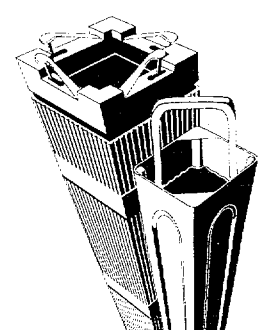
SIEMENS

EMF-2157(NP)
Revision 0

Boiling Water Reactor Licensing Methodology Compendium

December 1999



Siemens Power Corporation

Nuclear Division

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ISSUED IN SPC DOCUMENT SYSTEM
DATE: 12/23/99

EMF-2157(NP)
Revision 0

Boiling Water Reactor
Licensing Methodology Compendium

Prepared: H. D. Curet — 12/20/99
H. D. Curet, Manager
Product Licensing
Date

Approved: R. R. Schnepp for MEG — 12/21/99
M. E. Garrett, Manager
BWR Safety Analysis
Date

Approved: O. C. Brown — 20 Dec 99
O. C. Brown, Manager
BWR Neutronics
Date

Approved: for R ... — 12/22/99
T. M. Howe, Manager
Product Mechanical Engineering
Date

Approved: ... for — 12/21/99
D. J. Denver, Manager
Commercial Operations
Date

Approved: ______________________
J. F. Mallay, Director
Regulatory Affairs
Date

arn

U.S. Nuclear Regulatory Commission
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Abstract

This report is a compendium of Siemens Power Corporation (SPC) methodologies and design criteria which are described in topical reports that the NRC has found acceptable for referencing in boiling water reactor (BWR) licensing applications. This compendium provides a concise, organized source for NRC-approved BWR topical reports.

The methodologies and topical reports addressed in this report are designed to give BWR licensees using SPC fuel and co-resident fuel the methodologies needed to conform to their original licensing bases and to meet cycle-specific parameter limits that have been established using NRC-approved methodologies. These methodologies may also be used to predict changes to limits consistent with all applicable limits of the plant safety analysis that are addressed in the UFSAR.

Nature of Changes

Item	Paragraph or Page(s)	Description and Justification
1.	All	This is a new document.

Contents

Tables

Nomenclature

ANF	Advanced Nuclear Fuels
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
BOC	Beginning-of-Cycle
BWR	Boiling Water Reactors
CHF	Critical Heat Flux
CHFR	Critical Heat Flux Ratio
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRDA	Control Rod Drop Accident
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
ENC	Exxon Nuclear Company
EOC	End-of-Cycle
EOL	End-of-Life
FCTF	Fuel Cooling Test Facility
FDL	Fuel Design Limit
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HPCI	High Pressure Coolant Injection
LBLOCA	Large Break Loss-of-Coolant Accident
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
MCHFR	Minimum Critical Heat Flux Ratio
MCPR	Minimum Critical Power Ratio
MEOD	Maximum Extended Operating Domain
MFC	Manual Flow Control
MFLCPR	OLMCPR/Limiting Assembly MCPR
MSIV	Main Steam Isolation Valve
MWR	Metal-Water Reaction
OLMCPR	Operating Limit MCPR
PAPT	Protection Against the Power Transient
PCT	Peak Cladding Temperature

POI	Plane-of-Interest
PWR	Pressurized Water Reactor
RIA	Reactivity Initiated Accident
RPS	Recirculation Pump Seizure
SBLOCA	Small Break Loss-of-Coolant Accident
SER	Safety Evaluation Report
SLMCPR	Safety-Limit Minimum Critical Power Ratio
SRP	Standard Review Plan
TER	Technical Evaluation Report
TIP	Traversing Incore Probe
UFSAR	Updated Final Safety Analysis Report

1.0 Introduction

This report is a compendium of Siemens Power Corporation (SPC) methodologies and design criteria which are described in topical reports that the NRC has found acceptable for referencing in boiling water reactor (BWR) licensing applications. This compendium provides a concise, organized source for BWR topical reports. It presents information about the application of each topical report, the associated safety evaluation report (SER) and its conclusions and restrictions for each topical report, the relationships among the topical reports, and, for certain methodologies, descriptions of their unique characteristics or applications.

