Attachment 3

WCAP-17322-NP (proprietary version)

Westinghouse Non-Proprietary Class 3

WCAP-17322-NP
Revision 0

September 2010

Reference Safety Report for Boiling Water Reactor Fuel and Core Analyses
Supplement 1 to CENPD-300-P-A

Westinghouse

Page Intentionally Left Blank

WCAP-17322-NP
Revision 0

Reference Safety Report for Boiling Water Reactor Fuel and Core Analyses Supplement 1 to CENPD-300-P-A

Authors:
Yun Long
Gunilla Norbäck
Angel Enica
Håkan Svensson
John Blaisdell
Camilla Rotander
Andreas Wikström

Reviewed By:
Sven-Birger Johannesson
Patricia Quaglia

Prepared by:
*Ryan Lenahan

Approved by:
*Kris Cummings
Fuel Engineering Licensing

September 2010

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Page Intentionally Left Blank

Executive Summary

Licensing Topical Report (LTR) CENPD-300-P-A, "Reference Safety Report (RSR) for Boiling Water Reactor (BWR) Reload Fuel" (Reference 7) describes the overall BWR reload fuel design and safety analysis process which has been approved for use by Westinghouse (formerly ABB/CE) in specific plant applications.

Throughout Reference 7, numerous references are provided to NRC approved reload analysis methodologies that are documented in other topical reports. Certain other analysis approaches and techniques are also presented in Reference 7. Overall, Reference 7 provides an integrated summary of the applicable portions of the LTRs 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. Further, as provided in Reference 7, 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.

This report is Supplement 1 to CENPD-300-P-A. This report is provided as a supplement to the original LTR (Reference 7) in part due to the length and complex nature of the methodology described and approved in Reference 7. Supplement 1 continues to be presented in a generic manner, and the basic organization and structure provided in the original LTR has not been changed.

The purpose of this report (Supplement 1) is to update and extend the applicability of the Westinghouse BWR reload fuel design and safety analysis process to include use for ABWR plants based on applicable methodologies either approved or now under review by the NRC. This supplement also addresses staff recommended changes including administrative updates needed to reflect revisions and the current application of applicable referenced topicals, to update the owner organization name and, where needed, to clarify existing descriptions by incorporating responses to RAIs, including SER limitations and restrictions, that have been previously approved during the review of Reference 7.

With the addition of Supplement 1, the purpose of the CENPD-300 series of topical reports is to continue to provide an overall description of the complete Westinghouse reload processes and methodologies for use in support of licensing actions, whether it be related to reload fuel (including initial or first core designs), changes to the plant operating domain or equipment performance characteristics.

The original topical report provided references to topical reports that described the licensing basis which was current at that time. As part of this supplement, updated references have been identified and added consistent with the current licensing basis. Additional LTR that have also been identified and listed based on their submittal status to the NRC for review and approval. Acceptability of reload analysis remain subject to conditions cited in the methodology topical reports.

Changes which have been identified as administrative in nature have not been identified in this supplement. In this report administrative changes have been classified as those changes which do not extend or otherwise modify the currently approved methodology. These changes are identified by italics in the attached version. The changes which are classified as administrative include; changing ABB to Westinghouse, editorial changes, changing reload safety analysis to reload licensing analysis, changing three-dimensional nodal simulator to three-dimensional core simulator, and updating references to reflect the current reference section. These changes do not impact the qualification of the original report, and additional justification has therefore not been provided. Administrative changes are identified in Attachment 1 with italics text.

As an aid to staff reviewers and other users of this report; if a section has been unchanged, or only includes administrative changes, then it has not been repeated as part of the proposed supplement. Instead, a combined version of the original and supplement topical has been provided as Attachment 1 which combines all changes and administrative updates that are described in Supplement 1.

TABLE OF CONTENTS

LIST OF ACRONYMS

1D	One-Dimensional
2D	Two-Dimensional
3D	Three-Dimensional
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrences
APLHGR	Average Planar Linear Heat Generation Rate
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRDA	Control Rod Drop Accident
DAR	Design Analysis Record
ECCS	Emergency Core Cooling System
ELLLA	Extended Load Line Limit Analysis
EOC	End of Cycle
EPRI	Electric Power Research Institute
FCPR	Final CPR
FMCRD	Fine Motion Control Rod Drive
gap HTC	Heat Transfer Coefficient between pellet and cladding
GDC	General Design Criteria

LIST OF ACRONYMS (Continued)

HPCI	High Pressure Coolant Injection
ICF	Increased Core Flow
ICPR	Initial CPR
LHGR	Linear Heat Generation Rate
LLLA	Load Line Limit Analysis
LOCA	Loss of Coolant Accident
LPRM	Local Power Range Monitor
LWR	Light Water Reactor
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum CPR
MELLLA	Maximum Extended Load Line Limit Analysis
MEOD	Maximum Extended Operating Domain
MSIV	Main Steam Isolation Valve
NAI	Numerical Applications Incorporated
NRC	Nuclear Regulatory Commission
OLMCPR	Operating Limit MCPR
RBM	Rod Block Monitor
RAI	Request for Additional Information
RCIS	Reactor Coolant Isolation System
RHR	Residual Heat Removal
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RSR	Reference Safety Report

LIST OF ACRONYMS (Continued)

RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
SAFDL	Specified Acceptable Fuel Design Limit
SDM	Shutdown Margin
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit MCPR
SLO	Single Loop Operation
SRP	Standard Review Plan
TIP	Traversing In-core Probe
TMOL	Thermal-Mechanical Operating Limit
TTMOL	Transient TMOL
UNC	Uncertainty
ΔCPR	CPR variation during a transient
ΔMCPR	Minimum CPR variation during a transient

1 INTRODUCTION

Information added to this section comes from the response to RAI F1. It is added to better identify the scope and purpose of this supplement.

This Reference Safety Report (RSR) for boiling water reactor (BWR) reload fuel describes the reload and initial core fuel design and safety analysis process used in specific plant applications. Specific topics related to the *Westinghouse* 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.

The objective of this report is to obtain generic Nuclear Regulatory Commission (NRC) approval for the Westinghouse reload fuel design and safety analysis process that utilizes the Westinghouse 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 Westinghouse 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 Westinghouse 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 licensing 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 Westinghouse uses the current plant licensing basis as an inherent part of the process for updating the plant safety analysis. By using the current plant licensing 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 Westinghouse methodology, because these differences are contained in the current plant licensing basis.

A standard reload of Westinghouse reload fuel is a typical plant change that is not expected to result in 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.

1.1 Background

The following paragraph will be added to this section for clarification:

In April 2000 ABB nuclear businesses were acquired by the parent company of Westinghouse Electric Company (the successor company of the Westinghouse Electric Corporation nuclear business). After this second consolidation, new Licensing Topical Reports were approved by the NRC (References 66 through 76). Quality control, maintenance, and implementation for the complete Westinghouse U.S. BWR reload fuel licensing methodologies resides with the same cognizant organization and persons, now a part of Westinghouse Electric Company.

1.2 BWR Reload Licensing Documents

In order to address SER Condition 1, the following will be added to this section:

All conditions in the referenced licensing reports and the reload methodology will be met during future reload analyses. Compliance with the NRC conditions for each discipline is noted in a relevant Design Analysis Record (DAR).

1.3 Report Overview

Changes were made to this section for clarification purposes.

The last paragraph of this section now reads as follows:

Appendix A to this report provides a brief description of the computer codes used in *Westinghouse* reload analysis methodology. No changes have been made to Appendices B, C, D, E, and F. Therefore they are not attached to this supplemental update.

Table 1-1 and Table 1-2 have been updated in this supplement to remove out of date References and incorporate the location of current methodology and code qualifications. Topicals identified with a -P are currently under review. When issuing the approved version of this report the review status will be updated.

TABLE 1-1

WESTINGHOUSE BWR RELOAD FUEL LICENSING TOPICAL REPORTS

Report Number	Report Title	Discipline
CENPD-285-P-A	Fuel Rod Design Methods for Boiling Water Reactors	Mechanical
WCAP-15836-P-A	Fuel Rod Design Methods for Boiling Water Reactors: Supplement 1	Mechanical
CENPD-287-P-A	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors	Mechanical
WCAP-15942-P-A	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors: Supplement 1	Mechanical
CENPD-288-P-A	ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel	Mechanical
CENPD-390-P-A	The Advanced PHOENIX and POLCA code for Nuclear Design of Boiling Water Reactor	Nuclear
UR 89-210-P-A	SVEA-96 Critical Power Experiments on a Full Scale 24-rod Sub-Bundle	Thermal-Hydraulic
CENPD-389-P-A	10x10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96+	Thermal-Hydraulic
CENPD-392-P-A	10x10 SVEA Fuel Critical Power Experiments and CPR Correlations:SVEA-96	Thermal-Hydraulic
WCAP-16047-P-A	Improved Application of Westinghouse Boiling-Length CPR Correlation for BWR SVEA Fuel	Thermal-Hydraulic
WCAP-16081-P-A	10x10 SVEA Fuel Critical Power Experiments and CPR Correlations:SVEA-96 Optima2	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
WCAP-16606-P-A	Supplement 2 to BISON Topical Report RPA 90-90-P-A	Special Events: ATWS
WCAP-17079-P	Supplement 3 to BISON Topical Report RPA 90-90-P-A SAFIR Control System Simulator	AOO: Fast Transient
WCAP-17202-P	Supplement 4 to BISON Topical Report RPA 90-90-P-A	AOO: Fast Transient
WCAP-17203-P	Fast Transient and ATWS Methodology	AOO: Fast Transient Special Events: ATWS
WCAP-16747-P Appendix C	POLCA-T: Application for Transient Analysis	AOO: Fast Transient
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
WCAP-15682-P-A	Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 2 to Code Description, Qualification and Application	Accidents: LOCA
WCAP-16078-P-A	Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2	Accidents: LOCA
WCAP-16747-P-A, Appendix A	POLCA-T: Control Rod Drop Accident Analysis (CRDA)	Accidents: CRDA
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
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
WCAP-16747-P-A, Appendix B	POLCA-T: Application for Stability Analysis	Special Events: Stability
WCAP 17137-P	Westinghouse Stability Methodology for the ABWR	Special Events: Stability
WCAP-16747-P Appendix D	POLCA-T: Application for Anticipated Transient without Scam Analysis	Special Events: ATWS
WCAP-16608-P-A	Westinghouse Containment Analysis Methodology	Containment Analysis
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 WCAP-15942-P-A (Normal Operation/AOO) CENPD-288-P-A (Accidents)	CENPD-285-P-A WCAP-15835-P-A (Fuel Rod) CENPD-287-P-A WCAP-15942-P-A (Fuel Assembly)	CENPD-285-P-A WCAP-15835-P-A	CENPD-287-P-A WCAP-15942-P-A
Nuclear	CENPD-300-P-A	BR 91-402-P-A (POLCA4) CENPD-390-P-A (POLCA7)	BR 91-402-P-A CENPD-390-P-A	CENPD-300-P-A
Thermal-Hydraulic	CENPD-300-P-A	BR 91-255-P-A, Rev. 1 CENPD-390-P-A UR 89-210-P-A CENPD-389-P-A CENPD-392-P-A WCAP-16047-P-A WCAP-16081-P-A (CPR Correlations)	BR 91-255-P-A, Rev. 1 CENPD-390-P-A UR 89-210-P-A CENPD-389-P-A CENPD-392-P-A WCAP-16047-P-A WCAP-16081-P-A (CPR Correlations)	CENPD-300-P-A
AOO: Fast Transients	CENPD-300-P-A WCAP-17203-P	RPA 90-90-P-A CENPD-292-P-A (BISON) WCAP-16747-P-A (POLCA-T)	RPA 90-90-P-A CENPD-292-P-A WCAP-16606-P-A WCAP-17202-P WCAP-16747-P Appendix C	CENPD-300-P-A WCAP-16747-P Appendix C
AOO: Slow Transients	CENPD-300-P-A	BR 91-402-P-A CENPD-390-P-A	BR 91-402-P-A CENPD-390-P-A	CENPD-300-P-A
Accidents: LOCA	RPB 90-94-P-A CENPD-283-P-A CENPD-300-P-A	RPB 90-93-P-A CENPD-293-P-A WCAP-15682-P-A WCAP-16078-P-A	RPB 90-93-P-A	CENPD-300-P-A
Accidents: CRDA	CENPD-284-P-A	BR 91-402-P-A (RAMONA) WCAP-16747-P-A (POLCA-T)	CENPD-284-P-A WCAP-16747-P-A, Appendix A (POLCA-T)	CENPD-284-P-A WCAP-16747-P-A, Appendix A (POLCA-T)
Accidents: Others	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A

Discipline	Design Bases and Methodology	Code Methods	Qualification	Application
Special Events: Stability	CENPD-300-P-A CENPD-295-P-A WCAP-16747-P-A, Appendix B (POLCA-T) WCAP-17137-P (POLCA-T)	CENPD-294-P-A (RAMONA-3) WCAP-16747-P-A (POLCA-T)	CENPD-294-P-A (RAMONA-3) WCAP-16747-P-A, Appendix B (POLCA-T)	CENPD-300-P-A CENPD-295-P-A WCAP-16747-P-A, Appendix B (POLCA-T) WCAP-17137-P (POLCA-T)
Special Events: Overpressure Protection	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A
Special Events: ATWS	CENPD-300-P-A WCAP-16606-P-A WCAP-17203-P	RPA 90-90-P-A CENPD-292-P-A (BISON) WCAP-16747-P-A (POLCA-T) WCAP-16608-P-A (GOTHIC)	WCAP-16606-P-A (BISON) WCAP-16747-P Appendix D (POLCA-T) WCAP-16608-P-A (GOTHIC)	CENPD-300-P-A WCAP-16606-P-A (BISON) WCAP-16747-P Appendix D (POLCA-T) WCAP-16608-P-A (GOTHIC)

2 SUMMARY AND CONCLUSIONS

2.1 Summary

Only administrative changes were made to this section.

2.2 Conclusions

Only administrative changes were made to this section.

3 MECHANICAL DESIGN

3.1 Summary

Changes were made to this section for clarification purposes. In addition the last paragraph of this section has been added from information from RAI F3.

The *Westinghouse* mechanical design methodology addresses the fuel assembly and fuel rod mechanical evaluation identified in Section 4.2 of the Standard Review Plan, NUREG-0800 (Reference 43). An overview of mechanical design criteria and methodology for the fuel assembly and fuel rod performance analyses is provided in this section.

Detailed methodology is provided in a separate mechanical design methodology topical, Reference 70. Specifically, *Reference 70* contains mechanical design criteria which assure that the requirements of NUREG-0800 (Reference 43) are satisfied, the methodology for performing mechanical design evaluations relative to those criteria, and an application of that methodology to the *Westinghouse* BWR fuel assembly which demonstrates that the fuel assembly satisfies the design criteria.

This chapter also provides the interface between the mechanical design of *Westinghouse* fuel and the other design activities. Specifically, the type of mechanical design and fuel performance 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.

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.

3.2 Design Criteria

Only administrative changes were made to this section.

3.3 Design Methodology

Only administrative changes were made to this section.

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

The following information was added for clarification:

This section provides the interface between the mechanical design of Westinghouse fuel and the other design activities.

3.4.1 Mechanical Design Input to Nuclear Design Analyses

Only administrative changes were made to this section.

3.4.2 Mechanical Design Input to Thermal-Hydraulic Design Analyses

Only administrative changes were made to this section.

3.4.3 Mechanical Design Input to the Transient Analyses

Information about Gap HTC was added for clarification. The Gap HTC paragraph now reads:

[

]a,c

3.4.4 Intentionally Deleted

3.4.5 Mechanical Design Input to LOCA Analyses

Information about Gap HTC was added for clarification. The Gap HTC paragraph now reads:

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. [

]a,c

Detailed methodology for providing Gap HTC to LOCA analyses is discussed in Sections 4.4.4 of Reference 70.

3.4.6 Mechanical Design Input to CRDA Analyses

Information about Gap HTC was added for clarification. The Gap HTC paragraph now reads:

The methodology for analyzing the Control Rod Drop Accident is described in References 33 and 72.

The description in References 33 and 72 include the treatment of mechanical input data, such as gap HTCs, and, therefore, are not repeated in this document.

Methodology for providing Gap HTC to CRDA analyses is discussed in Section 4.4.3 of Reference 70.

3.4.7 Mechanical Design Input to Stability Analyses

Information about Gap HTC was added for clarification. The Gap HTC paragraph now reads:

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. [

]^{a,c} Detailed methodology for providing Gap HTC to stability analyses is discussed in Section 4.4.5 of Reference 70.

4 NUCLEAR DESIGN

4.1 Summary and Conclusions

Outdated information has been removed, and administrative changes have been made to this section.

4.2 Nuclear Design Bases

Changes were made to this section for clarification purposes.

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

4.2.1 Cycle Energy and Fuel Burnup

Only administrative changes were made to this section.

Basis

The nuclear design basis is to install sufficient reactivity in the 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.

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 *Westinghouse* methodology for evaluating conformance to this design basis is discussed in Section 4.3.

4.2.2 Reactivity Coefficients

Only administrative changes have been made to this section.

4.2.3 Control of Power Distribution

Only administrative changes have been made to this section.

4.2.4 Shutdown Margin

Only administrative changes have been made to this section.

4.2.5 Stability

Only administrative changes have been made to this section.

4.3 Nuclear Design Methodology

4.3.1 Reference Core

Only administrative changes have been made to this section.

4.3.1.1 Bundle Design Cross Section Calculations

Text on burnable absorbers has been modified. Items 1 and 2 now read:

[

]^{a,c}

4.3.1.2 Loading Pattern and Control Rod Sequences

Outdated text discussing the Shuffling-File Processing code and the mono-sequence control rod patterns was removed. No other changes were made to this section.

4.3.1.3 Deviations from the Reference Core

The only clarification change to this section occurs In Item 4 of the assembly inventory discussion, which now reads:

[

]^{a,c}

[

]a,c

4.3.1.4 Reload Cycle Design Model

Only administrative changes were made to this section.

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

4.3.2.1 Cycle Energy and Fuel Burnup

Changes were made to this section for clarification purposes.

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 *Westinghouse* 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 Section 3.

4.3.2.2 Reactivity Coefficients

Only administrative changes were made to the general discussion of Reactivity Coefficients.

<u>Moderator Void Reactivity Coefficient</u>

The only change to the Moderator Void Reactivity Coefficient discussion occurs in point a) which now reads as follows:

[

]a,c

Doppler Coefficient of Reactivity

No changes were made to the Doppler Coefficient discussion.

Delayed Neutron Fractions

No changes were made to the Delayed Neutron Fraction discussion.

Inverse Velocities and Prompt Neutron Lifetimes

No changes were made to the Inverse Velocities and Prompt Neutron Lifetimes discussion.

Energy Deposition Fractions

No changes were made to the Energy Deposition Fractions discussion.

4.3.2.3 Control of Power Distribution

Changes were made to this section for clarification purposes.

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 feed fuel bundle and Reference Core loading 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 70*, a Thermal-Mechanical Operating Limit (TMOL) is established for which all mechanical design bases are satisfied [

]a,c Confirmation that the TMOL is not exceeded demonstrates that all mechanical fuel rod design bases are satisfied. [

]a,c

[

]^{a,c}

(3) The feed fuel bundle and Reference Core loading 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. [

]^{a,c}

(4) The reload feed fuel 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 MAPLHGR limits at each statepoint to confirm that the design provides sufficient margin to assure that the MAPLHGR limits will not be approached during normal operation in the plant application.

4.3.2.4 Shutdown Margin

Outdated information was removed, and administrative changes have been made to this section.

4.4 Nuclear Design Input to Other Disciplines

4.4.1 Nuclear Design Input to Mechanical Design

Changes were made to this section for clarification purposes. Outdated information was removed from this section.

<u>Methodology</u>

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 70*. These calculations are performed to confirm that the TMOL is in fact bounding for a specific application. [

]^{a,c}

[

]a,c

Discussion

An example of the selection of limiting fuel rods and the resulting power histories is provided in Section 4.3.0 of *Reference 70.*

4.4.2 Nuclear Design Input to Thermal-Hydraulic Design

Changes were made to this section for clarification purposes.

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 *as* discussed in Section 4.3.1. The term "conservative" refers in this case to selecting 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. [

]a,c

4.4.3 Nuclear Design Input to Transient Analyses

Changes were made to this section for clarification purposes.

