110% of the reactor vessel design pressure (e.g., 1.1 x 1250 = 1375 psig). The use of the Code "upset" limit is based on a conservative interpretation of the ASME Code for the allowable increase in pressure associated with the frequency and duration of the event.

For analysis of anticipated transients without scram (ATWS), the plant specific licensing basis limits shall not be exceeded. Section 9.5.2 of this report list the design bases and the typically used acceptance limit for primary system integrity. Plant specific licensing bases generally use the ASME Code "emergency" limit of 120% of the reactor design pressure (e.g., 1.2 x 1250 = 1500 psig) as an acceptance limit for peak reactor vessel pressure.

The discussions presented in Section 6 of this report include reference to the above mentioned typical plant licensing bases acceptance limits.

NRC Question F17

*Describe completely and thoroughly the methodology (methods, codes, assumed statepoints, bounding analyses and transient assumptions, where applicable) to develop the following limits (ref. Figs. 5-2 & 5-3) (if they are presented in other topical report or previously reviewed and approved, provide references):*

*a.* *Allowable Operating Domain*

*b.* *MEOD (maximum extended operating domain)*

*c.* *Operating Limit MCPR*

*d.* *Operating Limit LHGR*

*e.* *MAPLHGR*

*f.* *SAFDL*

*g.* *ECCS Acceptance Limit*

*h.* *Stability Limit*

*i.* *Control Rod Drop Acceptance Limit*

*j.* *Reactor Vessel Pressure Limit*

*k.* *Standby Liquid Control Capacity Limit*

ABB Response to Question F17

The following discusses each of the terms listed in the question.

*a. Allowable Operating Domain*

The allowable operating domain refers to the spectrum of steady state conditions (operating states) that is covered by the safety analysis. The allowable operating domain represents the license basis for planned operation. For analysis purposes, it is typically considered to be the allowable power/flow map, burnup range, and operational flexibility options that are evaluated for the plant. Thus, the allowable operating domain is used as an input to the safety analysis process. In the ABB reload safety analysis process, the allowable operating domain is treated in the same manner as the current plant licensing basis. Each specific analyses of the current plant licensing basis is described in Section 6 of this report. A description of the process for establishing the plant allowable operating domain is provided in Section 6.5 of this report. The specific analyses performed to validate the allowable operating domain are described in Section 7 (anticipated operational occurrences), Section 8 (accidents), and Section 9 (special events). Examples of these analyses are provided in Appendix D.

*b. MEOD (Maximum Extended Operating Domain)*

The maximum extended operating domain (MEOD) is the term used to represent a combination of three other flexibility options:

- Maximum Extension to the Load Line Limit Analysis "MELLA" (Appendix C.1 of this report)
- Increased Core Flow "ICF" (Appendix C.2 of this report)
- APRM - RBM Technical Specification Improvements "ARTS" (Appendix C.12 of this report)

The MEOD option was so named because its objective is to justify the maximum operation region on the power flow map. The MEOD operating flexibility option is discussed further in Appendix C.3 of this report, along with the analyses and evaluations that are performed to demonstrate its acceptability.

The remaining terms are acceptance and operating limits which are considered during the reload safety analysis process diagrammed in Figure 6-3. Each of these terms are discussed individually below.

*c. Operating Limit MCPR*

The operating limit Minimum Critical Power Ratio (MCPR) is the core operating limit necessary to assure that the safety limit MCPR will not be exceeded as a result of an anticipated operational occurrence or that the plant will not operate in an unanalyzed condition. The operating limit MCPR is established as the greater of either: (1) the combination of the safety limit MCPR and the change in critical power ratio (ΔCPR) for the limiting anticipated operational occurrence; or (2) the operating limit MCPR assumed as an initial condition for the loss of coolant accident analysis. The specific methodology to establish the MCPR operating limit is provided in Section 7.3.4.1 of this report.

*d. Operating Limit LHGR*

The operating limit linear heat generation rate (LHGR) is the core operating limit necessary to assure that the fuel thermal-mechanical performance limits (e.g., fuel plastic strain design limit or fuel centerline melt limits) will not be exceeded as a result of an anticipated operational occurrence or that the plant will not operate in an unanalyzed condition. The operating limit LHGR is established as the lesser of either: (1) the LHGR required to assure the fuel plastic strain design limit or the fuel centerline melt limit will not be exceeded; or (2) the operating limit LHGR required to satisfy the ECCS acceptance limits for the loss of coolant accident analysis. The specific methodology to establish the LHGR operating limit is provided in Section 7.3.4.2 of this report.

*e. MAPLGHR*

The maximum average planar linear heat generation rate (MAPLHGR) is used as a core operating limit as a function of fuel assembly exposure for some plants. The allowable MAPLHGR is usually derived from the results of the loss of coolant accident analysis and/or the design LHGR (see the further discussion in the Response to Question F33).

*f. SAFDL*

Specified acceptable fuel design limits (SAFDLs) are used as the figure of merit for the results of the analysis of anticipated operational occurrences consistent with the requirements of General Design Criterion (GDC-10). In the safety analysis process, there are three specific SAFDLs of interest. These are: (1) the safety limit MCPR; (2) the fuel clad strain and centerline melt temperature limits; and (3) the peak fuel enthalpy design limit. The methodology for establishing the safety limit MCPR is provided in Section 5.3.2.1 of this report. The fuel assembly mechanical design methods (References F9 and F10) are used to establish the exposure-dependent LHGR limits which assure that fuel clad strains in excess of acceptable limits and centerline melt will not occur during normal operations or AOOs. The peak fuel enthalpy design limit is 170 cal/g, which is consistent with Section 4.2 for NUREG-0800 (NRC Standard Review Plan).