The methods and topical reports addressed herein are designed to give BWR licensees using SPC 9x9-2, ATRIUM™*-9, and/or ATRIUM-10 fuel the methodologies needed to conform to their original licensing bases and to meet "...cycle-specific parameter limits that have been established using an NRC-approved methodology...," as stated in Generic Letter 88-16. These methodologies may also be used to predict "...changes [to limits]...consistent with all applicable limits of the plant safety analysis that are addressed in the [updated] final safety analysis report ([U]FSAR)." Additionally, these methodologies provide assurance that SPC fuel is compatible with co-resident fuel.

The organization of this report parallels the major sections of the Standard Review Plan[1] (SRP) that apply to reload fuel, specifically, 4.2 Fuel System Design, 4.3 Nuclear Design, 4.4 Thermal and Hydraulic Design of Chapter 4 Reactor, and all appropriate sub-chapters of Chapter 15 Accident Analysis. Table 1.1 includes a list of all the SRP numbers addressed by SPC BWR methodologies. Table 1.2 provides an index to topical reports that may be used to establish operating limits reported in the core operating limits reports (COLR) and that may be referenced in the technical specifications. Table 1.2 notes which topical reports are applicable to specific SPC fuel designs and where in this report each topical report is addressed. Table A-1 found in Attachment A includes a list of all of the methodologies, in the order of their appearance, discussed in this report; shows the major interfaces between and among analyses of each of the SRPs; and lists key parameters of the methodologies which address specific SRP numbers. Table A-1 is not to be considered inclusive of all parameters or methodology interfaces, it is provided for the user of this document as a convenient cross reference to methodologies and their associated SRPs.

There are two styles of citations of references used herein. References to an approved methodology addressed within Section 2.0, 3.0, 4.0, and 5.0 are cited as "Reference section number-number." Other supporting references found in Section 7.0 References are cited by superscript number.

* ATRIUM is a trademark of Siemens.

Table 1.1 SRP No. Addressed by SPC Methodologies

SRP No.	**Chapter 4 Reactor**
4.2	Fuel System Design
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
SRP No.	**Chapter 15 Accident Analysis**
15.1.1 - 15.1.2	Decrease in Feedwater Temperature and Increase in Feedwater Flow
15.2.1 - 15.2.2	Loss of External Load and Turbine Trip
15.2.4	Closure of Main Steam Isolation Valve (BWR)
15.3.3	Reactor Coolant Pump Rotor Seizure
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power
15.4.5	Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
15.4.9	Spectrum of Rod Drop Accidents (BWR)
15.4.9A	Radiological Consequences or Rod Drop Accident (BWR)
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.5	Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary
15.7.4	Radiological Consequences of Fuel Handling Accidents

Table 1.2 Reference Index

Methodology	Fuel Design[a]	Page No.(s)
ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.	9&10	2-11
XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.	9&10	2-12
XN-NF-82-06(P)(A), Supplement 1 Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.	9&10	2-12
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.	9&10	2-13
XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.	9&10	2-13; 5-19
XN-NF-75-32(P)(A), Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)	9&10	2-14
XN-NF-82-06(P)(A), Revision 1 and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.	9&10	2-14
XN-NF-81-51(P)(A), "LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, May 1986.	9&10	2-14
XN-NF-84-97(P)(A), "LOCA – Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, August 1986.	9&10	2-15
XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.	9&10	2-15
EMF-93-177(P)(A) and Supplement 1, "Mechanical Design for BWR Fuel Channels," Siemens Power Corporation, August 1995.	9&10	2-15

Table 1.2 Reference Index

Methodology	Fuel Design[a]	Page No.(s)
ANF-90-82(P)(A) Revision 1 and Revision 1 and Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.	9&10	2-16; 3-6; 4-10
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983.	9&10	3-3; 5-16
XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, November 1990.	9&10	3-4; 5-15
XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.	9&10	3-5
EMF-CC-074(P)(A) Volume 1, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994.	9&10	3-5; 4-4
XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.	9&10	4-4; 5-13
XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983.	9&10	4-5
ANF-1125(P)(A) and Supplements 1 and 2, "ANFB Critical Power Correlation," Advanced Nuclear Fuels Corporation, April 1990.	9	4-5
EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998.	10	4-6
EMF-1997 Supplement 1(P)(A) Revision 0, "ANFB-10 Critical Power Correlation: High Local Peaking Results," July 1998.	10	4-6