The AOOs discussed in Section 7 can be categorized as "fast" or "slow". The slow 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 core simulator model discussed in Section 4.3.1.

The current NRC approved codes to analyze fast transients are shown in Appendix A.

4.4.3.1 1D Kinetics and Average Channel Analysis Model

This section title was added for clarification between 1D and 3D. Changes were made to this section for clarification purposes.

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. [

]^{a,c}

4.4.3.2 3D Kinetics and Parallel Channel Model

This section was added to capture 3D codes that have been developed since the original submittal of the topical in 1996.

If the fast transients are analyzed using a *Westinghouse* NRC approved 3D kinetics dynamic analysis code with parallel channels, all nuclear and thermal-hydraulic data simulating the three-dimensional situation can be taken directly from the static core simulator and no collapsing is needed. Current 3D kinetics and static core simulator codes used in this analysis are presented in Appendix A.

4.4.4 Nuclear Design Input to the Accident Analyses

4.4.4.1 Nuclear Design Input to LOCA Analyses

Changes were made to this section for clarification purposes.

The LOCA analysis methodology is described in Section 8.2. 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. [

]^{a,c}

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

[

]^{a,c}

4.4.4.2 Nuclear Design Input to CRDA Analyses

Changes were made to this section for clarification purposes.

Section 8.3 describes *Westinghouse* CRDA methodology. The 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. [

]^{a,c}

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. [

]^{a,c}

4.4.4.3 Nuclear Design Input to Fuel Handling Accident Analyses

Only administrative changes were made to this section.

4.4.4.4 Mislocated and Rotated Fuel Assembly Analyses

Only administrative changes were made to this section.

4.4.5 Nuclear Design Input to Special Events Analyses

4.4.5.1 Stability Analysis

The discussion of Frequency Domain Methodology was removed from this supplement as it is no longer part of current methodology.

As discussed in Section 9.2 the *Westinghouse* stability analysis methodology utilizes time domain codes.

The stability evaluation is performed with the three dimensional systems transient codes as described in Section 9.2.2. 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. [

]^{a,c}

4.4.5.2 Overpressurization Protection

No changes were made to this section.

4.4.5.3 Standby Liquid Control System

Only administrative changes were made to this section.

a, c

Figure 4-1 Iterative Process for Determining Reference Core Design

a, c

Figure 4-2 Data Flow to 3-D Transient Systems Code

5 THERMAL-HYDRAULIC DESIGN

Throughout this Chapter, editorial changes were made to correct typos so now all instances read "thermal-hydraulic" and "inter-assembly."

5.1 Summary and Conclusions

5.1.1 Summary

Outdated information was removed, and administrative changes have been made to this section.

5.1.2 Conclusions

Changes were made to this section for clarification purposes.

The information contained in this section supports the following conclusions regarding the *Westinghouse* thermal-hydraulic methodology and the thermal-hydraulic characteristics of the fuel assemblies:

(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-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 *Westinghouse* as well as non-*Westinghouse* fuel is acceptable for design and licensing applications.

5.2 Thermal-Hydraulic Design Bases

Changes were made to this section for clarification purposes.

The principal objective of the thermal-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 the 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

Changes were made to this section for clarification purposes.

Basis

The minimum value of the 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 in Section 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 liquid film on the most limiting rod locally has completely evaporated causing a rapid loss of heat transfer capability in that fuel rod for a given pressure, flow, inlet enthalpy and axial power shape. This critical heat flux 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 critical power. 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 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 Report ("COLR")*. The treatment of MCPR for *Westinghouse* and non-*Westinghouse* fuel to assure that the OLMCPR is satisfied during reactor operation and during the design phase *is* discussed in Section 5.3.

5.2.2 Hydraulic Compatibility

Outdated information has been removed, and administrative changes have been made.

5.2.3 Bypass, Water Rod and Water Cross Flow

Changes were made to this section for clarification purposes.

Basis

The fuel assembly shall be designed to maintain the *inter-assembly* 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 fuel is maintained such that significant boiling will not occur.

Discussion

The Design Basis in Section 5.2.2 addresses *inter-assembly* bypass flow to assure acceptable flow distributions. This design basis is intended to assure that sufficient *inter-assembly* 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 *are* maintained and that there is sufficient cooling flow to the in-core nuclear instrumentation. 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 *inter-assembly* bypass and interior assembly flow channels is provided in Section 5.3.4.

5.3 Methodology for Thermal-Hydraulic Design

5.3.1 Thermal-Hydraulic Design Models

Changes were made to this section for clarification purposes.

Accurate computer models simulating the thermal-hydraulic behavior of the plant and the different types of fuel assemblies in the core are established for the following purposes:

(1) Evaluate and establish thermal-hydraulic compatibility of the Reload fuel with the Resident fuel and the core, (if not first core fuel)

(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.

[

]a,c

Computer codes accepted for licensing applications by the NRC are used for all thermal-hydraulic analyses. The steady-state thermal hydraulics performance models are incorporated into the *Westinghouse* BWR three dimensional *core* simulator discussed in Appendix A.

5.3.1.1 Core and Assembly Models

Changes were made to this section for clarification purposes.

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 *inter-assembly* 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-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 (assembly outlet) represents the upper tie plates or outlet spacers (downstream the active fuel zone) and section of the channel above the upper tie plates or outlet spacers, including handle.

The core inlet orifice, bottom nozzle, lower tie plate, spacer grid, assembly outlet, 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 *inter-assembly* 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 *inter-assembly* 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.

[

]a,c

5.3.1.2 Plant and Resident Fuel Hydraulic Data

Only administrative changes were made to this section.

5.3.1.3 Hydraulic Data for Westinghouse Fuel

Changes were made to this section for clarification purposes. Outdated information has been removed.

Extensive test loop data are used to verify the validity of the analytical modeling of the *Westinghouse* fuel.

Specifically, test data are used to verify the modeling of SVEA-type design water cross and water wing modeling, design of bypass holes, tie plates, spacers, and flow distribution to the SVEA-type design sub-bundles as well as friction pressure drop multipliers. *Westinghouse* 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 *Westinghouse* test program is provided by the hydraulic testing of various *Westinghouse* bundle designs summarized in Tables 5-1, 5-2 and 5-3.

5.3.2 Thermal Design

5.3.2.1 Safety Limit Minimum Critical Power Ratio

Information has been added to extend the generic methodology for SLMCPR of mixed cores. This sections methodology on mixed cores safety limit has been modified to account for cycle specific variations for non-Westinghouse fuel in the determination of SLMCPR.

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 design specific, the methodology is acceptable for design and licensing purposes for all BWR cores containing *Westinghouse* fuel, as well as for mixed cores containing both *Westinghouse* and non-*Westinghouse* fuel assemblies, provided adequate input data are available.

For *Westinghouse* 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 phenomenological-based expression to critical power data. The SLMCPR is established to protect the fuel from reaching critical power during steady state operation and anticipated transients. The SLMCPR is established to provide that at least 99.9% of the fuel rods avoid reaching critical power.

<u>Methodology</u>

[

]a,c

[

]a,c

Discussion

[

]a,c

5.3.2.2 Monte Carlo Safety Limit Evaluation

Changes have been made to this section for clarification purposes

[

]a,c

[

]a,c

Methodology

Information added to the Methodology discussion was originally presented in response to RAI F-11.

[

]a,c

[

$]^{a,c}$

Mixed core discussion has been moved to Section 5.3.2.1.

5.3.2.3 Channel Bow Evaluation

No changes were made to this section.

5.3.2.4 Minimum Critical Power Evaluation for Reload Fuel

Changes were made to this section for clarification purposes.

For reload applications, a *Westinghouse* CPR 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 Optima2 assembly currently being marketed in the U.S. for BWR applications has been accepted by the NRC and is documented in *Reference 66.*

5.3.2.5 Minimum Critical Power Evaluation for Resident Fuel

Changes were made to this section for clarification purposes. Outdated information has been removed.

Methodology

If the Resident fuel is a *Westinghouse* design, the CPR is treated in the same manner as for the Reload fuel assembly. A *Westinghouse* CPR 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 a *Westinghouse* design, a CPR 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 a *Westinghouse* design, *Westinghouse* may or may not have direct access to the accepted correlation for the Resident fuel. If *Westinghouse* does have direct access to that correlation, it is used for design and licensing analyses. [

]^{a,c}

[

]a,c

Discussion

[

]a,c

5.3.3 Hydraulic Compatibility

Changes were made to this section for clarification purposes. The only change occurred in point 2 which now reads:

[

]a,c

5.3.4 Bypass, Water Cross, and Water Rod Flow

Changes were made to this section for clarification purposes.

The bypass flow fraction is a function of the size of the bypass flow holes in the bottom nozzle.

[

]a,c

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

5.4.1 Thermal-Hydraulic Design Input to Mechanical Design

Only administrative changes were made to this section.

5.4.2 Thermal-Hydraulic Design Input to Nuclear Design

Only administrative changes were made to this section.

5.4.3 Thermal-Hydraulic Design Input to Transient Analyses

No changes were made to this section.

5.4.4 Thermal-Hydraulic Design Input to LOCA Analyses

No changes were made to this section.

5.4.5 Thermal-Hydraulic Design Input to CRDA Analyses

Only administrative changes were made to this section.

5.4.6 Thermal-Hydraulic Design Input to Stability Analyses

Changes were made to this section for clarification purposes. Outdated information has been removed.

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.

[

]a,c

TABLE 5-1

SUMMARY OF THE SVEA-64 THERMAL-HYDRAULIC TEST PROGRAM

a,c

TABLE 5-2

SUMMARY OF THE SVEA-96 AND SVEA-100 THERMAL-HYDRAULIC TEST PROGRAM

a,c

TABLE 5-3

SUMMARY OF THE SVEA-96 OPTIMA, SVEA-96 OPTIMA2 AND SVEA-96 OPTIMA3 THERMAL-HYDRAULIC TEST PROGRAM

a,c

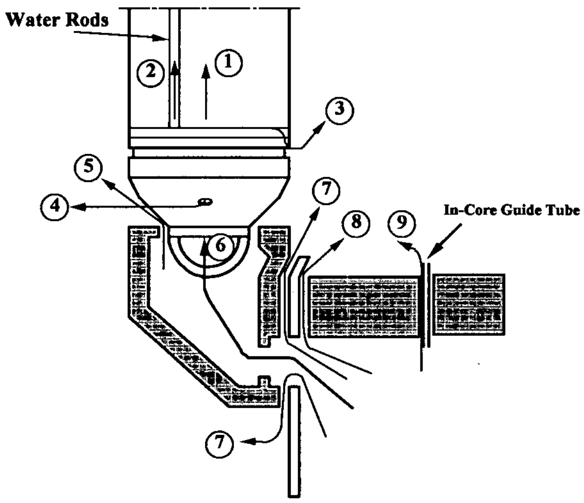
TABLE 5-4

EXAMPLE OF UNCERTAINTIES FOR SLMCPR CALCULATIONS

[ ]a,c

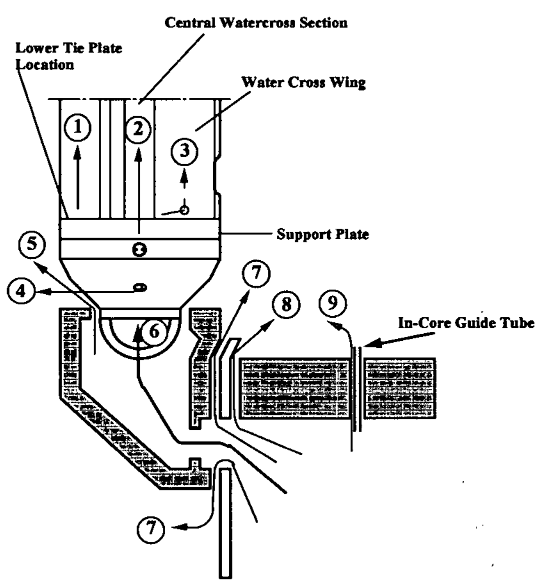
[ ]a,c

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. The schematic for SVEA-96 apply to all SVEA-96 versions, including SVEA-96 Optima2.

a,c

Figure 5-2 Calculation Scheme Monte Carlo Safety Limit Methodology

6 RELOAD *LICENSING* ANALYSIS

Changes have been made to this chapter to extend the applicability of the reload methodology to include the ABWR. The methodology contained in this topical remains unchanged as it was found to be applicable to the ABWR.

6.1 Summary and Conclusions

Only administrative changes have been made to this section.

6.2 Reload *Licensing* Analysis Process

Only administrative changes were made to this section.

6.3 Reload Safety Analysis Events Assessment

Changes were made to this section for clarification purposes.

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 set of potentially limiting events for the typical plant safety analysis that can be impacted by a reload application or a plant operational modification are identified. This subset of potentially limiting events is evaluated as a part of the plant-specific reload licensing 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 *Westinghouse* 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. A road map of the process is given in Figure 6-2.

Each of the identified events is evaluated for the first *Westinghouse* reload application and for each subsequent reload if an applicable generic or bounding analysis is not available. In addition, *Westinghouse* 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, *Westinghouse* 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 a new fuel type 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.

6.3.1 Event Categorization

Changes were made to this section for clarification purposes.

As discussed in Section 6.2, the plant safety analysis contains the evaluation of a wide spectrum of postulated events and is 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 *Westinghouse* 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 *Westinghouse* reload *licensing* analysis process. The generic safety analysis events that are covered in the *Westinghouse* reload *licensing* 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 *Westinghouse* methodology. The results of these evaluations are included in the plant specific reload safety evaluation.

In addition, the plant safety analysis is reviewed to identify any events different than those generic events identified in Table 6-1 which may be potentially limiting. Potentially limiting events from this additional subset of events, along with their plant-specific commitments, are also included in the reload licensing evaluation.

The next three sections discuss the categorization of events in the three groups: Anticipated Operational Occurrences, Accidents, and Special Events. Section 6.3.2 summarizes the methodology for determining the potentially limiting events to be analyzed for the introduction of *Westinghouse* fuel or a plant modification.

6.3.1.1 Anticipated Operational Occurrences

Changes were made to this section for clarification purposes, as well as to extend applicability to the ABWR.

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.

<u>Single Failures as Initiating Events</u>

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) Failure to open or close on demand of any single valve (a check valve is not assumed to close against normal flow).

(2) Failure to start or stop on demand of 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.

<u>Reactor Vessel Pressure Increase Events</u>

Only administrative changes were made to the Reactor Vessel Pressure Increase Events sub-section.

Reactor Core Coolant Temperature Decrease Events

Decrease in core coolant temperature includes 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 inadvertant HPCI system start) can lead to significant changes in critical power ratio (CPR). [

]^{a,c} Therefore, evaluation of the loss of feedwater heater in the *Westinghouse* reload *licensing* 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 *Westinghouse* reload *licensing* analysis process.

Reactor Core Positive Reactivity Insertion Events

Only administrative changes were made to the Reactor Core Positive Reactivity Insertion Events sub-section.

Reactor Vessel Coolant Inventory Decrease Events

Only administrative changes were made to the Reactor Vessel Coolant Inventory Decrease Events sub-section.

Reactor Core Coolant Flow Decrease Events

Only administrative changes were made to the Reactor Core Coolant Flow Decrease Events sub-section.

Reactor Core Coolant Flow Increase Events

Only administrative changes were made to the Reactor Core Coolant Flow Increase Events sub-section.

Reactor Core Coolant Temperature Increase Events

Only administrative changes were made to the Reactor Core Coolant Temperature Increase Events sub-section.

Reactor Vessel Coolant Inventory Increase Events

Only administrative changes were made to the Reactor Vessel Coolant Inventory Increase Events sub-section.

6.3.1.2 Design Bases Accidents

Changes were made to this section for clarification purposes.

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

Single Failure in Accidents Evaluation

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 radiological consequences of the spectrum of postulated pipe break locations *are* 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 reactor scram occurs as the MSIVs close and the release of radioactivity is terminated when the MSIVs are fully closed.

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 *Westinghouse* reload *licensing* 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 largest connected piping system. 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 *Westinghouse* reload *licensing* analysis process.

<u>Control Rod Drop Accident</u>

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 categorized as Accidents.

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 *Westinghouse* reload applications. If required, plant safety analysis is modified to reflect the radiological consequences of the accident. In the *Westinghouse* 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

Only administrative changes were made to the Fuel Handling Accident sub-section.

Fuel Loading Errors

Only administrative changes were made to the Fuel Loading Errors sub-section.

Recirculation Pump Failure Accident

Only administrative changes were made to the Recirculation Pump Failure Accident sub-section.

Instrument Line Breaks

Only administrative changes were made to the Instrument Line Breaks sub-section.

6.3.1.3 Special Events

Only administrative changes have been made to this section.

Core Thermal-Hydraulic Stability

Only administrative changes were made to the Core Thermal-Hydraulic Stability sub-section.

Reactor Overpressure Protection

Only administrative changes were made to the Reactor Overpressure Protection sub-section.

Shutdown Without Control Rods

Only administrative changes were made to the Shutdown Without Control Rods sub-section.

Anticipated Transients Without Scram

In order to capture this event for all reactor types, additional discussion was added to the last paragraph of this section, which now reads as follows:

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 for BWR/2 to BWR/6. For ABWR plants the reactor shutdown is accomplished by automatic initiation of the standby liquid control system or the electrical insertion of the control rods via the fine motion control rod drive system.

6.3.2 Potentially Limiting Events

Changes were made to this section for clarification purposes.

Not all of the plant's safety analysis events are required to be reanalyzed 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 *Westinghouse* reload *licensing* analysis process.

To identify the potentially limiting events, each event in the plant safety analysis is evaluated to determine that, for a *Westinghouse* 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, *Westinghouse* has developed a process to determine the potentially limiting events that assure coverage of all applicable potentially limiting events. 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 *Westinghouse* reload application or change to plant operating domain justified by the use of *Westinghouse* 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 *Westinghouse* 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. [

]^{a,c}

The loss of coolant accident is evaluated for the initial application of *Westinghouse* reload fuel and then only supplemented to establish the core operating limits associated with new fuel types. [

]^{a,c} 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. [

]^{a,c}

6.4 Design Bases and Acceptance Limits

Only administrative changes were made to this section.

6.4.1 Anticipated Operational Occurrences

Changes were made to this section for clarification purposes.

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.

<u>Radioactive Effluents</u>

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) reactor isolation at power operation (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 rods 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

Only administrative changes were made to the Specified Acceptable Fuel Design Limits sub-section.

Peak Reactor Vessel Pressure

Only administrative changes were made to the Peak Reactor Vessel Pressure sub-section.

Suppression Pool Temperature

Only administrative changes were made to the Suppression Pool Temperature sub-section.

6.4.2 Design Bases Accidents

Changes were made to this section for clarification purposes.

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 consequences. The event acceptance limit for offsite radiological consequences 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 consequences; (2) the ECCS acceptance criteria of 10CFR50.46 (Reference 42); and (3) the primary containment design limits.

The event acceptance limit for offsite radiological consequences is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological consequences 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. [

]^{a,c}

Control Rod Drop Accident

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

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

The limits for the control rod drop accident are described in a code specific topical. If no limits are defined in the code specific topical the defined limits are the current interim limits. Once the final limits are defined by the NRC they will be followed.

Fuel Handling Accident

For the fuel handling accident, the figures of merit are the onsite and offsite radiological consequences. The event acceptance limit for offsite radiological consequences is well within (25% or less) the guideline dose value of 10CFR100, and the event acceptance limit for onsite radiological consequences is the limit identified in GDC 19.

Fuel Loading Error

No changes were made to the Fuel Loading Error sub-section.

Recirculation Pump Failure Accident

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

Instrument Line Break

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

6.4.3 Special Events

Changes were made to this section for clarification purposes.

As described above the event acceptance limits for special events are dependent on the specific event being analyzed. The specific events 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

No changes were made to the Core Thermal-Hydraulic Stability sub-section.

Overpressure Protection

Only administrative changes were made to the Overpressure Protection sub-section.

Shutdown Without Control Rods

No changes were made to the Shutdown Without Control Rods sub-section.

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 consequences; 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 consequences is the guideline dose values of 10CFR100.

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

Changes were made to this section for clarification purposes.

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 *Westinghouse* reload *licensing* 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, MELLLA, MELLLA+), 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 *Westinghouse* reload *licensing* analysis process for *a* 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 *the* 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

Only administrative changes were made to this section.