*g. ECCS Acceptance Limit*

The emergency core cooling system (ECCS) acceptance limits are contained in 10CFR50.46. There are five acceptance limits: (1) the calculated maximum fuel element cladding temperature is not to exceed 2200 °F; (2) the calculated local oxidation of the cladding is not to exceed 0.17 times the local cladding thickness before oxidation; (3) the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam is not to exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react; (4) calculated changes in core geometry are such that the core remains amenable to cooling; and (5) after any calculated successful operation of the emergency core cooling system, the calculated core temperature shall be maintained for the extended period of time required by the long-lived radioactivity remaining in the core. The ABB methodology showing compliance with the ECCS Acceptance limits is discussed in Section 8.2 of this report.

*h. Stability Limit*

The stability limits are based on the requirements of GDC-12. GDC-12 requires, "The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed." Thus, the SAFDLs identified above are considered to be the stability design limits. The stability acceptance limits are established to comply with design limit. Specific acceptance limits are based on the analysis methods being used (Reference F3).

*i. Control Rod Drop Acceptance Limit*

There are two basic control rod drop acceptance limits: (1) the onsite and offsite radiological effects and (2) the peak fuel enthalpy limit. The event acceptance limit for offsite radiological effects is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological effects is the limits identified in the GDC 19. The peak fuel enthalpy limit is a calculated peak fuel enthalpy of 280 cal/g to assure the integrity of the reactor coolant pressure boundary. For radiological evaluation purposes, all fuel rods evaluated to exceed a peak enthalpy of 170 cal/g are assumed to fail. The specific methodology used to evaluate CRDA is provided in Reference F7.

*j. Reactor Vessel Pressure Limit*

The reactor vessel pressure limit is determined based on the specific plant license basis for the event being analyzed. The Response to Question F16 provides additional information on the identification of the appropriate reactor vessel pressure limits.

*k. Standby Liquid Control Capacity Limit*

The standby liquid control system capability limit is a $k_{eff} < 1.0$ at the most reactive temperature. This value provides assurance that the reactor will be subcritical at the most reactive temperature. Further discussion of the standby liquid control system capability analysis is provided in the Response to Question F25.

NRC Question F18

*Discuss the process by which ABB assures that a more realistic analysis, when the bounding analysis is found to be too restrictive, retains an appropriate level of conservatism (p. 7-5)*

ABB Response to Question F18

The objective of the plant safety analysis is to demonstrate that the plant does not represent an undue risk to the health and safety of the public. To make this demonstration conservative, event acceptance limits are established consistent with the event probability. The analysis of the events is then performed in a conservative manner to demonstrate that there is a very high probability that the event acceptance limits will not be exceeded. In accordance with the requirements of 10CFR50.59, the ABB reload fuel design and safety analysis processes have been developed to provide the required demonstration that plant modifications (e.g., reload fuel applications and changes impacting core configuration) satisfy the required standards for updating the plant safety analysis consistent with the current plant license basis.

ABB has identified four different approaches (see Section 7.3.3.1 of this report), designated A, B, C, and D, that can be used to provide the necessary assurance that the event acceptance limits will not be exceeded as a result of a proposed plant modification. Approaches A, B, and C are currently used for licensing analyses. Approach D is currently not used for licensing analyses. Approaches A, B, and C are each formulated to assure that plant technical specification limits are not violated. Approach C is the most precise of the three approaches in providing a conservative estimate in the uncertainties associated with the analysis. Approaches B and A represent increasingly conservative alternatives which are utilized for convenience when margins to limits are sufficiently large that they can be demonstrated with these less precise alternatives. It must be emphasized that all these approaches include a level of conservatism as depicted in Figures F18-1 through F18-4.

The four approaches are described below.

*Approach A*

The most conservative approach is the deterministic analysis method (Approach A, see Figure F18-1). In the deterministic method, conservative analysis parameters (license basis inputs) and models are used in the safety analysis process to provide the necessary assurance that the event acceptance limits will not be exceeded. The deterministic analysis process is the "traditional" approach to the analysis of the events in the plant safety analysis, and the majority of events are analyzed using this approach. In this process, no specific quantification of the probability of exceeding the event acceptance limits is made.

*Approach B*

The second most conservative approach is the simplified statistical analysis process (Approach B, see Figure F18-2). In the simplified statistical analysis process, selected model inputs are input to the analysis process on a nominal basis. [ Proprietary Information Deleted ] The use of the deterministic input parameters for the remainder of the analysis inputs provides additional conservatism.

*Approach C*

The next most realistic, but still conservative, approach is the response surface function method (Approach C, see Figure F18-3). In the response surface function method, a polynomial representation of the system analysis code (see the Response to Question F12) is developed. [ Proprietary Information Deleted ] This approach is consistent with the current industry standard for performing this type of analysis. As with Approach B, the use of the deterministic input parameters for the remainder of the analysis inputs provides additional conservatism.

*Approach D*

The final approach is the modified response surface method (Approach D, see Figure F18-4). The modified response surface method is the same as Approach C, [ Proprietary Information Deleted ]

NRC Question F19

*Discuss in detail the methods of determining the following:*

a. *The licensing bias (Method A)*

b. *The expected mean value of the operating limit (Method A)*

c. *The biased licensing value of the operating limit (Method A)*

d. *Selection of input parameters and their independence in Method B and the assumption that these parameters are normally distributed (Method B)*

e. *Statistical method used in Method B*

f. *Justify Equation 7-9*

g. *Methods C and D with respect to uncertainties (Eqs. 7-13 & 7-15)*

ABB Response to Question F19

The ABB approach for treating uncertainties in the safety analysis process is described in the Response to Question F18. Each of these approaches provides an acceptable level of assurance that the event acceptance limits will not be exceeded.