Table 1.2 Reference Index

Methodology	Fuel Design[a]	Page No.(s)
EMF-1125(P)(A) Supplement 1 Appendix C, "ANFB Critical Power Correlation Application for Co-Resident Fuel, " Siemens Power Corporation, August 1997.	b.	4-7
XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.	9&10	4-7; 5-13
ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.	9&10	4-8; 5-12
ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.	9&10	4-9; 5-14
ANF-1125(P)(A) Supplement 1 Appendix E, "ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties," Siemens Power Corporation, September 1998.	9	4-9
ANF-1358(P)(A) Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Siemens Power Corporation, September 1992.	9&10	5-15
XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$," Exxon Nuclear Company, May 1986.	9&10	5-17
XN-NF-825(P)(A) Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain," Exxon Nuclear Company, October 1986.	9&10	5-17
XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.	9&10	5-18
ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993.	9&10	5-18
ANF-91-048(P)(A) Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997.	9&10	5-19

Table 1.2 Reference Index

Methodology	Fuel Design[a]	Page No.(s)
XN-CC-33(P)(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975.	9&10	5-19
XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.	9&10	5-20
ANF-CC-33(P)(A) Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option," Advanced Nuclear Fuels Corporation, February 1991.	9&10	5-20
XN-NF-929(P)(A) and Supplements 1 through 4, "Spray Heat Transfer Coefficients for Jet Pump BWR Fuel Assemblies with Water Rods," Advanced Nuclear Fuels Corporation, March 1992.	9	5-20

a. 9 and 10 denotes applicability to SPC 9x9 and 10x10 fuel designs, respectively.

b. Other vendor's fuel

2.0 Fuel System Design

SPC builds fuel assemblies to several specific design criteria to ensure that:

- The fuel assembly shall not fail as a result of normal operation and anticipated operational occurrences. The fuel assembly dimensions shall be designed to remain within operational tolerances and the functional capabilities of the fuel shall be established to either meet or exceed those assumed in the safety analysis.
- Fuel assembly damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.
- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from in-plant handling and shipping.

The first four objectives are those cited in Section I. of 4.2 Fuel System Design of the SRP. The last two objectives were established by SPC to ensure structural integrity of the fuel and the compatibility of the fuel with existing reload fuel. All six of these objectives, which are found in Reference 2-1, are satisfied by SPC design criteria approved by the NRC.

- Preparing controlled documentation of the fuel system description and fuel assembly design drawings.
- Performing analyses with NRC-approved and accepted models and methods for SPC fuels.
- Testing significant new design features with prototype testing and/or lead test assemblies prior to full reload implementation.
- Continued irradiation surveillance programs including post irradiation examinations to confirm fuel assembly performance.
- Using SPC's approved QA procedures, QC inspection program, and design control requirements identified in the EMF-1(A)[2], "Quality Assurance Program for Nuclear Fuels, Services, and Packaging and Transportation of Radioactive Materials."

2.1 *Regulatory Requirements*

SRP Section 4.2 Fuel System Design, establishes criteria to provide assurance that the fuel system is not damaged as a result of normal operations or anticipated operational occurrences, that fuel system damage is never so severe that control rod insertion is prevented when it is required, that the number of fuel rod failures is not underestimated for postulated accidents, and that coolability is always maintained. These design criteria are necessary to meet the requirements of General Design Criteria[3] (GDC) 10, 27, and 35; 10 CFR Part 100,[4] and 10 CFR Part 50[5] (50.46 and Appendix K).

2.2 *Fuel System Design Analyses*

The design criteria used for fuel system design analyses should not be exceeded during normal operations, including anticipated operational occurrences (AOOs). These criteria, described below, address the physical aspects of fuel assemblies and the behavior of the fuel and cladding.

2.2.1 Stress

Design Criteria. The design criteria for evaluating the structural integrity of the fuel assemblies are:

- Fuel assembly handling - The assembly must withstand dynamic axial loads approximately 2.5 times assembly weight.

- For all applied loads for normal operation and anticipated operational occurrences - The fuel assembly component structural design criteria are established for the two primary material categories: austenitic stainless steels (tie plates) and Zircaloy (tie rods, grids, spacer capture rod tubes). The stress categories and strength theory for austenitic stainless steel presented in the ASME Boiler and Pressure Vessel Code, Section III[6] are used as a general guide.