6.6.1 Methods and Analyses

Changes were made for clarification purposes.

The primary methods used in the overall reload safety analysis process include: (1) the lattice physics nuclear design methods; (2) the 3D *core* 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 *licensing* 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.

Fuel Assembly and Core Design

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 *core* 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 the ECCS evaluation.

The 3D *core* 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 *core* 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 *core* simulator is used as a part of the nuclear and thermal-hydraulic 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 CPR correlation as input, the 3D *core* simulator is used to predict the anticipated MCPR throughout the operating cycle.

Static and Quasi Steady-State Transient Events

Only administrative changes were made to the Static and Quasi Steady-State Transient Events sub-section.

Dynamic Transient Events

No changes were made to the Dynamic Transient Events sub-section.

LOCA

The results of the LOCA analysis are required to demonstrate compliance to the ECCS acceptance limits. The LOCA analysis is performed using an approved ECCS evaluation model, which requires detailed inputs to describe the reactor pressure vessel internals, the reactor protection system, the performance of the ECCS equipment and its actuation, fuel performance parameters, and rod peaking parameters. The LOCA analysis inputs make use of a conservative power operating history to develop the fuel performance parameters and a conservative MCPR operating limit to establish conservative boundary conditions for the heat-up calculation. The heat-up analysis establishes the maximum average planar linear heat generation rate (MAPLHGR) operating limits, which ensures compliance with the ECCS acceptance limits during plant operation.

6.6.2 Operating Limits

Changes were made to this section for clarification purposes.

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, manufacturing uncertainties, and a conservative core power distribution as inputs. The operating limit MCPR is established such that the transient CPR will not decrease below the safety limit MCPR. In establishing the operating limit MCPR, the ΔCPR for the AOOs 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.

[

]^{a,c}

6.6.3 Input Data

Only administrative changes were made to this section.

6.6.4 Reload Safety Evaluation Confirmation

Only administrative changes were made to this section.

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)

Anticipated Transients without Scram

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/g
- 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/g
- Fuel Handling Accident
  - Offsite Dose ≤ Well within the 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/g

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

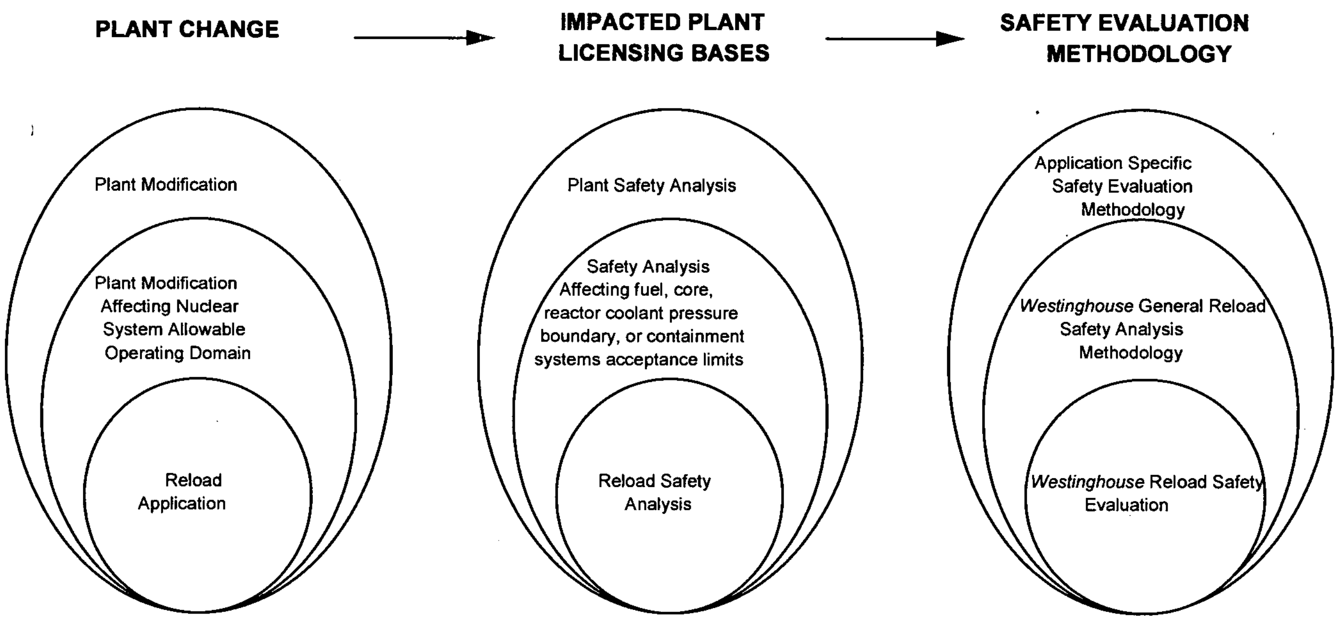


Figure 6-1 Safety Evaluations Process for Plant Modifications

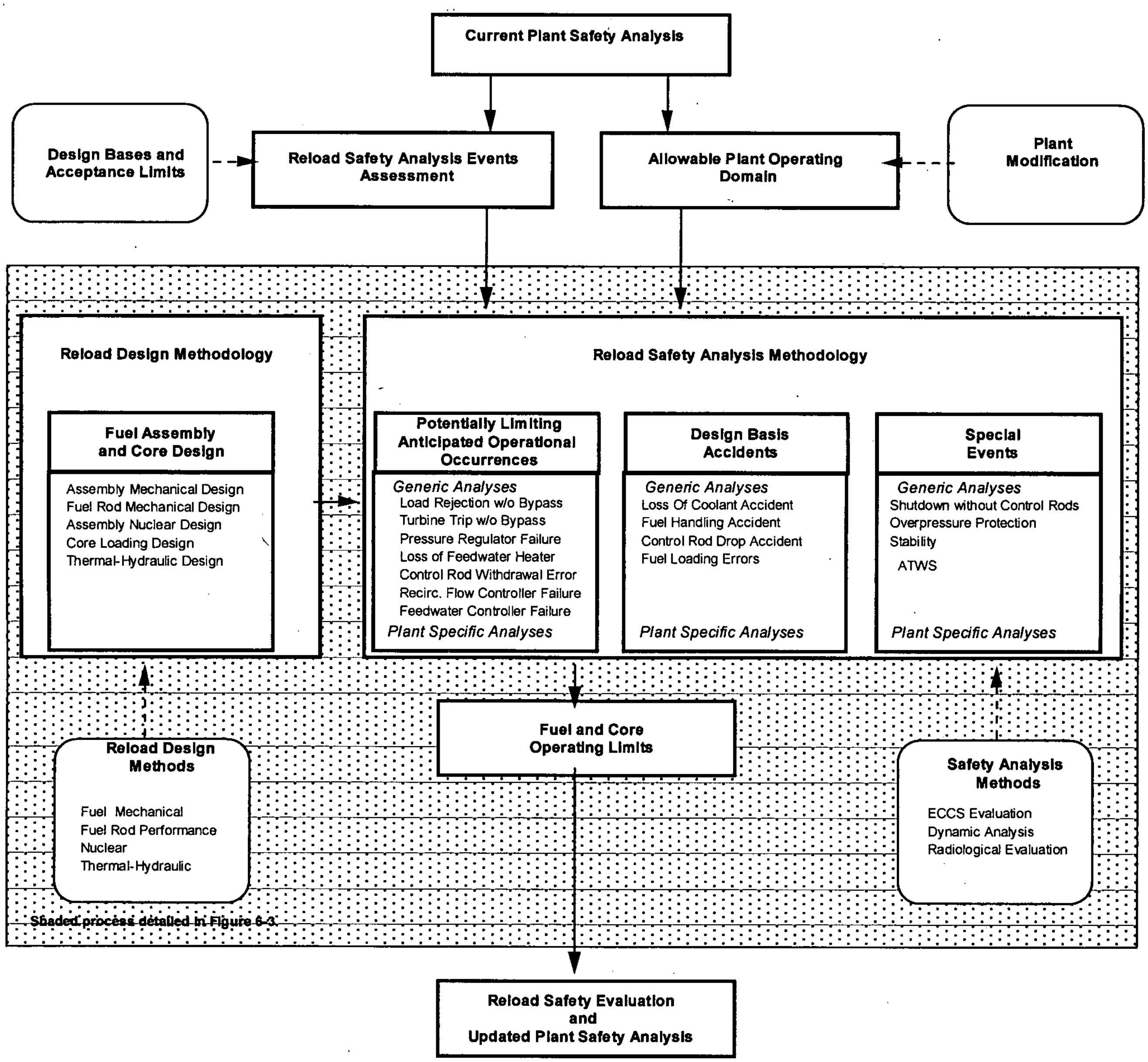


Figure 6-2 Overall Reload Safety Analysis Process

a,c

Figure 6-3 Reload Safety Analysis Methodology Flow Chart

7 ANTICIPATED OPERATIONAL OCCURRENCES (AOO)

Only administrative changes were made to this section. Outdated information in the tables at the end of this section have been removed.

7.1 Summary and Conclusions

Only administrative changes were made to this section.

7.2 Design Bases and Acceptance Limits

No changes were made to this section.

7.2.1 Core Design Cladding Integrity

Changes were made to this section for clarification purposes.

Basis

The minimum allowed value of the Critical Power Ratio (CPR), denoted MCPR, is established such that at least 99.9% of the fuel rods in the core with 95% probability and 95% confidence would not be expected to experience boiling transition during normal operation or anticipated operational occurrences.

Discussion

The acceptance limit for this design criterion 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 core 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 core design and safety analysis is provided in Section 5.2.1.

7.2.2 Fuel Design Cladding Integrity

Changes were made to this section for clarification purposes.

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

[

]a,c

[

]a,c

7.3 AOO Methodology

Only administrative changes were made to this section.

7.3.1 AOO Events and Analysis Method

Information added to the Methodology section comes from the response to RAI-F9.

<u>Methodology</u>

Table 6-2 of Section 6 listed the potentially limiting AOO events evaluated in the *Westinghouse* 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. These events typically result in a scram being initiated on either the event initiator (e.g., valve position detection) or high neutron flux. "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 and Sections 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

No changes were made to this section.

7.3.3 Analyses Calculational Uncertainty

Changes were made to this section for clarification purposes.

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. [

]a,c

In addition to the treatment of uncertainties described below a Monte Carlo based uncertainty analysis method (Reference 73) has been submitted to the NRC for review and approval. Once approved, this new method may be used for the determination of the uncertainties in transient analysis.

To remain in compliance with SER Condition 2 of CENPD-300 Revision 0 Method A remains generically applicable for use in OLMCPR uncertainty determinations. Methods B, C, and D are not generically approved for use in the determination of OLMCPR uncertainty. Use of Methods B, C, or D for OLMCPR uncertainty determination needs to be sufficiently justified in a site specific application.

7.3.3.1 Treatment of Analysis Uncertainty

Changes were made to this section for clarification purposes. The discussion section of Approach A now reads as follows:

[

]a,c

[

]a,c

7.3.3.2 Slow Transient Analysis Uncertainty

No changes were made to this section.

7.3.3.3 Fast Pressurization Transient Analysis Uncertainty

Only administrative changes were made to this section.

7.3.4 Fuel and Core Operating Limits

No changes were made to this section.

7.3.4.1 MCPR Operating Limit

No changes were made to this section.

7.3.4.2 LHGR Operating Limit

No changes were made to this section.

7.4 Fast Transient Methodology

Changes were made to this section for clarification purposes. Redundant information has been removed from this section, and is now presented in WCAP-17203.

CENPD-300 Revision 0 described the fast transient methodology for the limiting transients identified in Section 7.3.1:

- Generator Load Rejection Without Bypass
- Turbine Trip Without Bypass
- Feedwater Controller Failure - Maximum Demand
- Pressure regulator Failure - Closed

For addressing all fast transients required for a first core application, i.e. both limiting and non-limiting transients, a new Licensing Topical Report has been submitted for NRC review (Reference 73). For the fast transients listed above the methodology is consistent with Revision 0 of this topical report (CENPD-300-P-A, Revision 0).

The methodology described in Reference 73 is code-independent and is applicable to both 1D and 3D transient analysis codes described in Appendix A.

7.5 Slow Transient Methodology

No changes were made to this section.

7.5.1 Analysis Codes

Changes were made to this section for clarification purposes.

7.5.2 Analysis Calculational Procedure

Only administrative changes were made to this section.

7.5.3 Recirculation Flow Controller Failure - Increasing Flow

7.5.3.1 Event Description

No changes were made to this section.

7.5.3.2 Analysis Methodology

Only administrative changes were made to this section.

7.5.4 Rod Withdrawal Error

7.5.4.1 Event Description

Changes were made to this section for clarification purposes.

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. Newer boiling water reactors such ABWR are equipped with a redundant automated thermal limit monitor (ATLM) system.

7.5.4.2 Analysis Methodology

Changes were made to this section for clarification purposes, as well as to extend applicability to the ABWR.

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, BWR/6, and ABWR plants are subsequently described.

<u>BWR/3-5 Plants</u>

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:

[

]^{a,c}

(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.

[

]^{a,c}

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

[

]^{a,c}

[

]a,c

(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 *Westinghouse* NRC approved three dimensional core simulator. The full core is modeled to describe detector strings and error rods as accurately as possible.

[

]a,c

[

]a,c

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 cannot be assumed to be the fully inserted group and all intermediate control rod positions for the error rod must be investigated.

Consequently, the same calculation 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.

ABWR Plants

In the ABWR, the automated thermal limit monitor (ATLM) and the multi-channel rod block monitor (RBM) subsystem logic issues a rod block signal used in the reactor coolant isolation system (RCIS) logic to enforce rod blocks. This feature acts to prevent fuel damage by ensuring that the MCPR and maximum linear heat generation rate (MLHGR) do not violate the fuel thermal operating and safety limits. The operating thermal limits rod block function will block rod withdrawal when the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no RWE transient event and thus the event is not analyzed as an AOO.

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

No changes were made to this section.

7.5.5.2 Analysis Methodology

Only administrative changes were made to this section.

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

a,c

Figure 7-1 Fast Transient Analysis Code Interfaces

a,c

[ ]

Figure 7-2 Nodal Neutronic Data for Fast Transient Calculations

8 ACCIDENT ANALYSIS

Only administrative changes were made to this section.

8.1 Summary and Conclusions

Only administrative changes were made to this section.

8.2 Loss of Coolant Accident

Changes were made to this section for clarification purposes.

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 pipe breaks to be considered encompass all sizes and locations up to and including the rapid circumferential failure of the largest piping system connected to the Reactor Pressure Vessel (RPV). 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

Only administrative changes were made to this section.

8.2.1.1 Peak Cladding Temperature

Only administrative changes were made to this section.

8.2.1.2 Local Oxidation

No changes were made to this section.

8.2.1.3 Total Hydrogen Generation

No changes were made to this section.

8.2.1.4 Coolable Geometry

Only administrative changes were made to this section.

8.2.1.5 Long Term Cooling

Changes were made to this section for clarification purposes.

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 water from certain ECCS equipment. Once the RPV has been reflooded, all fuel cladding temperatures would return to near saturation temperatures. 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 provided ECCS performance is not changed. Hence this criterion is not required to be addressed for *Westinghouse* reload applications.

8.2.2 Event Description

Changes were made to this section for clarification purposes. The event description is now applicable to the ABWR.

The LOCA event described here is for a large double-ended guillotine break in the recirculation suction line of a modern BWR with two external recirculation loops that drive the internal jet pumps. Other plant designs have different transient characteristics. For example, the ABWR design has internal recirculation pumps and no large piping systems connected to the RPV below the top of active fuel. This design does not experience significant uncovery of the core.

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 RPV 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 isolation of the steam lines. Following reactor shutdown the pressure begins to fall rapidly. After several seconds the two-phase mixture level in the downcomer falls to the jet pump suction elevation. Uncovery of the jet pump suction lines increases the fluid quality upstream of the break resulting in a sudden decrease in break mass flow rate.

Flashing in the jet pumps and subsequently in the lower plenum occurs when the pressure decreases below the local saturation pressure. This results in a two-phase mixture level rise in the core and downcomer. Following this level swell, the continued inventory decrease results in falling mixture level in the downcomer which initiates the ECCS. Core two-phase mixture level will drop exposing the fuel rods to a steam environment. The downflow of injected coolant from the upper plenum into the core and the upflow of steam from lower plenum flashing provide 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 cladding temperature transient is terminated by emergency core cooling refilling the RPV 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 decay heat generation.

8.2.3 Analysis Methodology

Only administrative changes were made to this section.

8.2.3.1 ECCS Evaluation Model

Only administrative changes were made to this section.

8.2.3.2 Limiting LOCA Design Basis Event

Changes were made to this section for clarification purposes.

<u>Methodology</u>

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.

<u>Discussion</u>

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

]^{a,c}

8.2.3.3 Design Basis Event Analysis

Changes were made to this section for clarification purposes.

<u>Methodology</u>

The plant system response to the postulated design basis 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 MAPLHGR limit is determined that ensures compliance with the LOCA design acceptance criteria.

<u>Discussion</u>

The *Westinghouse* ECCS Evaluation Model contains sufficient conservatism to assure that the LOCA design acceptance criteria are met with a significant safety margin. [

]^{a,c}

8.2.3.4 Total Hydrogen Generation

Changes were made to this section for clarification purposes.

Methodology

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

[

]a,c

Discussion

In the total hydrogen generation analysis, the uncertainty in core-wide bundle power distribution will be bounded [

]a,c As commonly acknowledged, the small number of high-power bundles contributes the largest portion of the total cladding oxidation during a LOCA. [

]a,c

8.2.3.5 MAPLHGR Operating Limit

Changes were made to this section for clarification purposes.

Fuel type specific operating limits are included in the plant technical specifications to ensure that ECCS acceptance criteria are not violated. The fuel type specific operating limit established to meet ECCS LOCA requirements is the MAPLHGR.

Methodology

The plant MAPLHGR operating limit is specified for each fuel type present in the cycle. The plant MAPLHGR operating limit is the most restrictive of:

(1) The MAPLHGR established to comply with the LOCA ECCS acceptance criteria, and

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

Discussion

[

]a,c

8.3 Control Rod Drop Accident

Changes were made to this section for clarification purpose, as well as to extend applicability to ABWRs.

The Control Rod Drop Accident (CRDA) Methodology has been provided in detail in NRC approved topical reports such as Reference 33 and Reference 72. These NRC approved reports describe the *Westinghouse* design basis and the analysis methodology for the CRDA analysis. Note that the need for the CRDA analyses for BWR 2 through 6 designs is because the locking piston in the control rod drive (CRD) mechanism cannot detect separation of the control rod from the drive mechanism during normal rod movements. Newer boiling water reactor designs such as advanced boiling water reactors (ABWR) are equipped with the fine motion control rod drive (FMCRD) which is designed to detect the separation of the control rod from the drive mechanism. Two redundant switches are provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. Actuation of either of these switches cause an immediate rod block and initiates an alarm in the control room. Therefore cycle specific CRDA analyses are not necessary for the ABWR.

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 well within=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

Changes were made to extend applicability to the ABWR

The refueling accident is postulated to provide an upper bound on the release of radioactive materials outside of the drywell. For BWR/2 through BWR/5 and ABWR plants, 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/6 plants, the refueling accident can occur within containment or within the reactor 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

Changes were made to this section for clarification purposes, and to extend applicability to first core applications.

The Fuel Handling Accident analysis in Revision 0 of this document assumes that a competitor fuel assembly, referred to as the "reference assembly", has previously been evaluated for a licensing basis fuel handling accident for limiting conditions in the plant to which a Westinghouse feed assembly, referred to as the "new assembly", is to be installed. This situation is typically encountered when a Westinghouse reload fuel assembly is initially loaded in a core containing Westinghouse fuel with a design different than the new assembly design or competitor fuel and does not account for the case when Westinghouse fuel is loaded when the plant initially starts up. Under this circumstance, the new assembly being installed is evaluated and becomes the reference assembly to which comparisons are made with subsequent Westinghouse feed fuel design assemblies. The remainder of Section 8.4 assumes the existence of a Reference Fuel analysis. The extension in this paragraph assures that there will be a Reference Fuel analysis for all loading combinations of Westinghouse and non-Westinghouse fuel.