Approach A is the deterministic analysis method that forms the basis for most of the event analyses that are contained in the specific plant safety analysis. The licensing bias in the operating limit is determined by using conservative (licensing basis) inputs and models which assure that the results are conservative. The biased conservative value of the operating limit is obtained directly from the results of the safety analysis. Because a conservative value for the operating limit is obtained directly from the analysis, it is not necessary to separate the mean value of the operating limit and the licensing bias. The equations presented in the discussion of Approach A are conceptual to illustrate the approach (Figure F18-1 further illustrates the approach.) The equations are not used in the application of Approach A.

Approach B is the simplified statistical analysis process. [ Proprietary Information Deleted ]

NRC Question F20

*Although ABB stated that when the conventionally determined limit is restrictive upon reload, it will select a "more realistic" method, ABB seems to already know that for recirculation flow controller failure, for example, it intends to use Method D. ABB should discuss the procedure of selecting a method to determine an appropriate operating limit. Discuss and justify the need for the impact of having several operating limits on safety analysis and plant operation. Explain what is meant by the approach "neglecting the inherent conservatism in the licensing analysis through the use of conservative and bounding input parameters for those inputs not treated statistically".*

ABB Response to Question F20

As discussed in the response to Question F4, the objective of the development of the Reference Core loading pattern is to identify an efficient reload fuel and core design that satisfies the plant operational requirements and all safety limits. In the process of developing the Reference Core loading pattern, expected operating limits are identified, and it is demonstrated that the core has sufficient margin to operate within the expected operating limits without unnecessary restrictions.

A safety analysis is considered acceptable if the operating limits identified as a result of the application of the safety analysis process can be accommodated without encountering unnecessary restrictions. As a general guide, the results of the safety analysis are considered acceptable if the operating limit identified through the performance of the safety analysis is the same as or less restrictive than the expected operating limit assumed in the development of the Reference Core loading pattern. For the majority of the events considered in the reload safety analysis process, sufficient operating margin is available through the use of deterministic analysis process (Approach A). For these events, it is not necessary to use a statistically based safety analysis approach. [ Proprietary Information Deleted ]

The reload safety analysis is used to define or confirm the suitability of two basic operating limits for each fuel type (lattice) in the core. These limits are the operating limit minimum critical power ratio and the operating limit linear heat generation rate or maximum average planar linear heat generation rate. The limits may be flow or power dependent and may be modified for different modes of operation (e.g., single loop operation). These limits are placed in the plant process computer and used to monitor plant operation for conformance to the limits. These limits have been used in plant operation for over 20 years. The ABB safety analysis process provides limits that are consistent with this experience base.

The terminology "neglecting the inherent conservatism in the licensing analysis through the use of conservative and bounding input parameters for those inputs not treated statistically" refers to the treatment of input parameters in the statistically based safety analysis process. [ Proprietary Information Deleted ]

For clarification, the terminology "subregion of the allowable operating domain" in Section 7.3.4.1 refers to situations where multiple OLMCPRs are provided to allow the utility operational flexibility. For example, two sets of OLMCPRs are typically provided to allow plant operation with and without the Recirculation Pump Trip operable. In this case, the utility uses the set of operating limits that apply to the status of equipment availability.

NRC Question F21

*Since the Rod Withdrawal Error analysis is a highly localized transient, justify not considering the fuel densification effect in analysis of maximum LHGR.*

ABB Response to Question F21

The ABB fuel pellets are highly sintered with a very stable microstructure and not subject to a significant amount of densification. The change in column length due to densification is less than the fuel pellet expansion in going from the cold condition to design linear heat generation rate. [ Proprietary Information Deleted ]

NRC Question F22

*The LOCA analysis section makes no reference to other ABB topical reports which have been submitted with respect to ABB LOCA methodology. ABB should certify that there is no difference in analysis methodology and basic assumption presented in CENPD-300 and those other submittals.*

ABB Response to Question F22

The LOCA Licensing Topical Reports are listed in Table 1-1 and Table 1-2 of this report. The Licensing Topical Reports RPB-90-93-P-A, RPB-90-94-P-A, CENPD-283-P-A, and CENPD-293-P-A are, respectively, References 21, 22, 35, and 40 in Section 10 of this report. Section 8.2 describes the LOCA reload evaluation process using an NRC approved LOCA analysis methodology. The following paragraphs summarize the LOCA analysis methodologies presented in the above reports.

Three sets of ABB LOCA codes are described in the above noted Licensing Topical Reports. For clarity, they are designated "USA1", "USA2", and "USA3". The original NRC approved ABB LOCA codes, designated "USA1", are documented in RPB-90-93-P-A and RPB 90-94-P-A.

The Licensing Topical Report CENPD-293-P-A updates the original "USA1" codes with the inclusion of a new fuel rod performance model, introducing mechanistic models for fuel rod cladding strain and rupture, and other minor code method modifications. The ABB LOCA codes incorporating the methods modifications described in CENPD-293-P-A are designated "USA2".

The third set of LOCA codes, designated "USA3", described in Section D.7.1 of this report is identical to "USA2", with one modification. [ Proprietary Information Deleted ]

The Licensing Topical Report CENPD-283-P-A expands the application of the LOCA methodologies to the SVEA-96 fuel design. The calculational examples presented in CENPD-283-P-A used the "USA1" codes. [ Proprietary Information Deleted ]

The LOCA analysis methodology and basic assumptions to be used in the reload evaluation process described in this report shall be NRC approved code methods, such as those summarized above. The "USA3" codes are used in the example presented in Appendix D.7.1 of this report. [ Proprietary Information Deleted ]

NRC Question F23

*In the reload analysis methodology discuss the sensitivity analyses to be performed for the plant specific reload applications (AOOs, special events and accidents).*

ABB Response to Question F23

For the first plant specific reload, ABB does perform sensitivity analyses at the extreme points on the allowable power/flow map (licensed operating domain) to provide assurance that the limiting power/flow condition is evaluated. Once the limiting point is established, it is not necessary to perform sensitivity analyses for future reloads, unless there is a significant change in the basic fuel design (e.g., a change in the number of fuel rods).