- Steady state stress design limits are given in Table 3.1 of Reference 2-1. Stress nomenclature is per the ASME Boiler and Pressure Vessel Code, Section III.

- Loads during postulated accidents - Deflection or failure of components shall not interfere with reactor shutdown or emergency cooling of the fuel rods.

Bases. In keeping with the GDC 10 specified acceptable fuel design limits (SAFDLs), the fuel damage design criteria for cladding stress assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Conservative stress limits are derived from the ASME Boiler Code, Section III, Article III-2000;[6] and the specified 0.2% offset yield strength and ultimate strength for Zircaloy.

The structural integrity of the fuel assemblies is assured by setting design limits on stresses, deformations, and loadings due to various handling, operational, and accident loads. These limits are applied to the design and evaluation of upper and lower tie plates, grid spacers, tie rods, spacer capture rod, water rods, water channels, fuel assembly cage, and springs where applicable. The allowable component stress limits are based on the ASME Boiler and Pressure Vessel Code, Section III, with some criteria derived from component tests. Cladding stress categories include the primary membrane and bending stresses, and the secondary stresses. The loadings considered are fluid pressure, internal gas pressure, thermal gradients, restrained mechanical bow, flow induced vibration, and spacer contact. Table 3.1 of Reference 2-1 gives the ASME stress level criteria.

The stress calculations use conventional, elasticity theory equations. A general purpose, finite element stress analysis code such as ANSYS[7] may be used to calculate the spacer spring contact stresses. The fuel assembly structural component stresses under faulted conditions are evaluated using primarily the criteria outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III.

The SPC analysis methods for calculating fuel rod cladding and assembly steady-state stresses are discussed and approved in References 2-2 and 2-3.

2.2.2 Strain

Design Criteria. The design criteria for fuel rod cladding strain is that the transient-induced deformations must be less than 1% uniform cladding strain up to 60 MWd/kgU pellet exposure and 0.75% for greater than 60 MWd/kgU.

Bases. The design criteria for cladding strain is intended to preclude excessive cladding deformation and failure from normal operations and AOOs. SPC uses the NRC-approved RODEX2A code (Reference 2-4) to calculate steady-state cladding strain during normal operation. Transient cladding strain is calculated as described in Supplement 1 of Reference 2-5.

2.2.3 Strain Fatigue

Design Criteria. The design criteria for strain fatigue limits the cumulative fatigue usage factor to less than [ ].

Bases. Cycle loading associated with relatively large changes in power can cause cumulative damage which may eventually lead to fatigue failure. Therefore, SPC requires that the cladding not exceed a cumulative fatigue usage factor of [ ]. The fatigue usage factor is the number of expected cycles divided by the number of allowed cycles. The total cladding usage factor is the sum of the individual usage factors for each duty cycle.

The SPC methodology for determining strain fatigue is based on Supplement 1 of Reference 2-5 and the O'Donnell and Langer fatigue design curves.[8] The fatigue curves have been adjusted to incorporate the recommended safety factor of [ ] on stress amplitude or [ ] on number of cycles, whichever is more conservative. The RODEX2 code is used to provide initial steady-state conditions for SPC transient and accident analysis.

2.2.4 Fretting Wear

Design Criteria. The design criteria for fretting wear requires that fuel rod failure due to fretting shall not occur.

Bases. SPC controls fretting wear by use of design features, such as a spacer spring dimple system, which assure that fuel rods are positively supported by the grid spacers throughout the expected irradiation period. Spacer grid spring systems are designed such that the minimum rod contact forces throughout the design life are greater than the maximum fuel rod flow vibration forces. SPC performs fretting tests to verify consistent fretting performance for new spacer designs. Examination of a large number of irradiated BWR rods has substantiated the absence of fretting in SPC designs.

2.2.5 Oxidation and Crud Buildup

Design Criteria. There is [ ] oxide thickness or crud buildup. The effects of oxidation and crud buildup are considered in the thermal and rod internal gas pressure analyses.