Based on the design of *Westinghouse* reload fuel assemblies, the introduction of *Westinghouse* fuel into the core has typically not increase*d* the potential of fission product release to the environment in the past or the dose to control room personnel as a result of a fuel handling accident. This conclusion has been a consequence of the structural characteristics of *Westinghouse* reload fuel. *Westinghouse* reload fuel has been typically found to be lighter than other fuel designs evaluated in the past and more resistant to failure mechanisms associated with fuel handling accidents.

To assess potential fuel handling accidents for *Westinghouse* 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 *Westinghouse* 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 Assembly* relative to the weight of 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 the *New 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 Assembly* is heavier than the *Reference Assembly*, the number of failed fuel rods may increase if the heavier *New Assembly* is dropped on *Reference 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 the 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

No changes were made to the Number of Damaged Fuel Rods sub-section.

[

]a,c

[

]a,c

8.5 Misplaced Assembly Accident

No changes were made to this section.

8.5.1 Mislocated Fuel Assembly

8.5.1.1 Design Basis

Changes were made to this section for clarification purposes.

Basis

This event is considered to be an accident in the *Westinghouse* cycle specific safety analysis process. The SLMCPR is used as the event acceptance limit for this accident.

Discussion

[

]a,c

8.5.1.2 Event Description

No changes were made to this section.

8.5.1.3 Analysis Methodology

No changes were made to this section.

8.5.2 Rotated Fuel Assembly Accident

8.5.2.1 Design Bases

Changes were made to this section for clarification purposes.

Basis

This event is considered to be an accident in the *Westinghouse* cycle specific safety analysis process. The SLMCPR is used as the event acceptance limit for this accident.

Discussion

[

]a,c

8.5.2.2 Event Description

No changes were made to this section.

8.5.2.3 Analysis Methodology

Changes were made to this section for clarification purposes.

[

]a,c

[

]a,c

9 SPECIAL EVENTS ANALYSIS

Only administrative changes were made to this section.

9.1 Summary and Conclusions

Changes were made to this section for clarification purposes.

<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 a *Westinghouse* reload application. Four Special Events are addressed in the *Westinghouse* safety analysis methodology.

The *Westinghouse* 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 cycle specific safety evaluations and plant modification evaluations, as required. *Westinghouse* also has advanced stability tools and 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 *Westinghouse* methodology performs Reactor ASME Overpressure Protection analysis to confirm for each 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 application. The *Westinghouse* 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 core design. As discussed in Section 6.3.1.3, it is not necessary to evaluate ATWS events for the use of *Westinghouse* fuel. However, the potential does exist for performing ATWS evaluations for certain types of plant modifications. The *Westinghouse* 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

Changes were made to this section for clarification purposes. Outdated information has been removed.

Westinghouse has analysis codes and methodologies to perform core thermal-hydraulic stability evaluations for plant specific reload applications and plant modifications as required. *Westinghouse* uses time domain codes for stability analysis (see Table 9-1). These stability analysis tools can be used for 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 *Westinghouse* stability analysis methodology, and the plant application methodology.

9.2.1 Design Bases

Only administrative changes were made to this section.

9.2.2 Stability Analysis Methodology

Changes were made to this section for clarification purposes. Outdated information has been removed.

Methodology

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

Discussion

The *Westinghouse* 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 safety analysis process.

The *Westinghouse* 3D time domain codes are described in References 44 *and 72*. References 44 *and 72* provide a description of the codes and qualification for core and channel stability performance evaluations. Three dimensional transient stability analysis methods are used in the *Westinghouse* stability methodology. Licensing Topical Report CENPD-295-P-A (Reference 45) together with Reference 74, submitted for review, provide a description of general stability analysis methodology using the stability codes.

9.2.3 Plant Reload Application Methodology

Only administrative changes were made to this section.

9.3 Overpressurization Protection

No changes were made to this section.

9.3.1 Design Bases

Changes were made to this section for clarification purposes.

Basis

The plant overpressure protection system capability shall be confirmed adequate for the cycle specific licensing analysis. 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 is the ASME Code upset limit of 110 % of the reactor pressure vessel design pressure as stated in Section 6.4.3.

9.3.2 Overpressurization Protection Methodology

Changes were made to this section for clarification purposes.

Methodology

The most severe pressurization event is analyzed for each cycle specific licensing analysis 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:

[

]a,c

[

]a,c

The overpressurization MSIV closure event is analyzed with the NRC approved dynamic analysis methods.

[

]a,c

Discussion

The overpressurization MSIV closure event could be treated as an emergency condition, with acceptable results compared to the ASME emergency condition limits (i.e., the reactor pressure acceptance limit of 120% of design pressure). However, the Westinghouse 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

Changes were made to this section for clarification purposes.

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 xenon-free core and no movement of the control rods. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical poison system, satisfying GDC-26 and 27 of 10 CFR 50, Appendix A (Reference 42).

9.4.2 SLCS Evaluation Methodology

Changes were made to this section for clarification purposes.

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 NRC approved lattice physics code and 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.

[

]a,c

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. [

]a,c

[

]a,c

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 *core* simulator Reference Core model are performed with the appropriate boron concentration. The lattice physics methods are used to explicitly model the fuel assembly contained in the reference core. The moderator cross section are developed assuming a uniform distribution of the boron concentration. These cross sections are utilized in the three-dimensional *core* simulator to evaluate the impact of the borated moderator on core reactivity.

9.5 Anticipated Transients Without Scram (ATWS)

Outdated information has been removed from this section.

9.5.1 Design Bases

Changes were made to this section for clarification purposes.

Bases

No changes were made to the Bases sub-section.

Discussion

[

]a,c

9.5.2 ATWS Evaluation Methodology

Changes were made to this section for clarification purposes.

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 *a Westinghouse* fuel design is described below.

Methodology

Each new *Westinghouse* 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. [

]^{a,c} Once the fuel design in confirmed not to have a significant impact in the current ATWS analysis, it is considered acceptable. Methodology for ATWS analysis is contained in Reference 73.

Discussion

[

]^{a,c}

TABLE 9-1

WESTINGHOUSE STABILITY ANALYSIS TOOLS

Tool	Methods	Methods Qualification	Analysis Methodology
3D Time Domain Codes	CENPD-294-P-A (RAMONA code) WCAP-16747-P-A (POLCA-T code)	CENPD-294-P-A WCAP-16747-P-A	CENPD-295-P-A WCAP-16747-P-A WCAP 17137-P

CENPD-294-P-A (Reference 44)
CENPD-295-P-A (Reference 45)
WCAP-16747-P-A (Reference 72)
WCAP-17137-P (Reference 74)
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) (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) Described in WCAP 17137 (Reference 74)

Westinghouse Stability Evaluation	Described in CENPD-295-P-A (Reference 45) Described in WCAP 17137 (Reference 74)

10 REFERENCES

1. Not used in this supplement.
2. Not used in this supplement.
3. Not used in this supplement.
4. Not used in this supplement.
5. Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity, Westinghouse Report WCAP-11427-P-A (proprietary), WCAP-12392-NP-A (nonproprietary), November 1989.
6. Not used in this supplement.
7. Reference Safety Report for Boiling Water Reactor Reload Fuel, CENPD-300-P-A (proprietary), CENPD-300-NP-A (non-proprietary), July 1996.
8. Not used in this supplement.
9. Not used in this supplement.
10. Not used in this supplement.
11. Not used in this supplement.
12. Not used in this supplement.
13. Not used in this supplement.
14. Not used in this supplement.
15. Not used in this supplement.
16. Not used in this supplement.
17. Not used in this supplement.
18. Not used in this supplement.
19. Not used in this supplement.
20. CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors, ABB Report BR-91-255-P-A, rev. 1 (proprietary), BR-91-266-NP-A (non-proprietary), May 1991.

21. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification, ABB Report RPB-90-93-P-A (proprietary), RPB-90-91-NP-A (nonproprietary), October 1991.

22. Not used in this supplement.

23. BISON - A One Dimensional Dynamic Analysis Code for Boiling Water Reactors, ABB Report RPA 90-90-P-A (proprietary), RPA 90-90-NP-A (nonproprietary), December 1991.

24. Not used in this supplement.

25. Not used in this supplement.

26. Not used in this supplement.

27. Not used in this supplement.

28. Not used in this supplement.

29. Not used in this supplement.

30. Not used in this supplement.

31. Letter A. C. Thadani (NRC) to N. O. Jonsson (ABB) "Acceptance for Referencing of ABB Report UR 89-210: 'SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-Bundle,' (TAC No. M75938)," January 13, 1993.

32. Letter A. C. Thadani (NRC) to D. B. Ebeling-Koning (ABB) "Acceptance for Referencing of ABB CE Supplemental Information Regarding the ABB SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-bundle, (TAC No. 86121) July 12, 1993.

33. 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.

34. Intentionally Deleted.

35. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel, ABB Report CENPD-283-P-A (proprietary), CENPD-283-NP-A (non-proprietary), July 1996.

36. Fuel Rod Design Methods for Boiling Water Reactors, ABB Report CENPD-285-P-A (proprietary), CENPD-285-NP-A (non-proprietary), July 1996.

37. Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, ABB Report CENPD-287-P-A (proprietary), CENPD-287-NP-A (non-proprietary), July 1996.

38. ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel, ABB Report CENPD-288-P-A (proprietary), CENPD-288-NP-A (non-proprietary), July 1996.

39. BISON - One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification ABB Report CENPD-292-P-A (proprietary), CENPD-292-NP-A (non-proprietary), July 1996.

40. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 1 to Code Description and Qualification, ABB Report CENPD-293-P-A (proprietary), CENPD-293-NP-A (non-proprietary), July 1996.

41. ABB Correlation for Predicting CPR Margin in Boiling Water Reactors ABB Report CENPD-286-P-A (proprietary) CENPD-286-NP-A (nonproprietary) (ABB designation of WCAP-10972 under NRC review).

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

43. U.S. Nuclear Regulatory Commission Standard Review Plan, NUREG-0800.

44. Thermal-Hydraulic Stability Analysis Methods for Boiling Water Reactors CENPD-294-P-A (proprietary), CENPD-294-NP-A (non-proprietary), July 1996.

45. Thermal-Hydraulic Stability Analysis Methodology for Boiling Water Reactors CENPD-295-P-A (proprietary), CENPD-295-NP-A (non-proprietary), July 1996.

46. General Electric Report NEDE-10958, General Electric BWR Thermal Analysis Basis (GETAB), November, 1973.

47. U.S. Nuclear Regulatory Commission Regulatory Guides.

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

49. ASME Boiler and Pressure Vessel Code, Section III, American Society of Mechanical Engineers.

50. Not used in this supplement.

51. Not used in this supplement.

52. United States Atomic Energy Commission, "Opinion of the Commission -- Acceptable Criteria for Emergency Core Cooling Systems -- Light-Water Cooled Nuclear Reactors," Docket No. RM-50-1, December 28, 1974.

53. Compendium of ECSS Research for Realistic LOCA Analysis, NUREG-1230, U.S. Nuclear Regulatory Commission, September 1988.

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

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

56. Not used in this supplement.

57. Not used in this supplement.

58. Not used in this supplement.

59. Not used in this supplement.

60. Not used in this supplement.

61. Statistical Combination of Uncertainties, Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 & 2, Combustion Engineering Topical Report CEN-124(B)-P, January, 1980.

62. Not used in this supplement.

63. Not used in this supplement.

64. Not used in this supplement.

65. The advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors, Westinghouse Report CENPD-390-P-A (proprietary), CENPD-390-NP-A (nonproprietary), December 2000.

66. 10x10 SVEA Critical Power Experiments and CPR Correlation: SVEA-96 Optima2, Westinghouse Report WCAP-16081-P-A (proprietary), WCAP-16081-NP-A (nonproprietary), March 2005.

67. Westinghouse BWR ECCS Evaluation model: Supplement 2 to Code Description, Qualification and Application, Westinghouse Report WCAP-15682-P-A (proprietary), WCAP-15682-NP-A (nonproprietary), April 2003.

68. 'Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel, Westinghouse Report WCAP-16078-P-A (proprietary), WCAP-16078-NP-A (nonproprietary), July 2004.

69. Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1, Westinghouse Report WCAP-15836-P-A (proprietary), WCAP-15836-NP-A (nonproprietary), April 2006.

70. BWR Fuel Assembly Mechanical Design Methodology - Supplement 1, Westinghouse Report WCAP-15942-P-A (proprietary), WCAP-15942-NP-A (nonproprietary), March 2006.

71. Supplement 2 to BISON Topical Report RPA 90-90-P-A, Westinghouse Report WCAP-16606-P-A (proprietary), WCAP-16606-NP-A (nonproprietary), January 2008.

72. POLCA-T: System Analysis Code with Three-Dimensional Core Mode, Westinghouse Report WCAP-16747-P-A (proprietary), WCAP-16747-NP-A (nonproprietary), September 2010.

73. Fast Transient and ATWS methodology, Westinghouse Report WCAP-17203-P (proprietary). Submitted to the NRC June 2010.

74 Westinghouse Stability Methodology for the ABWR, Westinghouse Report WCAP-17137-P (proprietary). Submitted to the NRC September 2010.

75. Westinghouse Containment Analysis Methodology, Westinghouse Report WCAP-16608-P-A (proprietary), WCAP-16608-NP-A (nonproprietary), March 2009.

76 NAI 8907-06, GOTHIC Containment Analysis Package Technical Manual.

APPENDIX A: DESCRIPTION OF CODES

A.1 Mechanical Design

A.1.1 Fuel Rod Performance Codes

A.1.1.1 VIK-3

The VIK-3 code was introduced after the submittal of CENPD-300 Revision 0. This text has been added to describe the VIK-3 code.

The computer code VIK-3 calculates stresses in light water reactor (LWR) fuel rod cladding as a function of fuel burnup or irradiation time. Both fully re-crystallized and cold work stress-relieved Zircaloy cladding can be evaluated. VIK-3 has an option allowing its execution in conjunction with STAV in order to provide cladding stress evaluations as a function of fuel rod burnup based on materials properties and STAV calculated parameters.

The code consists of a number of subroutines, each one calculating the stress due to the different sources or load cases. Stress levels are calculated at the clad inner and outer radii at three axial locations, namely at a spacer, between spacers and at the bottom end plug. Depending on the origin of the stress and on geometrical and material discontinuities in the design, each stress is classified with the appropriate stress category. The effective stresses are calculated using the Tresca relationship in accordance with Section III of the ASME Code.

A complete description of the VIK-3 code is provided in *Reference 69.*

A.1.1.2 STAV7.2

The STAV7.2 code was introduced after the submittal of CENPD-300 Revision 0. This text has been added to describe the STAV7.2 code.

The STAV7.2 code is the latest version of the STAV fuel rod performance code series developed and used at *Westinghouse*. This tool enables the evaluation of the steady-state performance of fuel rods under the conditions prevailing in a light water reactor (LWR). STAV7.2 can model both UO_2 and $(U,Gd)O_2$ fuel.

STAV7.2 is the primary analysis code used in fuel thermal mechanical design process.

STAV7.2 calculates the variation over time of all significant fuel rod performance parameters including fuel and cladding temperatures, fuel densification, fuel swelling, fission product gas release, rod internal pressure, and pellet-cladding gap conductance. Stresses and strains in the cladding due to elastic, thermal, creep and plastic deformations are calculated. Cladding oxidations is modeled and its influence on other parameters considered. Other submodels include burnup-dependent radial power distributions for both UO_2 and $(U,Gd)O_2$ fuel, fuel grain growth, and helium release.

For example, in the reload safety analysis process, STAV7.2 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 STAV7.2 code is provided in *Reference 69.*

A.1.1.3COLLAPS-II

Changes were made to this section for clarification purposes.

COLLAPS-II is used by *Westinghouse* for prediction of cladding ovality in BWRs fuel rods 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 pressure. The cross section of the tube is assumed to have a slight initial deviation from circularity. 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-II code is provided in *Reference 69.*

A.1.2 Finite Element Model Analysis Codes

A.1.2.1ANSYS

No changes were made to this section, but this section is included as a reference.

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

No changes were made to this section, but this section is included as a reference.

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 65.*

A.2.1 Two Dimensional Lattice Design

A.2.1.1PHOENIX

Changes were made to this section for clarification purposes.

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 *Westinghouse* depletion program for BWR 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-groups homogenized nuclear data 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 Core Simulator

A.2.2.1POLCA

Changes were made to this section for clarification purposes

POLCA is a core simulator which provides realistic three-dimensional simulations of the nuclear, thermal, and hydraulic conditions in a boiling water reactor (BWR) core. The POLCA code is described in Reference 65.

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 void content affects 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.

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.

The core physics model of POLCA is also included in the system analysis code POLCA-T; see Section A.4.2.2.

A.3 Thermal-Hydraulics Design

A.3.1 POLCA

Updates to this section have been made to describe the current Westinghouse codes.

Westinghouse has utilized the CONDOR code for the evaluation of the steady-state thermal-hydraulic performance of BWR primary systems. The same models were also used as the thermal-hydraulic module of the three-dimensional core simulator code, POLCA. The complete CONDOR code functionality is now included in the nodal code POLCA as described in Reference 65.

POLCA 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 CONDOR code is provided in Reference 20 and of the POLCA code in Reference 65.

A.4 Safety Analysis

A.4.1 One Dimensional Time Domain Dynamic Analysis

A.4.1.1BISON

No changes were made to this section, but this section is included as a reference.

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.4.2.1RAMONA-3

Information added to this section comes from RAI response F5.

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.4.2.2POLCA-T

This section was added to reflect the creation of a new 3D core physics modeling code.

POLCA-T is an advanced dynamic system analysis code with the three dimensional (3D) core physics modeling capabilities described by the nodal code POLCA presented above.

POLCA-T is a computer code for transient thermal hydraulic and neutron kinetics analysis of BWR plants. It can be used as a general tool for advanced simulation of single and two phase flow systems including non condensable gases. The code has a full-3D coupled core neutronics/thermo-hydraulics model where each fuel assembly in the reactor core can be explicitly represented in the thermal hydraulic model. The reactor pressure vessel, external pump loops, steam system, feedwater system, emergency core cooling systems, control systems, and steam relief system can be modeled in detail.

POLCA-T is, as RAMONA, specifically designed to simulate normal and abnormal operational plant transients, as well as accidents and special events like ATWS and stability requiring complete three-dimensional representation. Because of its unique feature of combining full 3D modeling of the reactor core and transient plant response, it is particularly suited for transients involving significant within-the-core effects.

A detailed description of the modeling capabilities of POLCA-T is provided in Reference 72. These capabilities make POLCA-T suitable to replace both RAMONA and BISON in their specific applications. The use of POLCA-T for those applications is being introduced in a staged process. The first two applications Control Rod Drop Accident (CRDA) Analysis and Stability Analysis have been reviewed and approved by the NRC. Subsequent applications (including Transient Analysis and ATWS) will be submitted prior to their use. Each application is included as an appendix to the code description which contains the evaluation model and the qualification of the code for performing the intended analysis.

A.4.3 ECCS Evaluation

A.4.3.1 GOBLIN Series

Changes were made to this section for clarification purposes.

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-3D. 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-3D calculates the detailed temperature distribution at a given axial cross section of the assembly analyzed by DRAGON. Its boundary conditions are supplied by GOBLIN-EM and DRAGON.

A detailed description of these codes is provided in References 21, 40, *67 and 68.*

A.4.4 Intentionally Deleted

A.4.4.1 Intentionally Deleted

A.4.4.2 Intentionally Deleted

A.5 Statistical Analysis

A.5.1 Industry Accepted Codes

A.5.1.1 SIGMA

No changes were made to this section, but this section is included as a reference.

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 involve 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

No changes were made to this section, but this section is included as a reference.

There are some codes used by *Westinghouse* to perform statistical analysis that are approved by NRC for use by the utility. The utility can provide these codes to *Westinghouse* 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.

A.6 Containment Analysis

A.6.1 GOTHIC

This section was added to include containment analysis codes which are described in this topical.

Westinghouse uses the GOTHIC computer code to perform design-basis containment analyses. The code has been developed by Numerical Applications Incorporated (NAI) with funding by the Electric Power Research Institute (EPRI).

GOTHIC solves the integral form of the conservation equations for mass, momentum, and energy for multi-components, two-phase flow. The conservation equations are solved for three fields; continuous liquid, liquid drops, and steam/gas phase. The three fields may be in thermal non-equilibrium within the same computational cell. This treatment allows the modeling of sub-cooled drops (e.g. containment spray) falling through an atmosphere of saturated steam. The gas component of the steam/gas field may be comprised of up to eight different non-condensable gases with mass balances performed for each component. Relative velocities are calculated for each field as well as the effects of two-phase slip on the pressure drop. Heat and mass transfer between the phases, surfaces, and the fluid is also allowed.