NRC Question F24

*Discuss the method by which the location of the most limiting error rod is determined.*

ABB Response to Question F24

The error rod for the control rod withdrawal error analysis is selected using the three-dimensional nuclear simulator. [ Proprietary Information Deleted ]

NRC Question F25

*In the SLCS analysis, discuss the method and assumptions used to determine the boron concentration and boron worth.*

ABB Response to Question F25

The standby liquid control system capability analysis is performed to demonstrate that the core can be made subcritical in the xenon free cold condition from the limiting hot equilibrium xenon condition without movement of the control rods. The analysis is performed for the two conditions using the three-dimensional nuclear simulator for the reference core configuration based on cross section inputs from the lattice physics methods. The lattice physics methods are used to explicitly model the fuel assemblies contained in the reference core. The moderator cross sections for the cold condition are developed assuming a uniform distribution of the specific plant license basis boron concentration (typically 600 to 660 ppm) in each fuel assembly lattice. The boron microscopic cross sections are used in the lattice physics analysis process to explicitly model effects of the liquid poison. The boron concentration assumed in the plant license basis has a mixing allowance (typically ≈25%) to assure that the uniform mixing assumption is appropriate.

NRC Question F26

*Discuss the factors which contribute to increase in nuclear instrumentation and core flow measurement uncertainties with single loop operation.*

ABB Response to Question F26

The nuclear instrumentation and core flow measurement uncertainties are the only uncertainties used in the determination of the Safety Limit CPR which change as a result of Single Loop Operation (SLO). However, only a small increase in the Safety Limit CPR (typically 0.01) results from increasing these two uncertainty components. SLO is the only operating condition for which the Safety Limit CPR is reevaluated.

The nuclear instrumentation uncertainty is a component of the total bundle power uncertainty. Tests at operating BWRs, confirm small increases in the nuclear instrumentation uncertainty. For example, Reference F4 indicated a change in effective TIP uncertainty from 8.7% to 9.1%. The overall impact on the SLMCPR depends on the determination of assembly power uncertainties in the plant core supervision system.

SLO has a direct impact on the core flow measurement system of a BWR. The core flow measurement system calibration procedure assumes forward flow through all jet pumps. The flow through some inactive jet pumps will, however, be reversed. The measured back flow in these jet pumps must be subtracted from the measured flow in the active loop to obtain the total core flow. The existence of reverse flow leads to an increase in the uncertainty in the measured core flow relative to the two-loop case. For example, Reference F4 established an increase in core flow uncertainty (standard deviation) from 2.5 to 6.0 percent associated with going from two-loop to single loop operation.

NRC Question F27

*In Appendix C, ABB discusses potential extension of the operating limits and operating flexibility. In each of the subsections where specific limits or conditions are discussed, the information is not definitive. Discuss the intended use of information in this appendix and how specific numbers and facts will be made definitive.*

ABB Response to Question F27

The license basis for many BWRs incorporates operating flexibility options. Each BWR licensee may have different operating flexibility options, depending on the specific plant economic and operational goals. The spectrum of plant flexibility options considered in the ABB reload safety analysis process is identified in Appendix C. For plants that have any of these plant flexibility options included in their license basis, the analyses associated with the particular option in Appendix C is performed as a part of the reload safety analysis. These results of the plant flexibility option analyses are used as a part of the process to establish the core operating limits. Therefore, the core operating limits are established consistent with the operating flexibility options, if they are a part of the current plant licensing basis.

If the plant licensee desires to change its license basis to incorporate a new plant flexibility option, the change must be made in accordance with the requirements of 10CFR50.59. ABB has the capability of supporting changes in the plant license basis by performing the analyses associated with the operating flexibility option identified in Appendix C. However, the plant licensee is responsible for satisfying the regulatory requirements associated with plant modifications in 10CFR50.59.

NRC Question F28

*In Appendix A, ABB states that CPR edits are obtained from POLCA. Is POLCA still using AA-74 correlation as presented in BR 91-402-P-A? Has ABB implemented XL-96S into POLCA? Discuss the changes, if any, made to the POLCA/PHOENIX/CONDOR code package to enable computation of SVEA-96 fuel cores. Similarly, identify and discuss any changes to update models in the codes presented in Appendix A necessitated by new fuel designs or for any other purpose. If there are any changes, justify them or reference reports where they have been justified and present the NRC review status. In addition, identify and discuss if any of these codes are replacing codes which have become obsolete.*

ABB Response to Question F28

CPR correlations for ABB fuel are based on loop test data which are applicable to the specific fuel design and are submitted for NRC review. ABB has implemented the XL-S96 CPR correlation described in Reference F5 in the steady-state and transient thermal-hydraulic codes in which the CPR performance of the SVEA-96 assembly is required (e.g., CONDOR, GOBLIN, and BISON). The correlation is in the three-dimensional nuclear simulator, POLCA, through the CONDOR code which is an integral part of POLCA. Since the fuel for which the AA-74 CPR correlation is appropriate will not be utilized in the U.S., application of this correlation in the U.S. is not anticipated.

[ Proprietary Information Deleted ]

ABB engages in continuing programs to update and improve both fuel designs and design methods. As discussed in Reference F9 and the response to Question F3, the NRC is notified of the first application of new fuel designs prior to loading into a reactor. New fuel designs and design features are provided to the NRC for information as supplements to Reference F9. [ Proprietary Information Deleted ]

NRC Question F29

*Discuss the significance of the "(16,36)" point on Figure D.2.2-1.*

ABB Response to Question F29

Figure D.2.2-1 shows a typical Thermal-Mechanical Operating Limit for the SVEA-96 fuel assembly. [ Proprietary Information Deleted ] Development of the Thermal-Mechanical Operating Limit is addressed CENPD-287-P-A (Reference 37).