Bases. The SPC fuel design basis for cladding corrosion and crud buildup is to prevent 1) significant degradation of the cladding strength, and 2) unacceptable temperature increases. Cladding corrosion reduces cladding wall thickness and results in less cladding load carrying capacity. At normal light water reactor operating conditions, this mechanism is not limiting except under unusual conditions where high cladding temperatures greatly accelerate the corrosion rate. Because of the thermal resistance of corrosion and crud layers, formation of these products on the cladding results in an elevation of temperature within the fuel as well as the cladding.

There is [ ] for crud buildup. However, the BWR fuel performance code RODEX2A (Reference 2-4) includes the crud buildup in the fuel performance predictions. That is, the crud and oxidation models are a part of the approved models and therefore impact the temperature calculation. SPC includes an enhancement in the RODEX2A calculations for the corrosion analysis and fuel temperature analysis. This enhancement is a factor that is input to the code. This factor increases the amount of oxidation predicted by the corrosion model. The factor is selected, based on the particular design power history, to provide an end of life oxidation thickness that is equivalent to the maximum peak oxidation observed for SPC BWR fuel.

SPC data show that even at higher exposures and residence times, cladding oxide thickness is relatively low. Mechanical properties of the cladding are not significantly affected by thin oxide or crud layers. For the thermal analyses, the effect of oxidation is included. There is sufficient conservatism in the gas pressure analysis to account for the effect of cladding oxidation without the use of an additional enhancement factor. For steady-state strain, transient strain and cyclic stress, the effect of wall thinning is [ ] dependent. That is, the change in cladding diameter during a power change is primarily determined by the change in the pellet diameter since pellet-cladding contact occurs at higher exposures. For the cladding end-of-life stress analysis, the wall thickness is reduced consistent with the peak oxide thickness.

2.2.6 Rod Bowing

Design Criteria. The SPC design criteria for rod bowing is [ ] fuel rods shall not be of sufficient magnitude to degrade thermal margins.

Bases. Differential expansion between the fuel rods, and lateral thermal and flux gradients can lead to lateral creep bow of the rods in the spans between spacer grids. This lateral creep bow alters the pitch between rods and may affect the peaking and local heat transfer. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the cladding overheating analysis by limiting fuel rod powers when bowing exceeds a predetermined amount. SPC uses an approved methodology (Reference 2-3) to determine a rod-to-rod clearance closure limit below which a penalty is addressed on the minimum critical power ratio (MCPR) and above which no reduction in MCPR is necessary. The methodology is based on empirical data (Reference 2-6) to calculate minimum [ ] rod spacing. The potential effect of [ ] thermal margin is negligible. Rod bow at extended burnup does not affect thermal margins due to the lower powers achieved at high exposure.

2.2.7 Axial Growth

Design Criteria. SPC requires that the fuel assembly be compatible with the channel throughout the fuel assembly lifetime. In addition, SPC requires that clearances and engagements in the fuel assembly structure be maintained throughout the lifetime of the fuel.

Bases. SPC evaluates fuel channel-fuel assembly differential growth to assure that the fuel channel to lower tie plate engagement is maintained to the design burnup. Another condition for BWR fuel assemblies is to maintain engagement between the fuel rod end cap shank and the assembly tie plates, i.e., to prevent fuel rod disengagement from the tie plates. The change in BWR fuel rod-to-tie plate engagement (and possible disengagement) is due to the differential growth rate between the fuel rods and the tie rods for 9x9 fuel designs. For the 10x10 fuel, [ ], the goal is to ensure adequate clearance for growth of the fuel rods.

The analysis method (Reference 2-3) for evaluating rod-to-tie plate engagement is based on a statistical upper bound of measured differential rod-to-tie plate growth data (Reference 2-4) for 9x9 and 10x10 designs. The correlation predicts differential growth that bounds [ ] confidence level. This analysis uses [ ] in order to maintain conservatism in the calculated initial engagements and clearances.

2.2.8 Rod Internal Pressure

Design Criteria. SPC limits maximum fuel rod internal pressure relative to system pressure. In addition, SPC requires that when fuel rod pressure exceeds system pressure, the pellet-clad gap has to [ ] power operations.

Bases. Rod internal pressure is limited to prevent unstable thermal behavior and to maintain the integrity of the cladding. Outward circumferential creep which may cause an increase in pellet-to-cladding gap must be prevented since it would lead to higher fuel temperature and higher fission gas release. The maximum internal pressure is also limited to protect against embrittlement of the cladding caused by hydride reorientation during cooldown and depressurization conditions. A proprietary limit above system pressure has been justified by SPC in Reference 2-7.