The GOTHIC code is capable of performing calculations in three modes. The code can be used in the lumped parameter nodal network mode, the two-dimensional finite difference mode, and the three-dimensional finite difference mode. The code also contains the options to model a large number of structures and components such as heated and unheated conductors, pumps, fans, valves, heat exchangers, ice condensors, etc. These components can be coupled to simulate typical containment systems.

A detailed description of the GOTHIC code is provided in Reference 76. *Westinghouse* methodology for Mark I containment analyses and for the ABWR containment analysis is provided in Reference 75.

Attachment 1

Page Intentionally Left Blank

TABLE OF CONTENTS

LIST OF ACRONYMS

1D	One-Dimensional
2D	Two-Dimensional
3D	Three-Dimensional
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrences
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARO	All Rods Out
ASME	American Society of Mechanical Engineers
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
BOLFFP	Beginning of Low Flow and Full Power
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRDA	Control Rod Drop Accident
DAR	Design Analysis Record
ECCS	Emergency Core Cooling System
EFPH	Effective Full Power Hours
ELLLA	Extended Load Line Limit Analysis
EOC	End of Cycle

LIST OF ACRONYMS (Continued)

EOFP	End of Full Power
EOLFFP	End of Low Flow and Full Power
EPRI	Electric Power Research Institute
FCPR	Final CPR
FFWTR	Final Feedwater Temperature Reduction
FGR	Fission Gas Release fraction
FMCPR	Final Minimum CPR
FMCRD	Fine Motion Control Rod Drive
FRAD	Radial Power Form Factor
FRSC	Failure of RHR Shutdown Cooling
FSAR	Final Safety Analysis Report
FWCF	Feedwater Controller Failure
gap HTC	Heat Transfer Coefficient between pellet and cladding
GDC	General Design Criteria
GLRNB	Generator Load Rejection No Bypass
GLRWOB	Generator Load Rejection Without Bypass
GLRWB	Generator Load Rejection With Bypass
HPCI	High Pressure Coolant Injection
ICF	Increased Core Flow
ICPR	Initial CPR
IMCPR	Initial Minimum CPR
LHGR	Linear Heat Generation Rate
LLLA	Load Line Limit Analysis

LIST OF ACRONYMS (Continued)

LOAP	Loss of Auxiliary Power
LOCA	Loss of Coolant Accident
LOCV	Loss of Condenser Vacuum
LOFH	Loss of Feedwater Heating
LOOP	Loss of Offsite Power
LPRM	Local Power Range Monitor
LWR	Light Water Reactor
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
NAI	Numerical Applications Incorporated
NBR	Nuclear Boiler Rated
NRC	Nuclear Regulatory Commission
OL	Operating Limit
OLMCPR	Operating Limit MCPR
PCI	Pellet Clad Interaction
RBM	Rod Block Monitor
RAI	Request for Additional Information
RCIS	Reactor Coolant Isolation System

LIST OF ACRONYMS (Continued)

RCPR	ΔCPR/ICPR
RHR	Residual Heat Removal
RPR	Recirculation Pump Runup
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RSR	Reference Safety Report
RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
SAFDL	Specified Acceptable Fuel Design Limit
SDM	Shutdown Margin
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit MCPR
SLO	Single Loop Operation
SRP	Standard Review Plan
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TIP	Traversing In-core Probe
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

This Reference Safety Report (RSR) for boiling water reactor (BWR) reload fuel describes the reload and initial core fuel design and safety analysis process used in specific plant applications. Specific topics related to the *Westinghouse* 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.

The objective of this report is to obtain generic Nuclear Regulatory Commission (NRC) approval for the Westinghouse reload fuel design and safety analysis process that utilizes the Westinghouse 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 Westinghouse 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 Westinghouse 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 licensing 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 Westinghouse uses the current plant licensing basis as an inherent part of the process for updating the plant safety analysis. By using the current plant licensing 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 Westinghouse methodology, because these differences are contained in the current plant licensing basis.

A standard reload of Westinghouse reload fuel is a typical plant change that is not expected to result in 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.

1.1 Background

The United States licensing of the *Westinghouse* 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*s* 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*s* (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. became the cognizant origination for BWR reload fuel application in the United States.

Subsequent to *this* consolidation, the NRC has issued approval for additional ABB Licensing Topical Reports (References 31 through 41 *and 65*).

In April 2000 ABB nuclear businesses were acquired by the parent company of Westinghouse Electric Company (the successor company of the Westinghouse Electric Corporation nuclear business). After this second consolidation, new Licensing Topical Reports were approved by the NRC (References 66 through 76). Quality control, maintenance, and implementation for the complete Westinghouse U.S. BWR reload fuel licensing methodologies resides with the same cognizant organization and persons, now a part of Westinghouse Electric Company.

This document, the "Reference Safety Report for BWR Fuel and Core Analyses" integrates the *Westinghouse* BWR reload methodology intended to be used for *Westinghouse* U.S. reload and plant operational modification applications.

1.2 BWR Reload Licensing Documents

The *Westinghouse* BWR reload fuel safety analysis methodology is contained in a number of other Licensing Topical Reports referenced throughout this report. Each report describes for one or more disciplines the code methods, code qualification, design bases, analysis methodology and/or sample applications and has been previously reviewed and accepted or is in the process of being reviewed by the NRC. 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. All conditions in the referenced licensing reports and the reload methodology will be met during future reload analyses. Compliance with the NRC conditions for each discipline is noted in a relevant Design Analysis Record (DAR).

1.3 Report Overview

This document describes the *Westinghouse* reload *licensing* 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 on the reload fuel and core design process is placed on the inputs and interfaces of the design with the reload *licensing* analysis. The reload *licensing* 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 an overview of the transient analysis methodology for anticipated operational occurrences (transient analyses). 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 *Westinghouse* reload analysis methodology. No changes have been made to Appendices B, C, D, E and F. Therefore they are not attached to this supplemental update.

TABLE 1-1

WESTINGHOUSE BWR RELOAD FUEL LICENSING TOPICAL REPORTS

Report Number	Report Title	Discipline
CENPD-285-P-A	Fuel Rod Design Methods for Boiling Water Reactors	Mechanical
WCAP-15836-P-A	Fuel Rod Design Methods for Boiling Water Reactors: Supplement 1	Mechanical
CENPD-287-P-A	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors	Mechanical
WCAP-15942-P-A	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors: Supplement 1	Mechanical
CENPD-288-P-A	ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel	Mechanical
CENPD-390-P-A	The Advanced PHOENIX and POLCA code for Nuclear Design of Boiling Water Reactor	Nuclear
UR 89-210-P-A	SVEA-96 Critical Power Experiments on a Full Scale 24-rod Sub-Bundle	Thermal-Hydraulic
CENPD-389-P-A	10x10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96+	Thermal-Hydraulic
CENPD-392-P-A	10x10 SVEA Fuel Critical Power Experiments and CPR Correlations:SVEA-96	Thermal-Hydraulic
WCAP-16047-P-A	Improved Application of Westinghouse Boiling-Length CPR Correlation for BWR SVEA Fuel	Thermal-Hydraulic
WCAP-16081-P-A	10x10 SVEA Fuel Critical Power Experiments and CPR Correlations:SVEA-96 Optima2	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
WCAP-16606-P-A	Supplement 2 to BISON Topical Report RPA 90-90-P-A	Special Events: ATWS
WCAP-17079-P	Supplement 3 to BISON Topical Report RPA 90-90-P-A SAFIR Control System Simulator	AOO: Fast Transient
WCAP-17202-P	Supplement 4 to BISON Topical Report RPA 90-90-P-A	AOO: Fast Transient
WCAP-17203-P	Fast Transient and ATWS Methodology	AOO: Fast Transient Special Events: ATWS
WCAP-16747-P Appendix C	POLCA-T: Application for Transient Analysis	AOO: Fast Transient
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
WCAP-15682-P-A	Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 2 to Code Description, Qualification and Application	Accidents: LOCA
WCAP-16078-P-A	Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2	Accidents: LOCA
WCAP-16747-P-A, Appendix A	POLCA-T: Control Rod Drop Accident Analysis (CRDA)	Accidents: CRDA
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
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
WCAP-16747-P-A, Appendix B	POLCA-T: Application for Stability Analysis	Special Events: Stability
WCAP 17137-P	Westinghouse Stability Methodology for the ABWR	Special Events: Stability
WCAP-16747-P Appendix D	POLCA-T: Application for Anticipated Transient without Scam Analysis	Special Events: ATWS
WCAP-16608-P-A	Westinghouse Containment Analysis Methodology	Containment Analysis
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 WCAP-15942-P-A (Normal Operation/AOO) CENPD-288-P-A (Accidents)	CENPD-285-P-A WCAP-15835-P-A (Fuel Rod) CENPD-287-P-A WCAP-15942-P-A (Fuel Assembly)	CENPD-285-P-A WCAP-15835-P-A	CENPD-287-P-A WCAP-15942-P-A
Nuclear	CENPD-300-P-A	BR 91-402-P-A (POLCA4) CENPD-390-P-A (POLCA7)	BR 91-402-P-A CENPD-390-P-A	CENPD-300-P-A
Thermal-Hydraulic	CENPD-300-P-A	BR 91-255-P-A, Rev. 1 CENPD-390-P-A UR 89-210-P-A CENPD-389-P-A CENPD-392-P-A WCAP-16047-P-A WCAP-16081-P-A (CPR Correlations)	BR 91-255-P-A, Rev. 1 CENPD-390-P-A UR 89-210-P-A CENPD-389-P-A CENPD-392-P-A WCAP-16047-P-A WCAP-16081-P-A (CPR Correlations)	CENPD-300-P-A
AOO: Fast Transients	CENPD-300-P-A WCAP-17203-P	RPA 90-90-P-A CENPD-292-P-A (BISON) WCAP-16747-P-A (POLCA-T)	RPA 90-90-P-A CENPD-292-P-A WCAP-16606-P-A WCAP-17202-P WCAP-16747-P Appendix C	CENPD-300-P-A WCAP-16747-P Appendix C
AOO: Slow Transients	CENPD-300-P-A	BR 91-402-P-A CENPD-390-P-A	BR 91-402-P-A CENPD-390-P-A	CENPD-300-P-A
Accidents: LOCA	RPB 90-94-P-A CENPD-283-P-A CENPD-300-P-A	RPB 90-93-P-A CENPD-293-P-A WCAP-15682-P-A WCAP-16078-P-A	RPB 90-93-P-A	CENPD-300-P-A
Accidents: CRDA	CENPD-284-P-A	BR 91-402-P-A (RAMONA) WCAP-16747-P-A (POLCA-T)	CENPD-284-P-A WCAP-16747-P-A, Appendix A (POLCA-T)	CENPD-284-P-A WCAP-16747-P-A, Appendix A (POLCA-T)
Accidents: Others	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A

Discipline	Design Bases and Methodology	Code Methods	Qualification	Application
Special Events: Stability	CENPD-300-P-A CENPD-295-P-A WCAP-16747-P-A, Appendix B (POLCA-T) WCAP-17137-P (POLCA-T)	CENPD-294-P-A (RAMONA-3) WCAP-16747-P-A (POLCA-T)	CENPD-294-P-A (RAMONA-3) WCAP-16747-P-A, Appendix B (POLCA-T)	CENPD-300-P-A CENPD-295-P-A WCAP-16747-P-A, Appendix B (POLCA-T) WCAP-17137-P (POLCA-T)
Special Events: Overpressure Protection	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A	CENPD-300-P-A
Special Events: ATWS	CENPD-300-P-A WCAP-16606-P-A WCAP-17203-P	RPA 90-90-P-A CENPD-292-P-A (BISON) WCAP-16747-P-A (POLCA-T) WCAP-16608-P-A (GOTHIC)	WCAP-16606-P-A (BISON) WCAP-16747-P Appendix D (POLCA-T) WCAP-16608-P-A (GOTHIC)	CENPD-300-P-A WCAP-16606-P-A (BISON) WCAP-16747-P Appendix D (POLCA-T) WCAP-16608-P-A (GOTHIC)

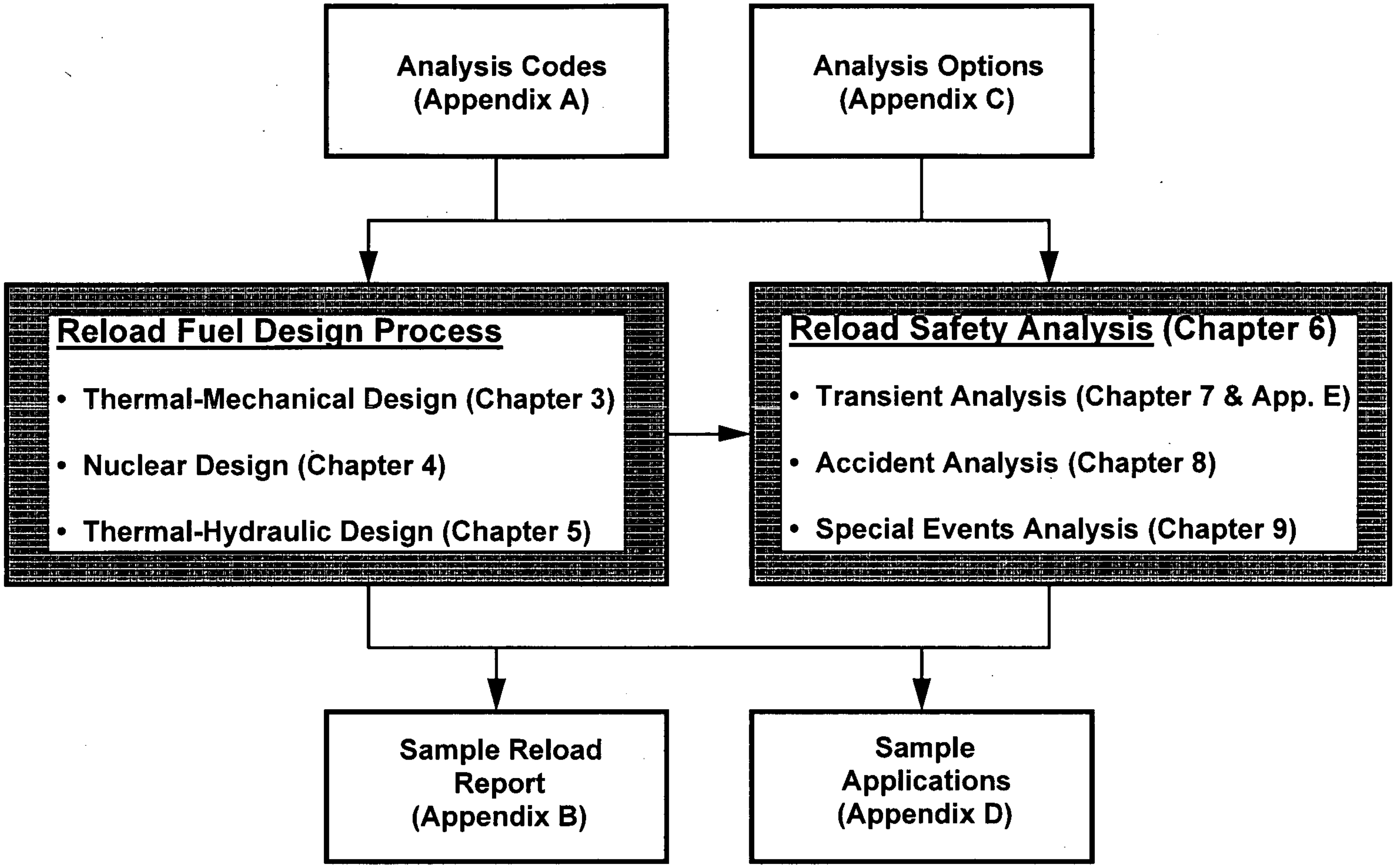


Figure 1-1. Reference Safety Report Structure

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 *Westinghouse* 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, *Westinghouse* 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 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 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 *Westinghouse* BWR fuel design and safety analysis methodology has 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 *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* reload fuel design and safety analysis methodology can be used to demonstrate the acceptability of a plant operating with *Westinghouse* reload fuel in a new core configuration consistent with operation in the allowable operating domain.

(4) The *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* 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 70* 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 *Westinghouse* mechanical design methodology addresses the fuel assembly and fuel rod mechanical evaluation identified in Section 4.2 of the Standard Review Plan, NUREG-0800 (Reference 43). An overview of mechanical design criteria and methodology for the fuel assembly and fuel rod performance analyses is provided in this section.

Detailed methodology is provided in a separate mechanical design methodology topical, Reference 70. Specifically, *Reference 70* contains mechanical design criteria which assure that the requirements of NUREG-0800 (Reference 43) are satisfied, the methodology for performing mechanical design evaluations relative to those criteria, and an application of that methodology to the *Westinghouse* BWR fuel assembly which demonstrates that the fuel assembly satisfies the design criteria.

This chapter also provides the interface between the mechanical design of *Westinghouse* fuel and the other design activities. Specifically, the type of mechanical design and fuel performance 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.

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.

3.2 Design Criteria

The principal objective of the 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 70*:

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 70*.

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 *Westinghouse* 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 70*. In addition, an evaluation of the fuel assembly relative to the design criteria provided in Section 3 of *Reference 70* 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 methodology described in *Reference 70* is applied to a *Westinghouse BWR fuel* design as an illustration. 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 *Westinghouse BWR fuel* 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

This section provides the interface between the mechanical design of Westinghouse fuel and the other design activities.

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 *Westinghouse* and other vendors.

All mechanical design data for *Westinghouse* 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 *Westinghouse*-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:

[

]a,c

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

3.4.3 Mechanical Design Input to the Transient Analyses

All mechanical design data for *Westinghouse* reload assemblies required for the transient analyses are formally provided internally. *Westinghouse* obtains from the utility the required data for non-*Westinghouse* fuel which resides in the reactor with *Westinghouse* fuel and which supports the transient analyses from the utility.

In general, the same dimensional data required for the *Westinghouse* assembly design are required for the non-*Westinghouse* fuel assemblies.

Assembly Input Data

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.

[

]a,c

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. [

]a,c

Detailed methodology for providing Gap HTC to LOCA analyses is discussed in Sections 4.4.4 of Reference 70.

3.4.6 Mechanical Design Input to CRDA Analyses

The methodology for analyzing the Control Rod Drop Accident is described in References 33 and 72.

The description in References 33 and 72 include the treatment of mechanical input data, such as gap HTCs, and, therefore, are not repeated in this document. Methodology for providing Gap HTC to CRDA analyses is discussed in Section 4.4.3 of Reference 70.

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. [

][a,c] Detailed methodology for providing Gap HTC to stability analyses is discussed in Section 4.4.5 of Reference 70.

4 NUCLEAR DESIGN

4.1 Summary and Conclusions

This section provides *Westinghouse* 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.

Specifically, this section contains the following:

- The *Westinghouse* nuclear design bases;
- The *Westinghouse* 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-hydraulic, AOO, accident, and special event analyses.

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.

The information in this section supports the following conclusions regarding the *Westinghouse* 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 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 *Westinghouse* as well as non- *Westinghouse* fuel is acceptable for design and licensing applications.

4.2 Nuclear Design Bases

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

4.2.1 Cycle Energy and Fuel Burnup

Basis

The nuclear design basis is to install sufficient reactivity in the 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.

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 *Westinghouse* 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 *Westinghouse* methodology for evaluating conformance to this design basis is discussed in Section 4.3.

4.2.3 Control of Power Distribution

<u>Basis</u>

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.

<u>Discussion</u>

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 *Westinghouse* 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 US 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 *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* treatment of hydrodynamic stability is discussed in Section 9. It should be noted that the *Westinghouse* time domain methods referred to in Section 9 treat variations in xenon concentration as well as the effects of void feedback.

4.3 Nuclear Design Methodology

4.3.1 Reference Core

The *Westinghouse* 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.

[

]a,c

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. *Westinghouse* 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. [

]a,c

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>

[

]^{a,c}

[

]a,c

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 increases 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 *core* 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 Cycle Energy and 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 *Westinghouse* 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 Section 3.

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.

[

]^{a,c}

[

]a,c

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. *Westinghouse* 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.