NRC Question F30

*Discuss the methods by which the general analysis values are selected. Discussion should be provided on a parameter-by-parameter basis.*

ABB Response to Question 30

The question is referring to the analysis values used for the transient analysis; for example, the parameters listed in Table D.6.1.1-1. Table D.6.1.1-1 lists the plant rated conditions and analysis initial and boundary conditions. The analysis conditions are selected based upon the plant heat balance using bounding Technical Specification limits and plant rated conditions. The basis of the analysis initial conditions are discussed below:

- Core Power: For a rated core power (e.g., 3484 MWt), an analysis core power of 102% (e.g., 3553.7 MWt) is used to account for uncertainty in the initial operating power state.

- Core Flow: For a rated core flow (e.g., 13671 kg/s), an analysis core flow of the maximum allowable flow is used. In the example given in Table D.6.1.1-1, the maximum core flow is 106% of rated conditions (e.g., 14491 kg/s).

- Steam / Feedwater Flowrate: From the steam / feedwater flowrate corresponding to rated conditions (e.g., 1884 kg/s), a steam / feedwater flowrate is obtained for the analysis conditions based upon the plant heat balance (e.g., 1926 kg/s).

- Feedwater Temperature: From the feedwater temperature corresponding to rated conditions (e.g., 215 °C), a feedwater temperature is obtained for the analysis conditions based upon the plant heat balance (e.g., 216 °C).

- Vessel Dome Pressure: From the vessel dome pressure corresponding to rated conditions (e.g., 7.136 MPa), a vessel dome pressure is obtained for the analysis conditions based upon the plant heat balance (e.g., 7.152 MPa).

- Core Inlet Enthalpy: From the core inlet enthalpy corresponding to rated conditions (e.g., 1229 kJ/kg), a core inlet enthalpy is obtained for the analysis conditions based upon the plant heat balance (e.g., 1232 kJ/kg).

NRC Question F31

*Discuss ABB's claim on flow dependence of the operating limit set by the recirculation flow controller failure transient, illustrated in Figure D.6.1.9-1 for the higher range of flow near the rated conditions. Discuss* *the details of a "more accurate licensing analysis" regarding the power dependence of the OLMCPR.*

ABB Response to Question F31

The MCPR operating limit to be used at any power and flow condition is the maximum value of: (1) the applicable rated power MCPR limit, (2) the power dependent MCPR limit, and (3) the flow dependent MCPR limit. This procedure ensures that the most limiting value is obtained throughout the power / flow domain.

This procedure is illustrated below for the example provided in Appendix D. [ Proprietary Information Deleted ]

In general, sensitivity studies have demonstrated that GLRNB and FWCF are the only AOOs for the application considered, which could be more limiting at low power conditions than full power conditions. Figure D.6.1.9-2 presents a sample power-dependent MCPR operating limit curve. [ Proprietary Information Deleted ]

The power dependent calculations presented in Figure D.6.1.9-2 are conservative, [ Proprietary Information Deleted ]

NRC Question F32

*The use of PB2 tests to determine methodology bias and bias standard deviation is not acceptable for generic approval. Widen, by use of other test data, the statistical base to a data base sufficient to make this determination over the entire range expected to occur.*

ABB Response to Question F32

Appendix E of this report shows an example application of determining the ABB fast transient analysis methodology bias and bias uncertainty for the Peach Bottom Unit 2 plant. The analysis bias and bias uncertainty can be determined for this plant, since actual plant data is available for three Peach Bottom Unit 2 turbine trip tests. [ Proprietary Information Deleted ]

NRC Question F33

*Explain and justify ABB's decision to "incorporate the thermal-mechanical and nuclear constraints into the MAPLHGR operating limit" to increase margins.*

ABB Response to Question F33

Exposure dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate

(LHGR) operating limits are imposed, respectively, to assure adequate emergency core cooling system (ECCS) performance through compliance with LOCA requirements, and to assure the fuel rod thermal-mechanical design requirements are met.

In some plant applications, fuel rod thermal-mechanical constraints are introduced into the evaluation of the exposure dependent MAPLHGR operating limit. This process does not increase operating margins. It actually reduces the MAPLHGR operating limit from values allowed based solely on emergency core coolant system (ECCS) design acceptance requirements (10 CFR 50.46). It represents a convenience which is sometimes used to avoid a separate LHGR limit. The choice is at the licensee's discretion, as long as the chosen limit(s) ensure compliance with both ECCS and fuel rod design acceptance requirements.

A nodal Average Planar Linear Heat Generation Rate (APLHGR) limit is a fuel rod LHGR limit imposed on a assembly averaged basis. The LHGR and APLHGR are directly related by the bundle nodal local pin peaking factor. Hence, a LHGR limit can be translated into a conservative APLHGR limit by using the upper bound description of the bundle local pin peaking. If the option to combine the limits is implemented, core operation is limited by the lower of the ECCS performance and fuel rod thermal-mechanical design limits.

<u>NRC Question F34</u>

*Since the use of time-varying power distribution in the hot channel calculation relaxes the conservatism existing in the use of a constant power distribution, provide benchmark analysis of ABB's method against other test data to broaden the base.*

<u>ABB Response to Question F34</u>

Historically, transient analyses have been performed with a constant axial power distribution in the hot channel. This methodology has been found to be nonconservative for some cases. Recently, other vendors and utilities have revised their NRC approved methods to address this potential nonconservatism. It is ABB's standard methodology to use the conservative, time varying axial power shape in the hot channel analyses.

Simulating the time variation of the Axial Power Distribution (APD) in the hot channel calculation is conservative with respect to assuming a constant APD. As discussed in Appendix E, Section E.3 of this report, the ABB fast pressurization transient methods simulate the time variation in the hot channel axial power shape.

Table F34-1 shows the results presented in Table D.6.1.8-1 (generated with time varying APD) along with results of the same cases rerun with the assumption of a constant APD. These sensitivity cases demonstrate the strong dependence of the time variation in the APD on the initial control rod insertion. The MOC and EOC cases presented have significantly different initial control fractions and reactivity insertion characteristics. [ Proprietary Information Deleted ] In all cases, the results generated with time varying APDs are more severe than those generated assuming a constant APD. Therefore, it is appropriate to include the effect of time variation in the APD in fast pressurization transient analysis.

## F.3 References

F1. "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB Report CENPD-300-P (proprietary), CENPD-300-NP (non-proprietary), November 1994.

F2. Letter from M. Chatterton (NRC) to D. B. Ebeling-Koning (ABB), "Requests for Additional Information on review of CENPD-300-P, Reference Safety Report for Boiling Water Reactor Reload Fuel'" July 6, 1995.

F3. Thermal-Hydraulic Stability Methodology for Boiling Water Reactors, ABB Report CENPD-295-P-A (proprietary), CENPD-295-NP-A (non-proprietary), July 1996.

F4. Single Loop Operation Analysis for River Bend Station Unit 1, NEDO-31441, General Electric Company, May 1987.

F5. SVEA-96 Critical Power Experiments on a Full Scale 24-rod Sub-Bundle, ABB Report UR-89-210-P-A (Proprietary), UR-89-210-NP-A (non-proprietary), October 1993.

F6. ABB Atom Nuclear Design and Analysis Programs for Boiling Water Reactors: Programs Description and Qualification, BR 91-402-P-A (proprietary), BR 91-403-NP-A (non proprietary), May 1991.

F7. Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification, ABB Report CENPD-284-P-A (proprietary), CENPD-284-NP-A (non-proprietary), July 1996.

F8. United States Federal Register, Vol. 53, No. 180, Friday, September 16, 1988, Rules and Regulations, Page 36000.

F9. Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, ABB Report CENPD-287-P-A (proprietary), CENPD-287-NP-A (non-proprietary), July 1996.

F10. Fuel Rod Design Methods for Boiling Water Reactors, ABB Report CENPD-285-P-A (proprietary), CENPD-285-NP-A (non-proprietary), July 1996.

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**TABLE F1-1**

**CATEGORIES OF INFORMATION CONTAINED IN RSR**

| Section | Title | Category | Additional Comments |
|---|---|---|---|
| 1 | Introduction | 2 | Identifies report purpose |
| 1.1 | Background | 1 | |
| 1.2 | Reload Licensing Document | 1 | |
| 1.3 | Report Overview | 2 | Overview of analysis process |
| 2 | Summary and Conclusion | N/A | |
| 2.1 | Summary | 2 | Summary of fuel design and safety analysis process |
| 2.2 | Conclusions | 2 | Provides ABB conclusions relative to fuel design and safety analysis process |
| 3 | Mechanical Design | 1 | |
| 3.1 | Summary | 2 | Summary of interface with mechanical design LTR |
| 3.2 | Design Criteria | 1 | |
| 3.3 | Design Methodology | 1 | |
| 3.4 | Methodology for Mechanical Design Input to Reload Design and Safety Analysis | 2 | Information transfer between codes |
| 4 | Nuclear Design | N/A | |
| 4.1 | Summary and Conclusions | 2 | Design process overview |
| 4.2 | Design Bases | 3 | Bases for development of nuclear design acceptance limits |
| 4.3 | Reload Nuclear Design Methodology | 2/3/4 | Identifies the design analyses and analysis process, including the development of inputs to the nuclear design codes |
| 4.4 | Nuclear Design Input to Other Disciplines | 2 | Information transfer between code |
| 5 | Thermal Hydraulic Design | N/A | |
| 5.1 | Summary and Conclusions | 2 | Design process overview |
| 5.2 | Design Basis | 3 | Basis for development of thermal hydraulic acceptance limits |
| 5.3 | Methodology for Reload Thermal and Hydraulic Design | N/A | |
| 5.3.1 | Thermal-Hydraulic Design Models | 1 | |
| 5.3.2 | Thermal Design | 3/4/5 | Describes the methodology for establishing the safety limit minimum critical power ratio, including the treatment of the analysis inputs and use of statistical analyses |

## TABLE F1-1 (CONTINUED)

## CATEGORIES OF INFORMATION CONTAINED IN RSR

| Section | Title | Category | Additional Comments |
|---|---|---|---|
| 5.3.3 | Hydraulic Compatibility | 3/4 | Describes the process for demonstrating thermal hydraulic compatibility for the different fuel design, including treatment of analysis inputs |
| 5.4 | Methodology for Thermal Hydraulic Input to Reload Design and Safety Analyses | 1 | |
| 6 | Reload Safety Analysis | N/A | |
| 6.1 | Summary and Conclusions | 2 | Reload analysis process overview |
| 6.2 | Reload Safety Analysis Process | 2 | Description of reload analysis process |
| 6.3 | Reload Safety Analysis Events Assessment | 3 | Assessment of safety analysis events to establish potentially limiting events |
| 6.4 | Design Bases and Acceptance Limits | 3 | Identification of event acceptance criteria and limits |
| 6.5 | Plant Allowable Operating Domain | 4 | Treatment of the allowable operating domain as an input to the safety analysis process |
| 6.6 | Reload Safety Analysis Methodology | 2 | Describes safety analysis codes and methods as they are used in the safety analysis process |
| 7 | Anticipated Operational Occurrences | 2 | Process overview |
| 7.1 | Summary and Conclusions | 3 | Identifies potentially limiting anticipated operational occurrences |
| 7.2 | Design Basis and Acceptance Limits | 3 | Identifies event acceptance limits for anticipated operational occurrences |
| 7.3 | AOO Methodology | 1/2/3/4/5 | Describes the analysis process, including the treatment of event acceptance limits, the development of analysis inputs, and the process for establishing core operating limits. |
| 7.4 | Fast Transient Methodology | 2/4 | Describes the analysis process and methodology for potentially limiting fast transients including the treatment of uncertainties |
| 7.5 | Slow Transient Methodology | 2/4 | Describes the analysis process and methodology for potentially limiting slow transients including the treatment of uncertainties |
| 8 | Accident Analysis | 2 | Process overview |
| 8.1 | Summary and Conclusions | 3 | Identifies potentially limiting accidents |