[

]a,c

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:

[

]^{a,c}

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:

[

]^{a,c}

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 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, [ ]a,c 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.

[

]a,c

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 feed fuel bundle and Reference Core loading 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 70*, a Thermal-Mechanical Operating Limit (TMOL) is established for which all

mechanical design bases are satisfied [

]^{a,c} Confirmation that the TMOL is not exceeded demonstrates that all mechanical fuel rod design bases are satisfied. [

]^{a,c}

(3) The feed fuel bundle and Reference Core loading 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. [

]^{a,c}

(4) The reload feed fuel 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 MAPLHGR limits at each statepoint to confirm that the design provides sufficient margin to assure that the MAPLHGR limits will not be approached during normal operation in the plant application.

4.3.2.4 Shutdown Margin

The *Westinghouse* 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.

[

]a,c

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 70*. These calculations are performed to confirm that the TMOL is in fact bounding for a specific application. [

]a,c

Discussion

An example of the selection of limiting fuel rods and the resulting power histories is provided in Section 4.3.0 of *Reference 70.*

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 *as* discussed in Section 4.3.1. The term "conservative" refers in this case to selecting 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. [

]^{a,c}

4.4.3 Nuclear Design Input to Transient Analyses

The AOOs discussed in Section 7 can be categorized as "fast" or "slow". The slow 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 core simulator model discussed in Section 4.3.1.

The current NRC approved codes to analyze fast transients are shown in Appendix A.

4.4.3.1 1D Kinetics and Average Channel Analysis Model

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. [

]^{a,c}

[

]a,c

4.4.3.2 3D Kinetics and Parallel Channel Model

If the fast transients are analyzed using a Westinghouse NRC approved 3D kinetics dynamic analysis code with parallel channels, all nuclear and thermal-hydraulic data simulating the three-dimensional situation can be taken directly from the static core simulator and no collapsing is needed. Current 3D kinetics and static core simulator codes used in this analysis are presented in Appendix A.

4.4.4 Nuclear Design Input to the Accident Analyses

4.4.4.1 Nuclear Design Input to LOCA Analyses

The LOCA analysis methodology is described in Section 8.2. 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. [

]a,c

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

[

]^{a,c}

4.4.4.2 Nuclear Design Input to CRDA Analyses

Section 8.3 describes Westinghouse CRDA methodology. The 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. [

]^{a,c}

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. [

]^{a,c}

4.4.4.3 Nuclear Design Input to Fuel Handling Accident Analyses

[

]a,c

4.4.4.4 Mislocated and Rotated Fuel Assembly Analyses

Mislocated and Rotated Fuel Assembly Analyses are performed with the three-dimensional *core* simulator and two-dimensional lattice physics codes as discussed in Sections 8.5.1 and 8.5.2.

4.4.5 Nuclear Design Input to Special Events Analyses

4.4.5.1 Stability Analysis

As discussed in Section 9.2 the *Westinghouse* stability analysis methodology utilizes time domain codes.

The stability evaluation is performed with the three dimensional systems transient codes as described in Section 9.2.2. 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. [

]a,c

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 *core* simulator and two-dimensional lattice physics codes as discussed in Section 9.

a, c

Figure 4-1 Iterative Process for Determining Reference Core Design

a, c

Figure 4-2 Data Flow to 3-D Transient Systems Code

5 THERMAL-HYDRAULIC DESIGN

5.1 Summary and Conclusions

5.1.1 Summary

This section provides the *Westinghouse* BWR thermal-hydraulic design bases and describes the methodology used to demonstrate compliance with those bases.

Specifically, this section contains the following:

- The *Westinghouse* thermal-hydraulic design bases,
- The *Westinghouse* methodology used to evaluate compliance with the thermal-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-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 *Westinghouse* thermal-hydraulic methodology and the thermal-hydraulic characteristics of the fuel assemblies:

(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-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 *Westinghouse* as well as non-*Westinghouse* fuel is acceptable for design and licensing applications.

5.2 Thermal-Hydraulic Design Bases

The principal objective of the thermal-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 the 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 the 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 in Section 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 liquid film on the most limiting rod locally has completely evaporated causing a rapid loss of heat transfer capability in that fuel rod for a given pressure, flow, inlet enthalpy and axial power shape. This critical heat flux 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 critical power. 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 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 Report ("COLR")*. The treatment of MCPR for *Westinghouse* and non-*Westinghouse* fuel to assure that the OLMCPR is satisfied during reactor operation and during the design phase *is* 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 *inter-assembly* bypass flow will be maintained within the design range of the plant. If the *inter-assembly* 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 a *Westinghouse* fuel assembly. The Resident fuel assemblies can be *Westinghouse* assemblies of a different design than the Reload fuel or fuel manufactured by a vendor other than *Westinghouse*.

[

]a,c

[

]^{a,c}

The methodology for assuring sufficient hydraulic compatibility is discussed in Section 5.3.3.

5.2.3 Bypass, Water Rod and Water Cross Flow

Basis

The fuel assembly shall be designed to maintain the *inter-assembly* 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 fuel is maintained such that significant boiling will not occur.

Discussion

The Design Basis in Section 5.2.2 addresses *inter-assembly* bypass flow to assure acceptable flow distributions. This design basis is intended to assure that sufficient *inter-assembly* 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 are maintained and that there is sufficient cooling flow to the in-core nuclear instrumentation. 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 *inter-assembly* bypass and interior assembly flow channels is provided in Section 5.3.4.

5.3 Methodology for Thermal-Hydraulic Design

5.3.1 Thermal-Hydraulic Design Models

Accurate computer models simulating the thermal-hydraulic behavior of the plant and the different types of fuel assemblies in the core are established for the following purposes:

(1) Evaluate and establish thermal-hydraulic compatibility of the Reload fuel with the Resident fuel and the core, (if not first core fuel)

(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.

[

]a,c

Computer codes accepted for licensing applications by the NRC are used for all thermal-hydraulic analyses. The steady-state thermal hydraulics performance models are incorporated into the *Westinghouse* BWR three dimensional *core* 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 *inter-assembly* 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-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 (assembly outlet) represents the upper tie plates or outlet spacers (downstream the active fuel zone) and section of the channel above the upper tie plates or outlet spacers, including handle.

The core inlet orifice, bottom nozzle, lower tie plate, spacer grid, assembly outlet, 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 *inter-assembly* 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 *inter-assembly* 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.

[

]a,c

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 *Westinghouse* fuel.

[

]a,c

5.3.1.3 Hydraulic Data for Westinghouse Fuel

Extensive test loop data are used to verify the validity of the analytical modeling of the *Westinghouse* fuel. Specifically, test data are used to verify the modeling of SVEA-type design water cross and water wing modeling, design of bypass holes, tie plates, spacers, and flow distribution to the SVEA-type design sub-bundles as well as friction pressure drop multipliers. *Westinghouse* 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 *Westinghouse* test program is provided by the hydraulic testing of various *Westinghouse* bundle designs summarized in Tables 5-1, 5-2 and 5-3.

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 design specific, the methodology is acceptable for design and licensing purposes for all BWR cores containing *Westinghouse* fuel, as well as for mixed cores containing both *Westinghouse* and non-*Westinghouse* fuel assemblies, provided adequate input data are available.

For *Westinghouse* 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 phenomenological-based expression to critical power data. The SLMCPR is established to protect the fuel from reaching critical power during steady state operation and anticipated transients. The SLMCPR is established to provide that at least 99.9% of the fuel rods avoid reaching critical power.

<u>*Methodology*</u>

[

]a,c

[

]a,c

Discussion

[

]a,c

5.3.2.2 Monte Carlo Safety Limit Evaluation

[

]a,c

[

]a,c

Methodology

[

]a,c

[

]a,c

5.3.2.3 Channel Bow Evaluation

The influence of channel bow on CPR performance is accounted for in the SLMCPR evaluation. Nominal inter-assembly gaps are assigned to the fuel assemblies, and the Monte Carlo method is used to evaluate 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, a *Westinghouse* CPR 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 Optima2 assembly currently being marketed in the U.S. for BWR applications has been accepted by the NRC and is documented in *Reference 66.*

5.3.2.5 Minimum Critical Power Evaluation for Resident Fuel

<u>*Methodology*</u>

If the Resident fuel is a *Westinghouse* design, the CPR is treated in the same manner as for the Reload fuel assembly. A *Westinghouse* CPR 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 a *Westinghouse* design, a CPR 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 a *Westinghouse* design, *Westinghouse* may or may not have direct access to the accepted correlation for the Resident fuel. If *Westinghouse* does have direct access to that correlation, it is used for design and licensing analyses. [

]^{a,c}

[

]a,c

Discussion

[

]a,c

5.3.3 Hydraulic Compatibility

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

[

]a,c

[

]a,c

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.

[

]a,c

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 70* are required for each plant application for the Reload fuel:

[

]a,c

5.4.2 Thermal-Hydraulic Design Input to Nuclear Design

The thermal-hydraulic models are incorporated in the *Westinghouse* 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-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-hydraulic models described in Section 5.3.1.

[

]^{a,c}

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.

[

]a,c

TABLE 5-1

SUMMARY OF THE SVEA-64 THERMAL-HYDRAULIC TEST PROGRAM

a,c

TABLE 5-2

SUMMARY OF THE SVEA-96 AND SVEA-100 THERMAL-HYDRAULIC TEST PROGRAM

a,c

TABLE 5-3

SUMMARY OF THE SVEA-96 OPTIMA, SVEA-96 OPTIMA2 AND SVEA-96 OPTIMA3 THERMAL-HYDRAULIC TEST PROGRAM

a,c

TABLE 5-4

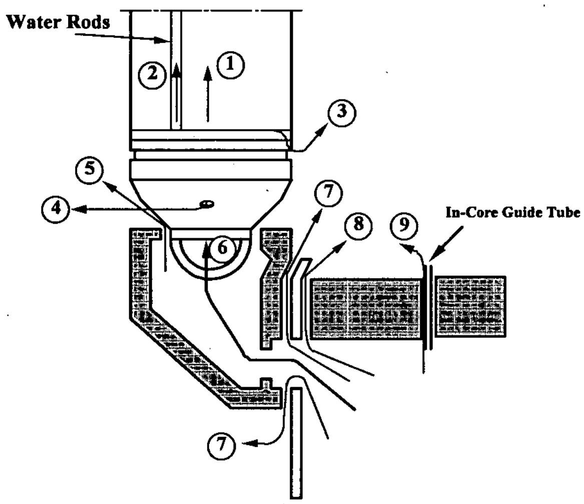
EXAMPLE OF UNCERTAINTIES FOR SLMCPR CALCULATIONS

a,c

[ ]

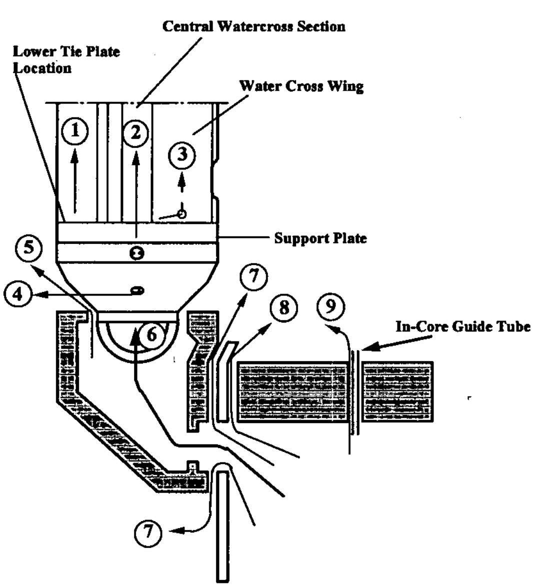
[ ]a,c

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. The schematic for SVEA-96 apply to all SVEA-96 versions, including SVEA-96 Optima2.

a,c

[ ]

Figure 5-2 Calculation Scheme Monte Carlo Safety Limit Methodology

6 RELOAD *LICENSING* ANALYSIS

6.1 Summary and Conclusions

<u>*Summary*</u>

This section describes the *Westinghouse* reload *licensing* analysis process for reload core applications and plant modifications. It also details the *Westinghouse* 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 *licensing* 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 *Westinghouse* BWR reload *licensing* analysis process categorizes safety analysis events and identifies potentially limiting events with respect to the plant design basis. The *Westinghouse* reload *licensing* 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 *Westinghouse* BWR reload *licensing* analysis design bases, methods, and methodology described in Sections 7, 8, and 9 (see also Table 1-2). The reload *licensing* analysis is performed for the fuel and core design developed with the methods and methodology described previously in Section 3, 4, and 5.

<u>*Conclusion*</u>

It is concluded that:

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

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

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

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

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

6.2 Reload *Licensing* 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 *Westinghouse* reload *licensing* analysis process.

The *Westinghouse* reload *licensing* analysis process is shown in Figure 6-2. The *Westinghouse* reload *licensing* 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 *Westinghouse* reload application. This is accomplished through a reload *licensing* analysis process that combines the results of generic safety analysis assessments and plant specific licensing commitment assessments. The *Westinghouse* reload *licensing* 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 *Westinghouse* reload safety analysis methodology to establish the events that are potentially limiting; that in those events that pose the most sever 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 *licensing* analysis process, using the *Westinghouse* 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 *licensing* 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 *Westinghouse* reload *licensing* 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 set of potentially limiting events for the typical plant safety analysis that can be impacted by a reload application or a plant operational modification are identified. This subset of potentially limiting events is evaluated as a part of the plant-specific reload licensing 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 *Westinghouse* 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. A road map of the process is given in Figure 6-2.

Each of the identified events is evaluated for the first *Westinghouse* reload application and for each subsequent reload if an applicable generic or bounding analysis is not available. In addition, *Westinghouse* 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, *Westinghouse* 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 a new fuel type 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.

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 is 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 *Westinghouse* 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 *Westinghouse* reload *licensing* analysis process. The generic safety analysis events that are covered in the *Westinghouse* reload *licensing* 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 *Westinghouse* methodology. The results of these evaluations are included in the plant specific reload safety evaluation.

In addition, the plant safety analysis is reviewed to identify any events different than those generic events identified in Table 6-1 which may be potentially limiting. Potentially limiting events from this additional subset of events, along with their plant-specific commitments, are also included in the reload licensing evaluation.

The next three sections discuss the categorization of events in the three groups: Anticipated Operational Occurrences, Accidents, and Special Events. Section 6.3.2 summarizes the methodology for determining the potentially limiting events to be analyzed for the introduction of *Westinghouse* fuel or a plant modification.

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.

<u>Single Failures as Initiating Events</u>

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) Failure to open or close on demand of any single valve (a check valve is not assumed to close against normal flow).

(2) Failure to start or stop on demand of 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 response 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)). [

]^{a,c} 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 *Westinghouse* 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 *Westinghouse* reload safety analysis process.

Reactor Core Coolant Temperature Decrease Events

Decrease in core coolant temperature includes 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 inadvertant HPCI system start) can lead to significant changes in critical power ratio (CPR). [

]^{a,c} Therefore, evaluation of the loss of feedwater heater in the *Westinghouse* reload *licensing* 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 *Westinghouse* reload *licensing* 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 *Westinghouse* reload *licensing* 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 a 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. [ ]^{a,c}

Events in this category other than the control rod withdrawal error are not evaluated in the standard *Westinghouse* reload *licensing* 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 *Westinghouse* reload *licensing* 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. [

]^{a,c}

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 *Westinghouse* reload *licensing* 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.

[

]^{a,c}

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 *Westinghouse* reload *licensing* analysis process.

Reactor Vessel Coolant Inventory Increase Events

Excess of coolant inventory events can result from a feedwater flow increase greater *than* 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 increase results 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. [

]^{a,c} 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 *Westinghouse* reload *licensing* 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) <u>Component Mechanical Failure:</u> 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

Single Failure in Accidents Evaluation

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 radiological consequences of the spectrum of postulated pipe break locations *are* 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 reactor scram occurs as the MSIVs close and the release of radioactivity is terminated when the MSIVs are fully closed.

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 *Westinghouse* reload *licensing* 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 largest connected piping system. 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 *Westinghouse* reload *licensing* 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 categorized as Accidents.

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 *Westinghouse* reload applications. If required, plant safety analysis is modified to reflect the radiological consequences of the accident. In the *Westinghouse* 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 *Westinghouse* reload *licensing* 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. [

]^{a,c} These events are considered as potentially limiting events in the plant safety analysis. In the *Westinghouse* reload *licensing* analysis process, fuel loading errors are evaluated for each reload to demonstrate conformance to the applicable event acceptance limits.

<u>Recirculation Pump Failure Accident</u>

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.

[

]a,c Therefore, the *Westinghouse* reload *licensing* analysis process does not require reanalysis of these events.

<u>Instrument Line Breaks</u>

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.

[

]a,c Therefore, this event is not evaluated as a part of the standard *Westinghouse* reload *licensing* 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 *Westinghouse* 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 *Westinghouse* reload *licensing* 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 *Westinghouse* reload *licensing* 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 *Westinghouse* reload *licensing* 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 for BWR/2 to BWR/6. For ABWR plants the reactor shutdown is accomplished by automatic initiation of the standby liquid control system or the electrical insertion of the control rods via the fine motion control rod drive system.

6.3.2 Potentially Limiting Events

Not all of the plant's safety analysis events are required to be reanalyzed 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 *Westinghouse* reload *licensing* analysis process.

To identify the potentially limiting events, each event in the plant safety analysis is evaluated to determine that, for a *Westinghouse* 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, *Westinghouse* has developed a process to determine the potentially limiting events that assure coverage of all applicable potentially limiting events. 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 *Westinghouse* reload application or change to plant operating domain justified by the use of *Westinghouse* 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 *Westinghouse* 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. [

]^{a,c}

The loss of coolant accident is evaluated for the initial application of *Westinghouse* reload fuel and then only supplemented to establish the core operating limits associated with new fuel types. [

]^{a,c} 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. [

]^{a,c}

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 *Westinghouse* reload *licensing* 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) reactor isolation at power operation (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 rods 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. [

]^{a,c}

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 *Westinghouse* reload *licensing* 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. [

]a,c

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 consequences. The event acceptance limit for offsite radiological consequences 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 consequences; (2) the ECCS acceptance criteria of 10CFR50.46 (Reference 42); and (3) the primary containment design limits.

The event acceptance limit for offsite radiological consequences is the guideline dose values of 10CFR100, and the event acceptance limit for onsite radiological consequences 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. [

]a,c

Control Rod Drop Accident

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

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

The limits for the control rod drop accident are described in a code specific topical. If no limits are defined in the code specific topical the defined limits are the current interim limits. Once the final limits are defined by the NRC they will be followed.

Fuel Handling Accident

For the fuel handling accident, the figures of merit are the onsite and offsite radiological consequences. The event acceptance limit for offsite radiological consequences is well within (25% or less) the guideline dose value of 10CFR100, and the event acceptance limit for onsite radiological consequences is the limit 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 consequences. The event acceptance limit for offsite radiological consequences is the guideline dose value of 10CFR100, and the event acceptance limit for onsite radiological consequences is the limit identified in GDC 19.

Instrument Line Break

For the instrument line break, the figures of merit are the onsite and offsite radiological consequences. The event acceptance limit for offsite radiological consequences is the guideline dose value of 10CFR100, and the event acceptance limit for onsite radiological consequences is the limit 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 events 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 (Reference 49) 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 consequences; 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 consequences is the guideline dose values of 10CFR100.

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 *Westinghouse* reload *licensing* 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, MELLLA, MELLLA+), 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 *Westinghouse* reload *licensing* analysis process for *a* 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 *the* 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 a *Westinghouse* plant reload application or plant operational modification. A detailed view of the *Westinghouse* 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 *Westinghouse* 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 *core* 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 *licensing* 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.

Fuel Assembly and Core Design

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 *core* 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 the ECCS evaluation.

The 3D *core* 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 *core* 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 *core* simulator is used as a part of the nuclear and thermal-hydraulic 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 CPR correlation as input, the 3D *core* simulator is used to predict the anticipated MCPR throughout the operating cycle.