## TABLE F1-1 (CONTINUED)

## CATEGORIES OF INFORMATION CONTAINED IN RSR

| Section | Title | Category | Additional Comments |
|---|---|---|---|
| 8.2 | Loss of Coolant Accident | 1/2/3/4/5 | Describes the analysis process with references for the loss of coolant accident through the development of core operating limits |
| 8.3 | Control Rod Drop Accident | 1 | |
| 8.4 | Fuel Handling Accident | 2/3/4 | Describes the analysis process for the fuel handling accident, including the event acceptance limits and the development of analysis inputs and the treatment of the analysis results |
| 8.5 | Misplaced Assembly Accident | 1/2/3/4/5 | Describes the analysis process for the misplaced assembly accident through the development of core operating limits |
| 9 | Special Event Analysis | 2/3 | Describes the analysis process and identifies potentially limiting special events |
| 9.1 | Summary and Conclusion | 3 | Describes the objective of the analysis of potentially limiting events |
| 9.2 | Core Thermal-Hydraulic Stability | 1/3 | References applicable Licensing Topical Reports and identifies the event acceptance limits |
| 9.3 | Overpressure Protection | 2/3/4 | Describes the analysis process for the code overpressure protection analysis, including the event acceptance limits and the development of the analysis inputs |
| 9.4 | Standby Liquid Control System Analysis | 2/3/4 | Describes the analysis process for the standby liquid control system analysis, including the event acceptance limits and the development of the analysis inputs |
| 9.5 | Anticipated Transients Without Scram | 2/3/4 | Describes the analysis process for the analysis of anticipated transients without scram, including the event acceptance limits and the development of the analysis inputs |
| Appendix A | Description of Codes | 1 | |
| Appendix B | Plant and Cycle Specific Reload Safety Analysis Summary Report (RSASR) | N/A | Provides a sample format for documenting the results of the reload safety analysis, which illustrates the primary content of the reload safety evaluation documentation |

## TABLE F1-1 (CONTINUED)

## CATEGORIES OF INFORMATION CONTAINED IN RSR

| Section | Title | Category | Additional Comments |
|---|---|---|---|
| Appendix C | Reload Licensing Plant Operating Flexibility | 2/3/4 | Identifies the analysis process for plant operating flexibility options, including the identification of potentially limiting events and the development of analysis inputs |
| Appendix D | Reload Methodology Sample Applications | N/A | Provides examples of typical design and safety analyses and provides an illustration of the application of the methodology. |
| Appendix E | Fast Pressurization Transient Analysis Qualification | N/A | Provides a detailed example of the application of the fast transient analysis methodology, which illustrates the analysis process for potentially limiting events |

## TABLE F2-1

## USE OF KEY UNCERTAINTIES IN MECHANICAL DESIGN DATA

| Analysis Type | Analysis Purpose | Important Mechanical Design Uncertainties | Reference in which uncertainties and Treated |
|---|---|---|---|
| Assembly Design | Mechanical compatibility and evaluation relative to design limits (e.g. stress, strain, fatigue) | Dimensions, material properties, duty cycles | F9 |
| Fuel Rod Design | Establishment of LHGR limits and evaluation relative to fuel rod design limits (e.g. stress, strain, fatigue) | Fuel rod dimensional, characteristics such as internal gas pressure, etc, and code uncertainties | F9 and F10 |
| Safety Limit MCPR (SLMCPR) | Establishment of cladding integrity thermal limits | Channel bow, flow areas | F1 |
| Operating Limit MCPR (OLMCPR) | Establishment of cladding integrity thermal limits | Gap heat transfer coefficient | F1 |
| Loss of Coolant Accident | Demonstration of compliance to emergency core cooling system performance limits and establishment of APLHGR limits | Gap heat transfer coefficient | F1 |
| Control Rod Drop Accident | Demonstration of compliance to acceptance limits | Gap heat transfer coefficient | F7 |
| Thermal-Hydraulic Stability | Demonstration of compliance to acceptance limits and establishment of acceptance power flow operating domain. | Gap heat transfer coefficient | F3 |

**TABLE F34-1**

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**FIGURE F8-1 THROUGH FIGURE F8-3**