Static and Quasi Steady-State Transient Events

The 3D *core* simulator is used in the analysis of static and quasi steady-state transients. The 3D *core* 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 *core* simulator also determine the change in LHGR for these events. In addition, the 3D *core* 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. 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 LOCA analysis are required to demonstrate compliance to the ECCS acceptance limits. The LOCA analysis is performed using an approved ECCS evaluation model, which requires detailed inputs to describe the reactor pressure vessel internals, the reactor protection system, the performance of the ECCS equipment and its actuation, fuel performance parameters, and rod peaking parameters. The LOCA analysis inputs make use of a conservative power operating history to develop the fuel performance parameters and a conservative MCPR operating limit to establish conservative boundary conditions for the heat-up calculation. The heat-up analysis establishes the maximum average planar linear heat generation rate (MAPLHGR) operating limits, which ensures compliance with the ECCS acceptance limits during plant operation.

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, manufacturing uncertainties, and a conservative core power distribution as inputs. The operating limit MCPR is established such that the transient CPR will not decrease below the safety limit MCPR. In establishing the operating limit MCPR, the ΔCPR for the AOOs 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.

[

]a,c

[

]a,c

6.6.3 Input Data

There are two basic types of inputs required for the *Westinghouse* reload *licensing* 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 *Westinghouse* based on instructions from *Westinghouse*. Once they are received by *Westinghouse*, they are maintained in accordance with applicable parts of the *Westinghouse* Quality Assurance Program. The key analysis inputs for reload safety analysis are identified on controlled documents. These documents are used in performing the safety analysis to demonstrate that the *Westinghouse* reload application or modified plant operational strategies is acceptable.

The fuel and design inputs to the reload *licensing* analysis process for the *Westinghouse* reload application are developed by *Westinghouse* 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 *Westinghouse* 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 *licensing* 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

<u>Accidents</u>

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

<u>Special Events</u>

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)

Anticipated Transients without Scram

Plant Specific Analyses

TABLE 6-3

DESIGN BASES EVENT ACCEPTANCE LIMITS

<u>Anticipated Operational Occurrences</u>

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/g

Peak Reactor Vessel Pressure ≤ 110% Vessel Design Pressure

Suppression Pool ≤ Heat Capacity Temperature Limit

<u>Accidents</u>

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/g

Fuel Handling Accident
- Offsite Dose ≤ Well within the 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/g

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

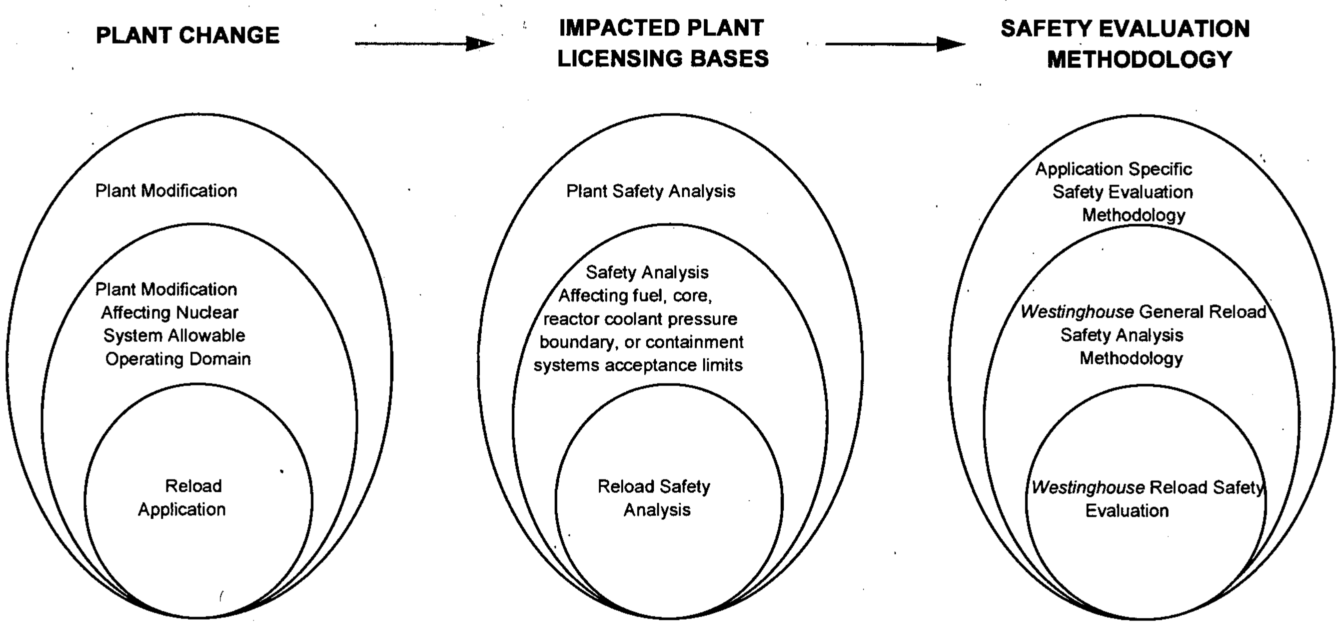


Figure 6-1 Safety Evaluations Process for Plant Modifications

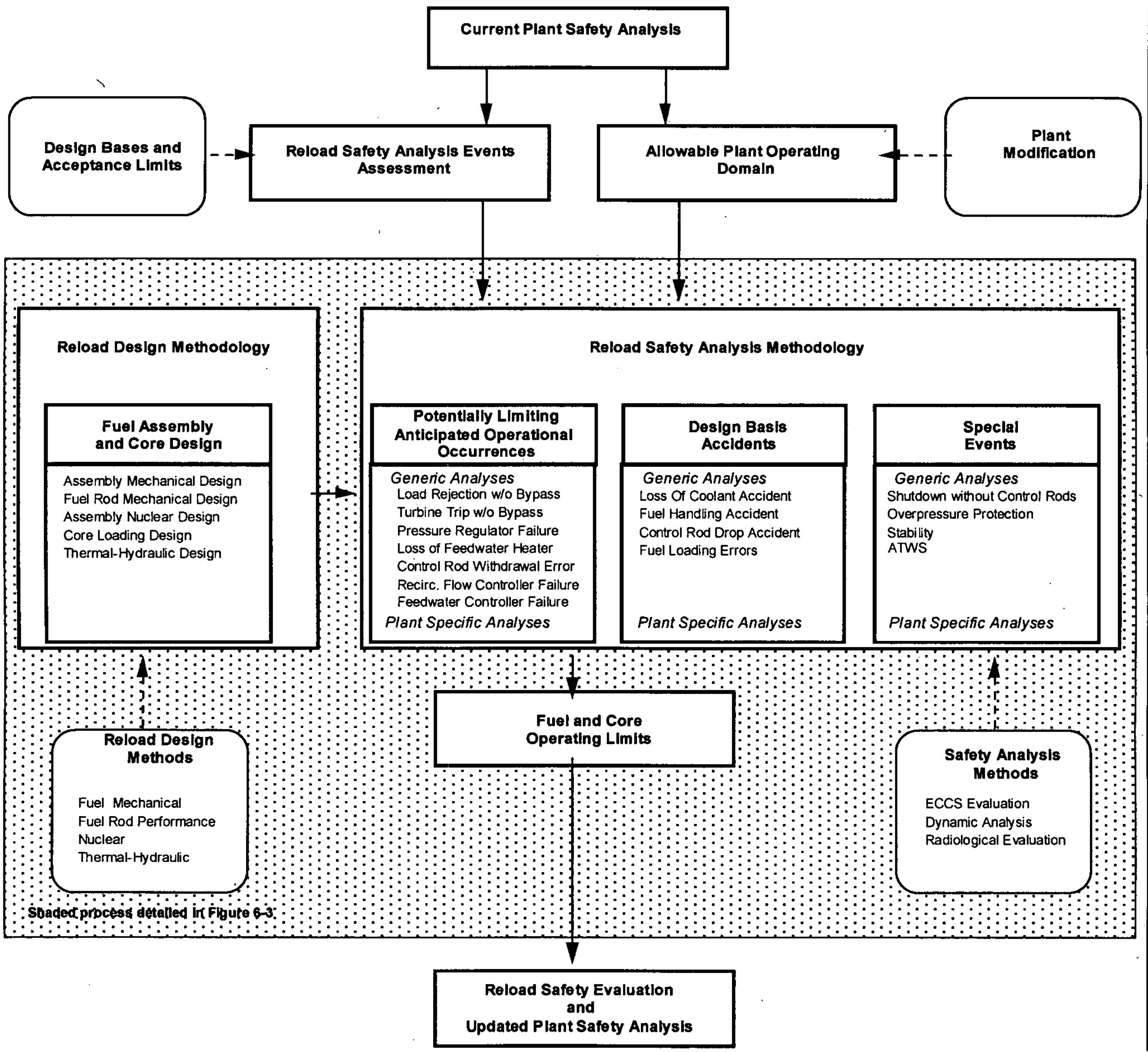


Figure 6-2 Overall Reload Safety Analysis Process

a,c

Figure 6-3 Reload Safety Analysis Methodology Flow Chart

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 *Westinghouse* licensing safety analysis process the potentially limiting anticipated operational occurrences are systematically determined and evaluated using the *Westinghouse* safety analysis methodology. The evaluation determines the plant operating limits within the allowable operating domain for the specific core loading application. The *Westinghouse* licensing analysis methodology for evaluating the potentially limiting AOOs is described in this section.

7.1 Summary and Conclusions

Summary

This section describes, for a *Westinghouse* 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 *Westinghouse* 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 *Westinghouse* 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 Core Design Cladding Integrity

Basis

The minimum allowed value of the Critical Power Ratio (CPR), denoted MCPR, is established such that at least 99.9% of the fuel rods in the core with 95% probability and 95% confidence would not be expected to experience boiling transition during normal operation or anticipated operational occurrences.

Discussion

The acceptance limit for this design criterion 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 core 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 core 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

[

]a,c

7.3 AOO Methodology

The overall *Westinghouse* reload *licensing* 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 *Westinghouse* reload safety 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 are 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 *Westinghouse* 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. These events typically result in a scram being initiated on either the event initiator (e.g., valve position detection) or high neutron flux. "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 and Sections 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. [

]a,c

In addition to the treatment of uncertainties described below a Monte Carlo based uncertainty analysis method (Reference 73) has been submitted to the NRC for review and approval. Once approved, this new method may be used for the determination of the uncertainties in transient analysis.

To remain in compliance with SER Condition 2 of CENPD-300 Revision 0 Method A remains generically applicable for use in OLMCPR uncertainty determinations. Methods B, C, and D are not generically approved for use in the determination of OLMCPR uncertainty. Use of Methods B, C, or D for OLMCPR uncertainty determination needs to be sufficiently justified in a site specific application.

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. [

]a,c

[ ]a,c

Approach A

Methodology

[ ]a,c

Discussion

[ ]a,c

Approach B

Methodology

[ ]a,c

[

]a,c

Discussion

[

]a,c

Approach C

Methodology

[

]a,c

[

]a,c

Discussion

[
]a,c

Approach D

Methodology

[

]a,c

[

]a,c

Discussion

[

]a,c

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).

[

]a,c

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.

[

]a,c

[

]a,c

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

<u>Methodology</u>

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). [

]a,c

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). [

]a,c

Discussion

[

]a,c

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:

[

]a,c

[

]a,c

Discussion

[

]a,c

7.4 Fast Transient Methodology

CENPD-300 Revision 0 described the fast transient methodology for the limiting transients identified in Section 7.3.1:

- Generator Load Rejection Without Bypass
- Turbine Trip Without Bypass
- Feedwater Controller Failure - Maximum Demand
- Pressure regulator Failure - Closed

For addressing all fast transients required for a first core application, i.e. both limiting and non-limiting transients, a new Licensing Topical Report has been submitted for NRC review (Reference 73). For the fast transients listed above the methodology is consistent with Revision 0 of this topical report (CENPD-300-P-A, Revision 0).

The methodology described in Reference 73 is code-independent and is applicable to both 1D and 3D transient analysis codes described in Appendix A.

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, CPR factors, detector constants, etc.) required for the three-dimensional 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

[

]a,c

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. [

]a,c

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

[

]a,c

[

]a,c

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. Newer boiling water reactors such ABWR are equipped with a redundant automated thermal limit monitor (ATLM) system.

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, BWR/6, and ABWR plants are subsequently described.

<u>BWR/3-5 Plants</u>

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:

[

]^{a,c}

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

[

]^{a,c}

[

]a,c

(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 *Westinghouse* NRC approved three dimensional core simulator. The full core is modeled to describe detector strings and error rods as accurately as possible.

[

]a,c

[

]a,c

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 cannot be assumed to be the fully inserted group and all intermediate control rod positions for the error rod must be investigated.

Consequently, the same calculation 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.

ABWR Plants

In the ABWR, the automated thermal limit monitor (ATLM) and the multi-channel rod block monitor (RBM) subsystem logic issues a rod block signal used in the reactor coolant isolation system (RCIS) logic to enforce rod blocks. This feature acts to prevent fuel damage by ensuring that the MCPR and maximum linear heat generation rate (MLHGR) do not violate the fuel thermal operating and safety limits. The operating thermal limits rod block function will block rod withdrawal when the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no RWE transient event and thus the event is not analyzed as an AOO.

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 *Westinghouse* 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:

[

]a,c

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

]a,c

[

]a,c

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

a,c

[ ]

Figure 7-1 Fast Transient Analysis Code Interfaces

a,c

Figure 7-2 Nodal Neutronic Data for Fast Transient Calculations

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 *Westinghouse* reload fuel licensing analysis process, the postulated accidents that require re-evaluation for the introduction of *Westinghouse* 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 *Westinghouse* licensing analysis methodology for evaluating the potentially limiting accidents is described in this section.

8.1 Summary and Conclusions

Summary

This section describes, for a *Westinghouse* 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 *Westinghouse* 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 pipe breaks to be considered encompass all sizes and locations up to and including the rapid circumferential failure of the largest piping system connected to the Reactor Pressure Vessel (RPV). 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

<u>Basis</u>

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

<u>Discussion</u>

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 these design acceptance criteria are not violated. The plant MAPLHGR operating limit or LHGR operating limit ensures compliance with this design basis.

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" (10CFR50.46).

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" (10CFR50.46).

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 *Westinghouse* licensing 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 water from certain ECCS equipment. Once the RPV has been reflooded, all fuel cladding temperatures would return to near saturation temperatures. 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 provided ECCS performance is not changed. Hence this criterion is not required to be addressed for *Westinghouse* reload applications.

8.2.2 Event Description

The LOCA event described here is for a large double-ended guillotine break in the recirculation suction line of a modern BWR with two external recirculation loops that drive the internal jet pumps. Other plant designs have different transient characteristics. For example, the ABWR design has internal recirculation pumps and no large piping systems connected to the RPV below the top of active fuel. This design does not experience significant uncovery of the core.

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 RPV 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 isolation of the steam lines. Following reactor shutdown the pressure begins to fall rapidly. After several seconds the two-phase mixture level in the downcomer falls to the jet pump suction elevation. Uncovery of the jet pump suction lines increases the fluid quality upstream of the break resulting in a sudden decrease in break mass flow rate.

Flashing in the jet pumps and subsequently in the lower plenum occurs when the pressure decreases below the local saturation pressure. This results in a two-phase mixture level rise in the core and downcomer. Following this level swell, the continued inventory decrease results in falling mixture level in the downcomer which initiates the ECCS. Core two-phase mixture level will drop exposing the fuel rods to a steam environment. The downflow of injected coolant from the upper plenum into the core and the upflow of steam from lower plenum flashing provide 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 cladding temperature transient is terminated by emergency core cooling refilling the RPV 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 decay heat generation.

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 *Westinghouse* reload *licensing* 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 licensing evaluations of new fuel designs introduced in a plant specific 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. [

]^{a,c}

8.2.3.3 Design Basis Event Analysis

Methodology

The plant system response to the postulated design basis 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 MAPLHGR limit is determined that ensures compliance with the LOCA design acceptance criteria.

Discussion

The *Westinghouse* ECCS Evaluation Model contains sufficient conservatism to assure that the LOCA design acceptance criteria are met with a significant safety margin. [

]^{a,c}

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:

[

]^{a,c}

Discussion

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

8.2.3.5 MAPLHGR Operating Limit

Fuel type specific operating limits are included in the plant technical specifications to ensure that ECCS acceptance criteria are not violated. The fuel type specific operating limit established to meet ECCS LOCA requirements is the MAPLHGR.

Methodology

The plant MAPLHGR operating limit is specified for each fuel type present in the cycle. The plant MAPLHGR operating limit is the most restrictive of:

(1) The MAPLHGR established to comply with the LOCA ECCS acceptance criteria, and

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

Discussion

[

]^{a,c}

8.3 Control Rod Drop Accident

The Control Rod Drop Accident (CRDA) Methodology has been provided in detail in NRC approved topical reports such as Reference 33 and Reference 72. These NRC approved reports describe the *Westinghouse* design basis and the analysis methodology for the CRDA analysis. Note that the need for the CRDA analyses for BWR 2 through 6 designs is because the locking piston in the control rod drive (CRD) mechanism cannot detect separation of the control rod from the drive mechanism during normal rod movements. Newer boiling water reactor designs such as advanced boiling water reactors (ABWR) are equipped with the fine motion control rod drive (FMCRD) which is designed to detect the separation of the control rod from the drive mechanism. Two redundant switches are provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. Actuation of either of these switches cause an immediate rod block and initiates an alarm in the control room. Therefore cycle specific CRDA analyses are not necessary for the ABWR.

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 well within 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/2 through BWR/5 and ABWR plants, 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/6 plants, the refueling accident can occur within containment or within the reactor 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

The Fuel Handling Accident analysis in Revision 0 of this document assumes that a competitor fuel assembly, referred to as the "reference assembly", has previously been evaluated for a licensing basis fuel handling accident for limiting conditions in the plant to which a Westinghouse feed assembly, referred to as the "new assembly", is to be installed. This situation is typically encountered when a Westinghouse reload fuel assembly is initially loaded in a core containing Westinghouse fuel with a design different than the new assembly design or competitor fuel and does not account for the case when Westinghouse fuel is loaded when the plant initially starts up. Under this circumstance, the new assembly being installed is evaluated and becomes the reference assembly to which comparisons are made with subsequent Westinghouse feed fuel design assemblies. The remainder of Section 8.4 assumes the existence of a Reference Fuel analysis. The extension in this paragraph assures that there will be a

Reference Fuel analysis for all loading combinations of Westinghouse and non-Westinghouse fuel.

Based on the design of *Westinghouse* reload fuel assemblies, the introduction of *Westinghouse* fuel into the core has typically not increase*d* the potential of fission product release to the environment in the past or the dose to control room personnel as a result of a fuel handling accident. This conclusion has been a consequence of the structural characteristics of *Westinghouse* reload fuel. *Westinghouse* reload fuel has been typically found to be lighter than other fuel designs evaluated in the past and more resistant to failure mechanisms associated with fuel handling accidents.

To assess potential fuel handling accidents for *Westinghouse* 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 *Westinghouse* 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 Assembly* relative to the weight of 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 the *New 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.

<u>Fuel Bundle Weight</u>

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 Assembly* is heavier than the *Reference Assembly*, the number of failed fuel rods may increase if the heavier *New Assembly* is dropped on *Reference 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 the 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 make 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.

[

]^{a,c}

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.

[

]^{a,c}

[

]a,c

[

]a,c

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.

[

]a,c

[

]a,c

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 an 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 *Westinghouse* cycle specific safety analysis process. The SLMCPR is used as the event acceptance limit for this accident.

Discussion

[

]^{a,c}

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.

[

]^{a,c}

[

]a,c

8.5.1.3 Analysis Methodology

The mislocated assembly analysis is performed under the following assumptions:

[

]a,c

[

]^{a,c}

8.5.2 Rotated Fuel Assembly Accident

8.5.2.1 Design Bases

Basis

This event is considered to be an accident in the *Westinghouse* cycle specific safety analysis process. The SLMCPR is used as the event acceptance limit for this accident.

Discussion

[

]a,c

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 tends 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

[

]a,c

[

]a,c

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 *Westinghouse* safety analysis methodology for evaluating Special Events is described in this section.

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

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

In addition, *Westinghouse* 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 a *Westinghouse* reload application. Four Special Events are addressed in the *Westinghouse* safety analysis methodology.

The *Westinghouse* 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 cycle specific safety evaluations and plant modification evaluations, as required. *Westinghouse* also has advanced stability tools and 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 *Westinghouse* methodology performs Reactor ASME Overpressure Protection analysis to confirm for each 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 application. The *Westinghouse* 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 core design. As discussed in Section 6.3.1.3, it is not necessary to evaluate ATWS events for the use of *Westinghouse* fuel. However, the potential does exist for performing ATWS evaluations for certain types of plant modifications. The *Westinghouse* safety analysis methodology does have the capability for evaluating ATWS events, if required in the evaluation of plant modifications.

<u>*Conclusions*</u>

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

Westinghouse has analysis codes and methodologies to perform core thermal-hydraulic stability evaluations for plant specific reload applications and plant modifications as required. *Westinghouse* uses time domain codes for stability analysis (see Table 9-1). These stability analysis tools can be used for 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 *Westinghouse* stability analysis methodology, and the plant application methodology.

9.2.1 Design Bases

<u>*Basis*</u>

The allowable plant operating domain for the 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.

<u>*Discussion*</u>

The above design basis establishes reactor thermal-hydraulic stability compliance with GDC 12 of 10CFR50 Appendix A (Reference 42). Design requirements are put on the 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 *Westinghouse* 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 safety analysis process.

The *Westinghouse* 3D time domain codes are described in References 44 *and 72*. References 44 *and 72* provide a description of the codes and qualification for core and channel stability performance evaluations. Three dimensional transient stability analysis methods are used in the *Westinghouse* stability methodology. Licensing Topical Report CENPD-295-P-A (Reference 45) together with Reference 74, submitted for review, provide a description of general stability analysis methodology using the stability codes.

9.2.3 Plant Reload Application Methodology

Methodology

The stability evaluation performed for a specific plant application will be consistent with plant-specific licensing commitments. The stability evaluation will use approved stability methods and 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. *Westinghouse* shall use an NRC approved 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 licensing analysis. 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 is the ASME Code upset limit of 110 % of the reactor pressure vessel design pressure as stated in Section 6.4.3.

9.3.2 Overpressurization Protection Methodology

Methodology

The most severe pressurization event is analyzed for each cycle specific licensing analysis 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:

[

]^{a,c}

The overpressurization MSIV closure event is analyzed with the NRC approved dynamic analysis methods.

[

]^{a,c}

[

]^{a,c}

Discussion

The overpressurization MSIV closure event could be treated as an emergency condition, with acceptable results compared to the ASME emergency condition limits (i.e., the reactor pressure acceptance limit of 120% of design pressure). However, the Westinghouse 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 xenon-free core and no movement of the control rods. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical poison system, satisfying GDC-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 NRC approved lattice physics code and 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.

[

]a,c

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. [

]a,c

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 *core* simulator Reference Core model are performed with the appropriate boron concentration. The lattice physics methods are used to explicitly model the fuel assembly contained in the reference core.

The moderator cross section are developed assuming a uniform distribution of the boron concentration. These cross sections are utilized in the three-dimensional *core* 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).

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

[

]a,c

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 *a Westinghouse* fuel design is described below.

Methodology

Each new *Westinghouse* 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. [

]a,c Once the fuel design in confirmed not to have a significant impact in the current ATWS analysis, it is considered acceptable. Methodology for ATWS analysis is contained in Reference 73.

Discussion

[

]a,c

TABLE 9-1

WESTINGHOUSE STABILITY ANALYSIS TOOLS

Tool	Methods	Methods Qualification	Analysis Methodology
3D Time Domain Codes	CENPD-294-P-A (RAMONA code) WCAP-16747-P-A (POLCA-T code)	CENPD-294-P-A WCAP-16747-P-A	CENPD-295-P-A WCAP-16747-P-A WCAP 17137-P

CENPD-294-P-A (Reference 44)
CENPD-295-P-A (Reference 45)
WCAP-16747-P-A (Reference 72)
WCAP-17137-P (Reference 74)
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) (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) Described in WCAP 17137 (Reference 74)

Westinghouse Stability Evaluation	Described in CENPD-295-P-A (Reference 45) Described in WCAP 17137 (Reference 74)

10 REFERENCES

1. A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactors, Westinghouse Report WCAP-10106-P-A (proprietary), WCAP-10999-NP-A (nonproprietary), September 1985.

2. Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Programs for Boiling Water Reactors, Westinghouse Report WCAP-10841-P-A (proprietary), WCAP-10842-NP-A (nonproprietary), June 1985.

3. CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors, Westinghouse Report WCAP-10107-P-A (proprietary), WCAP-10108-NP-A (nonproprietary), March 1986.

4. Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification, Westinghouse Report WCAP-11284-P-A (proprietary), WCAP-12391-NP-A (nonproprietary), November 1989.

5. Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity, Westinghouse Report WCAP-11427-P-A (proprietary), WCAP-12392-NP-A (nonproprietary), November 1989.

6. BISON - A One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Vol. 1 -Theory and Vol. 2 Qualification, WCAP-11236 (proprietary), August 1986.

7. Not used

8. Not used

9. Westinghouse Reference Safety Report for BWR Fuel, Westinghouse Report WCAP-11500 (proprietary), August 1987.

10. Westinghouse Correlation for Predicting CPR Margin in Boiling Water Reactors, Westinghouse Report WCAP-10972 (proprietary), December 1986.

11. Seismic Evaluation of the Westinghouse QUAD+ Fuel, Westinghouse Report WCAP-11686 (proprietary), December 1986.

12. Letter C. O. Thomas (NRC) to E. P. Rahe (W), "Acceptance for Referencing of Licensing Topical Report WCAP-10106, 'A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactor', " September 3, 1985.

13. Letter M. W. Hodges (NRC) to W. J. Johnson (W), "Acceptance for Referencing of Licensing Topical Report WCAP-10841, 'Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Programs for Boiling Water Reactor', " June 14, 1988.

14. Letter C. O. Thomas (NRC) to E. P. Rahe (W), "Acceptance for Referencing of Licensing Topical Report WCAP-10107(P). 'CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors' ," October 11, 1985.

15. Letter M. W. Hodges (NRC) to W. J. Johnson (W), "Acceptance for Referencing of Licensing Topical Reports WCAP-11284 and WCAP-11427 Regarding Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model," August 22, 1989.

16. Letter A. L. Thadani (NRC) to W. J. Johnson (W), "Acceptance for Referencing of Licensing Topical Report WCAP-11236 Regarding Westinghouse Boiling Water Reactor Transient Analysis Code, 'BISON', " October 29, 1989.

17. Not used

18. Not used

19. 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.

20. CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors, ABB Report BR-91-255-P-A, rev. 1 (proprietary), BR-91-266-NP-A (non-proprietary), May 1991.

21. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification, ABB Report RPB-90-93-P-A (proprietary), RPB-90-91-NP-A (nonproprietary), October 1991.

22. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity, ABB Report RPB-90-94-P-A (proprietary), RPB-90-92-NP-A (nonproprietary), October 1991.

23. BISON - A One Dimensional Dynamic Analysis Code for Boiling Water Reactors, ABB Report RPA 90-90-P-A (proprietary), RPA 90-90-NP-A (nonproprietary), December 1991.

24. Not used

25. Not used

26. Letter A. L. Thadani (NRC) to J. Lindner (ABB Atom), "Designation of ABB Atom Topical Reports Related to Licensing of ABB Atom Reload Fuel," June 18, 1992.

27. 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.

28. ABB Atom High Worth Control Rods for US BWRs, Rod Drop Accident Analysis, ABB Report RPA-89-053 (proprietary), August, 1989.

29. ABB Atom Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors, The RAMONA-3B Computer Code, ABB Report RPA-89-112 (proprietary), November, 1989.

30. Letter G. Heimersson (ABB Atom) and A. E. Scherer (ABB CE) to R. C. Jones, Jr. (NRC), "ABB BWR Fuel and Core Component Licensing Activities," September 4, 1992.

31. Letter A. C. Thadani (NRC) to N. O. Jonsson (ABB) "Acceptance for Referencing of ABB Report UR 89-210: 'SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-Bundle,' (TAC No. M75938)," January 13, 1993.

32. Letter A. C. Thadani (NRC) to D. B. Ebeling-Koning (ABB) "Acceptance for Referencing of ABB CE Supplemental Information Regarding the ABB SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-bundle, (TAC No. 86121) July 12, 1993.

33. 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.

34. Intentionally Deleted.

35. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel, ABB Report CENPD-283-P-A (proprietary), CENPD-283-NP-A (non-proprietary), July 1996.

36. Fuel Rod Design Methods for Boiling Water Reactors, ABB Report CENPD-285-P-A (proprietary), CENPD-285-NP-A (non-proprietary), July 1996.

37. Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, ABB Report CENPD-287-P-A (proprietary), CENPD-287-NP-A (non-proprietary), July 1996.

38. ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel, ABB Report CENPD-288-P-A (proprietary), CENPD-288-NP-A (non-proprietary), July 1996.

39. BISON - One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification ABB Report CENPD-292-P-A (proprietary), CENPD-292-NP-A (non-proprietary), July 1996.

40. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 1 to Code Description and Qualification, ABB Report CENPD-293-P-A (proprietary), CENPD-293-NP-A (non-proprietary), July 1996.

41. ABB Correlation for Predicting CPR Margin in Boiling Water Reactors ABB Report CENPD-286-P-A (proprietary) CENPD-286-NP-A (nonproprietary) (ABB designation of WCAP-10972 under NRC review).

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

43. U.S. Nuclear Regulatory Commission Standard Review Plan, NUREG-0800.

44. Thermal-Hydraulic Stability Analysis Methods for Boiling Water Reactors CENPD-294-P-A (proprietary), CENPD-294-NP-A (non-proprietary), July 1996.

45. Thermal-Hydraulic Stability Analysis Methodology for Boiling Water Reactors CENPD-295-P-A (proprietary), CENPD-295-NP-A (non-proprietary), July 1996.

46. General Electric Report NEDE-10958, General Electric BWR Thermal Analysis Basis (GETAB), November, 1973.

47. U.S. Nuclear Regulatory Commission Regulatory Guides.

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

49. ASME Boiler and Pressure Vessel Code, Section III, American Society of Mechanical Engineers.

50. Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Regulatory Guide 1.25, March, 1972.

51. S. Persson, A. R. Massih, "Some High Burnup Phenomena in BWR's: Fission Gas Release and Cladding Hydrogen Pickup", Paper HPR-339/10, Halden Project Group Meeting, Bolkesjo, Norway, June, 1991.

52. United States Atomic Energy Commission, "Opinion of the Commission -- Acceptable Criteria for Emergency Core Cooling Systems -- Light-Water Cooled Nuclear Reactors," Docket No. RM-50-1, December 28, 1974.

53. Compendium of ECSS Research for Realistic LOCA Analysis, NUREG-1230, U.S. Nuclear Regulatory Commission, September 1988.

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

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

56. Not used

57. Not used

58. K. J. Doran, R. S. May, and R. E. Engel, STARS - Statistical Transient Analysis by Response Surface Program: Theory and User's Manual, EPRI/NP-7558-CCML, Electric Power Research Institute, January, 1992.

59. Application Topical Report for BWR Design and Analysis, Washington Public Power Supply System Topical Report WPPSS-FTS-131 (TAC Nos. M77048 and M81723), October 25, 1993.

60. U. A. Ansari, V. Chandola, M. P. LeFrancois, and J. D. Robichaud, "Application of One-Dimensional Kinetics to Boiling Water Reactor Transient Analysis Methods," Yankee Atomic Electric Company Nuclear Services Division Topical Report YAEC-1693, Rev. 1, Nov., 1989.

61. Statistical Combination of Uncertainties, Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 & 2, Combustion Engineering Topical Report CEN-124(B)-P, January, 1980.

62. Core Design and Operation Data for Cycles 1 and 2 of Peach Bottom 2, EPRI NP-563, Electric Power Research Institute, June 1978.

63. Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2, EPRI NP-564, Electric Power Research Institute, June 1978.

64. RAMONA Analysis of the Peach Bottom-2 Turbine Trip Transients, EPRI NP-1869, Electric Power Research Institute, June 1981.

65. The advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors, Westinghouse Report CENPD-390-P-A (proprietary), CENPD-390-NP-A (nonproprietary), December 2000.

66. 10x10 SVEA Critical Power Experiments and CPR Correlation: SVEA-96 Optima2, Westinghouse Report WCAP-16081-P-A (proprietary), WCAP-16081-NP-A (nonproprietary), March 2005.

67. Westinghouse BWR ECCS Evaluation model: Supplement 2 to Code Description, Qualification and Application, Westinghouse Report WCAP-15682-P-A (proprietary), WCAP-15682-NP-A (nonproprietary), April 2003.

68. 'Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel, Westinghouse Report WCAP-16078-P-A (proprietary), WCAP-16078-NP-A (nonproprietary), July 2004.

69. Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1, Westinghouse Report WCAP-15836-P-A (proprietary), WCAP-15836-NP-A (nonproprietary), April 2006.

70. BWR Fuel Assembly Mechanical Design Methodology - Supplement 1, Westinghouse Report WCAP-15942-P-A (proprietary), WCAP-15942-NP-A (nonproprietary), March 2006.

71. Supplement 2 to BISON Topical Report RPA 90-90-P-A, Westinghouse Report WCAP-16606-P-A (proprietary), WCAP-16606-NP-A (nonproprietary), January 2008.

72. POLCA-T: System Analysis Code with Three-Dimensional Core Mode, Westinghouse Report WCAP-16747-P-A (proprietary), WCAP-16747-NP-A (nonproprietary), September 2010.

73. Fast Transient and ATWS methodology, Westinghouse Report WCAP-17203-P (proprietary). Submitted to the NRC June 2010.

74 Westinghouse Stability Methodology for the ABWR, Westinghouse Report WCAP-17137-P (proprietary). Submitted to the NRC September 2010.

75. Westinghouse Containment Analysis Methodology, Westinghouse Report WCAP-16608-P-A (proprietary), WCAP-16608-NP-A (nonproprietary), March 2009.

76 NAI 8907-06, GOTHIC Containment Analysis Package Technical Manual.

APPENDIX A: DESCRIPTION OF CODES

A.1 Mechanical Design

A.1.1 Fuel Rod Performance Codes

A.1.1.1VIK-3

The computer code VIK-3 calculates stresses in light water reactor (LWR) fuel rod cladding as a function of fuel burnup or irradiation time. Both fully re-crystallized and cold work stress-relieved Zircaloy cladding can be evaluated. VIK-3 has an option allowing its execution in conjunction with STAV in order to provide cladding stress evaluations as a function of fuel rod burnup based on materials properties and STAV calculated parameters.

The code consists of a number of subroutines, each one calculating the stress due to the different sources or load cases. Stress levels are calculated at the clad inner and outer radii at three axial locations, namely at a spacer, between spacers and at the bottom end plug. Depending on the origin of the stress and on geometrical and material discontinuities in the design, each stress is classified with the appropriate stress category. The effective stresses are calculated using the Tresca relationship in accordance with Section III of the ASME Code.

A complete description of the VIK-3 code is provided in *Reference 69.*

A.1.1.2STAV7.2

The STAV7.2 code is the latest version of the STAV fuel rod performance code series developed and used at *Westinghouse*. This tool enables the evaluation of the steady-state performance of fuel rods under the conditions prevailing in a light water reactor (LWR). STAV7.2 can model both UO_2 and $(U,Gd)O_2$ fuel.

STAV7.2 is the primary analysis code used in fuel thermal mechanical design process.

STAV7.2 calculates the variation over time of all significant fuel rod performance parameters including fuel and cladding temperatures, fuel densification, fuel swelling, fission product gas release, rod internal pressure, and pellet-cladding gap conductance. Stresses and strains in the cladding due to elastic, thermal, creep and plastic deformations are calculated. Cladding oxidations is modeled and its influence on other parameters considered. Other submodels include burnup-dependent radial power distributions for both UO_2 and $(U,Gd)O_2$ fuel, fuel grain growth, and helium release.

For example, in the reload safety analysis process, STAV7.2 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 STAV7.2 code is provided in *Reference 69.*

A.1.1.3 COLLAPS-II

COLLAPS-II is used by *Westinghouse* for prediction of cladding ovality in BWRs fuel rods 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 pressure. The cross section of the tube is assumed to have a slight initial deviation from circularity. 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-II code is provided in *Reference 69.*

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 65.*

A.2.1 Two Dimensional Lattice Design

A.2.1.1PHOENIX

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 *Westinghouse* depletion program for BWR 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-groups homogenized nuclear data 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 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 a boiling water reactor (BWR) core. The POLCA code is described in Reference 65.

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 void content affects 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.

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. The core physics model of POLCA is also included in the system analysis code POLCA-T; see Section A.4.2.2.

A.3 Thermal-Hydraulics Design

A.3.1 POLCA

Westinghouse has utilized the CONDOR code for the evaluation of the steady-state thermal-hydraulic performance of BWR primary systems. The same models were also used as the thermal-hydraulic module of the three-dimensional core simulator code, POLCA. The complete CONDOR code functionality is now included in the nodal code POLCA as described in Reference 65.

POLCA 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 CONDOR code is provided in Reference 20 and of the POLCA code in Reference 65.

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.4.2.2 POLCA-T

POLCA-T is an advanced dynamic system analysis code with the three dimensional (3D) core physics modeling capabilities described by the nodal code POLCA presented above.

POLCA-T is a computer code for transient thermal hydraulic and neutron kinetics analysis of BWR plants. It can be used as a general tool for advanced simulation of single and two phase flow systems including non condensable gases. The code has a full-3D coupled core neutronics/thermo-hydraulics model where each fuel assembly in the reactor core can be explicitly represented in the thermal hydraulic model. The reactor pressure vessel, external pump loops, steam system, feedwater system, emergency core cooling systems, control systems, and steam relief system can be modeled in detail.

POLCA-T is, as RAMONA, specifically designed to simulate normal and abnormal operational plant transients, as well as accidents and special events like ATWS and stability requiring complete three-dimensional representation. Because of its unique feature of combining full 3D modeling of the reactor core and transient plant response, it is particularly suited for transients involving significant within-the-core effects.

A detailed description of the modeling capabilities of POLCA-T is provided in Reference 72. These capabilities make POLCA-T suitable to replace both RAMONA and BISON in their specific applications. The use of POLCA-T for those applications is being introduced in a staged process. The first two applications Control Rod Drop Accident (CRDA) Analysis and Stability Analysis have been reviewed and approved by the NRC. Subsequent applications (including Transient Analysis and ATWS) will be submitted prior to their use. Each application is included as an appendix to the code description which contains the evaluation model and the qualification of the code for performing the intended analysis.

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-3D. 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-3D calculates the detailed temperature distribution at a given axial cross section of the assembly analyzed by DRAGON. Its boundary conditions are supplied by GOBLIN-EM and DRAGON.

A detailed description of these codes is provided in References 21, 40, *67 and 68.*

A.4.4 Intentionally Deleted

A.4.4.1 Intentionally Deleted

A.4.4.2 Intentionally Deleted

A.5 Statistical Analysis

A.5.1 Industry Accepted Codes

A.5.1.1SIGMA

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 involve 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 *Westinghouse* to perform statistical analysis that are approved by NRC for use by the utility. The utility can provide these codes to *Westinghouse* 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.

A.6 Containment Analysis

A.6.1 GOTHIC

Westinghouse uses the GOTHIC computer code to perform design-basis containment analyses. The code has been developed by Numerical Applications Incorporated (NAI) with funding by the Electric Power Research Institute (EPRI).

GOTHIC solves the integral form of the conservation equations for mass, momentum, and energy for multi-components, two-phase flow. The conservation equations are solved for three fields; continuous liquid, liquid drops, and steam/gas phase. The three fields may be in thermal non-equilibrium within the same computational cell. This treatment allows the modeling of sub-cooled drops (e.g. containment spray) falling through an atmosphere of saturated steam. The gas component of the steam/gas field may be comprised of up to eight different non-condensable gases with mass balances performed for each component. Relative velocities are calculated for each field as well as the effects of two-phase slip on the pressure drop. Heat and mass transfer between the phases, surfaces, and the fluid is also allowed.

The GOTHIC code is capable of performing calculations in three modes. The code can be used in the lumped parameter nodal network mode, the two-dimensional finite difference mode, and the three-dimensional finite difference mode. The code also contains the options to model a large number of structures and components such as heated and unheated conductors, pumps, fans, valves, heat exchangers, ice condensors, etc. These components can be coupled to simulate typical containment systems.

A detailed description of the GOTHIC code is provided in Reference 76. *Westinghouse* methodology for Mark I containment analyses and for the ABWR containment analysis is provided in Reference 75.