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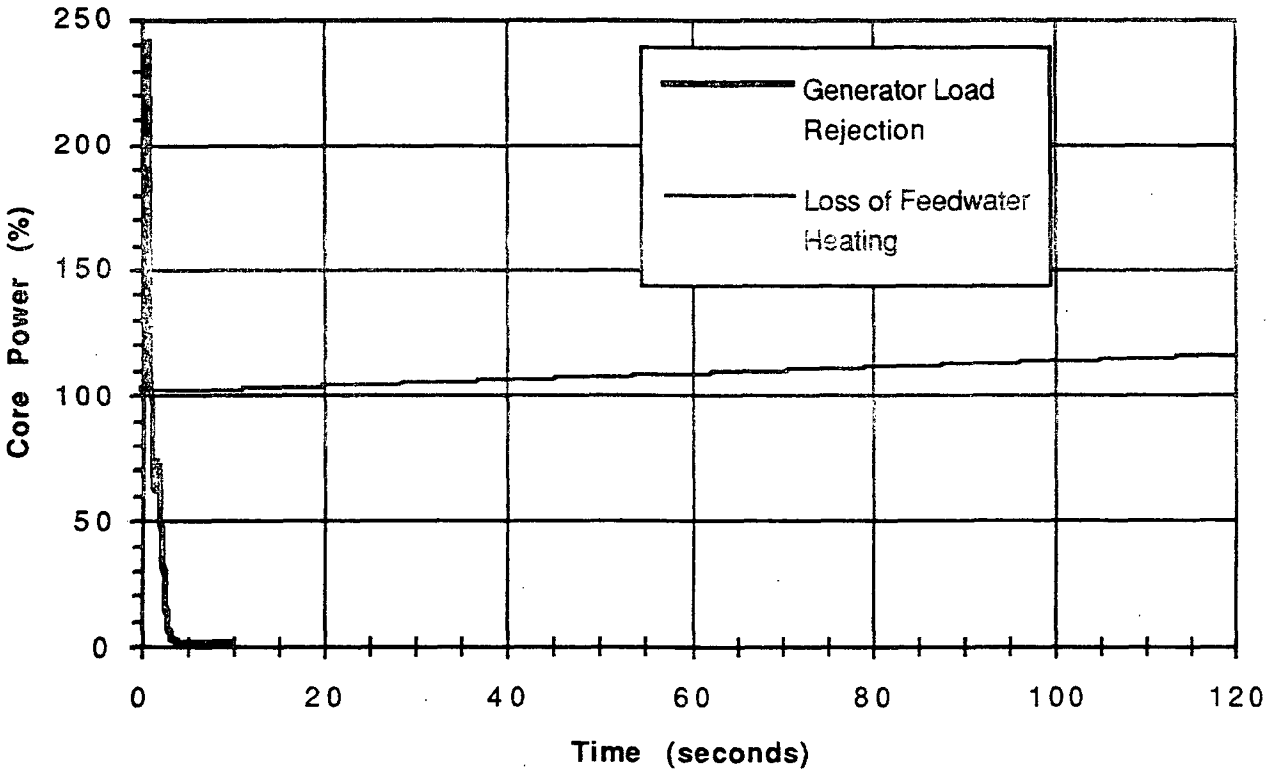


Figure F9-1 Typical Core Power Transient Response for Generator Load Rejection and Loss of Feedwater Heating Events

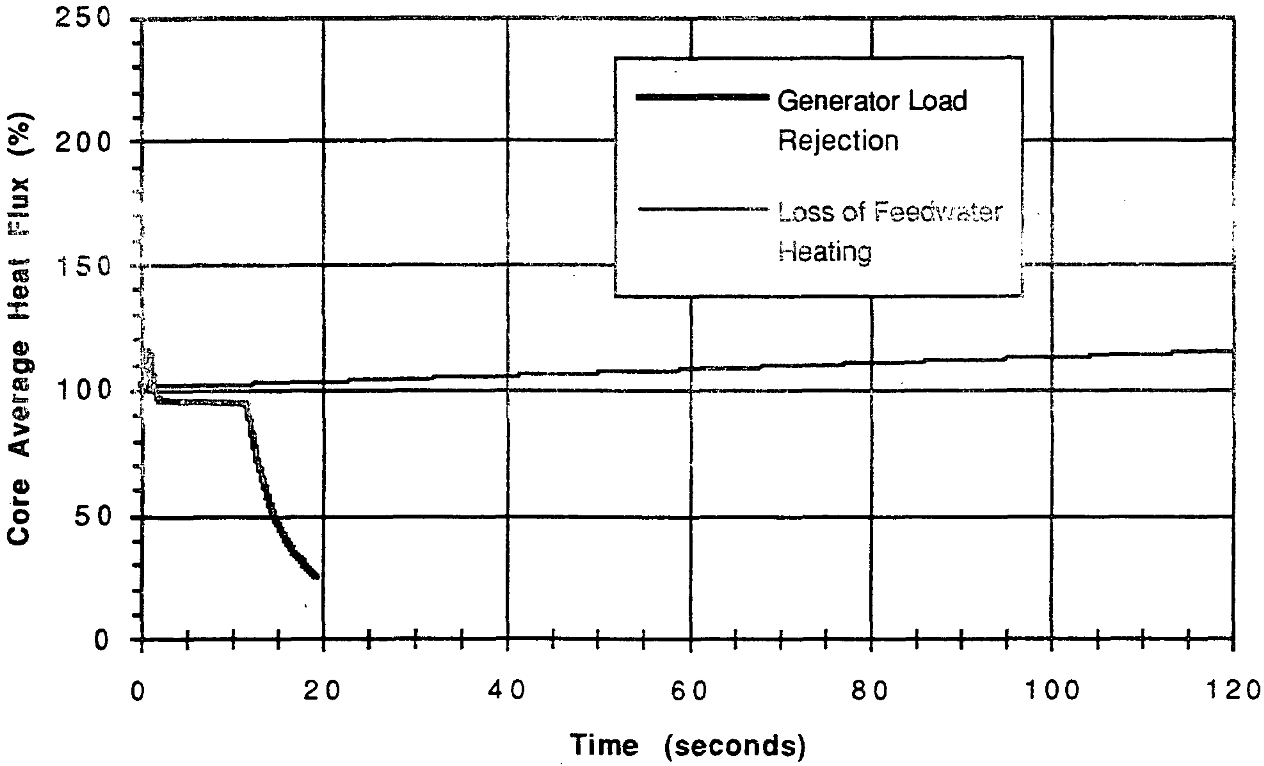


Figure F9-2 Typical Heat Flux Transient Response for Generator Load Rejection and Loss of Feedwater Heating Events

**FIGURE F15-1 THROUGH FIGURE F18-4**

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