2.2.9 Assembly Liftoff

Design Criteria. SPC requires that the assembly not levitate from hydraulic or accident loads.

Bases. Levitation of a fuel assembly could result in the assembly becoming disengaged from the fuel support and interfering with control rod movement. For normal operation, including AOOs, the submerged fuel assembly weight, including the channel, must be greater than the hydraulic loads. The criteria is applicable to both cold and hot conditions and uses the technical specification limits on total core flow. For accident conditions, the normal hydraulic loads plus additional accident loads shall not cause the assembly to become disengaged from the fuel support. This assures that control blade insertion is not impaired.

2.2.10 Fuel Assembly Handling

Design Criteria. The assembly design must withstand all normal axial loads from shipping and fuel handling operations without permanent deformation.

Bases. SPC uses either a stress analysis or testing to demonstrate compliance. The analysis or test uses an axial load of [ ] times the static fuel assembly weight. At this load, the fuel assembly structural components must not show any yielding. Because of design features, for example grooved end caps, failure from axial loads will occur at the tie rod end caps rather than in the cladding or tie plates.

The rod plenum spring also has design criteria associated with handling requirements. The spring must maintain a force against the stack weight to prevent column movement during handling. The component drawing specifies the fabricated cold spring force.

2.2.11 Miscellaneous Component Criteria

2.2.11.1 Compression Spring Forces

Design Criteria. The compression spring(s) must support the weight of the upper tie plate and the channel throughout the design life of the fuel. Therefore, there is a requirement on the minimum compression spring force. There is also a maximum spring force limit requirement that the force be less than the calculated fuel rod [ ] designs.

Bases. The compression springs aid in seating the fuel rods against the lower tie plate while allowing for non-uniform growth and expansion of the same. The compression springs also exert an upward load to maintain the upper tie plate against the latching mechanism. The design criteria for the minimum force ensures the upper tie plate is fully latched throughout the lifetime of the fuel. A maximum force limit for the compression spring ensures fuel rods are not inadvertently damaged during tie plate removal and installation. The maximum force requirements do not apply to the [ ] on the water channel.

2.2.11.2 Lower Tie Plate Seal Spring

Design Criteria. The seal accommodates the channel deformation to limit the leak rate of coolant between the lower tie plate and channel wall.

Bases. The lower tie plate seal spring limits the leak rate of coolant between the lower tie plate and the channel wall. The seal shall have adequate corrosion resistance and be able to withstand the operating stresses without yielding. The design also considers the differential axial growth between the channel and the bundle. Flow testing of prototypic components verifies the leakage rate and fretting resistance. A stress analysis provides the seal stresses.

2.2.12 Fuel Rod Failure

The fuel rod failure design criteria and bases cover normal operation conditions, including AOOs, and postulated accidents. When the fuel rod failure criteria are applied in normal operation, including AOOs, they are used as limits (Specified Acceptable Fuel Design Limits) since fuel failure under those conditions must not occur according to GDC 10.[3] When the criteria are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100.[4]

2.2.12.1 Internal Hydriding

Design Criteria. SPC limits internal hydriding by imposing a fabrication limit for total hydrogen in the fuel pellets of less than 2.0 ppm.

Bases. The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. Hydriding, as a cladding failure mechanism, is precluded by controlling the level of moisture and other hydrogenous impurities during fuel pellet fabrication. The hydrogen concentration criteria is met by maintaining moisture control during fuel fabrication (Reference 2-7).

2.2.12.2 Cladding Collapse

Design Criteria. Creep collapse of the cladding is avoided in the SPC fuel system design by eliminating the formation of significant axial gaps in the pellet column.

Bases. If axial gaps in the fuel pellet column were to occur due to handling, shipping, or fuel densification, the cladding would have the potential of collapsing into the gap. Because of the large local strains that would result from the collapse, the cladding is assumed to fail. Creep collapse of the cladding and the subsequent potential for fuel failure is avoided in the SPC fuel system design by eliminating the formation of significant axial gaps. The maximum cladding circumferential creep and ovalization consistent with the [ ] is computed by a creep collapse evaluation to demonstrate that no axial gaps are present. The evaluation must show that the pellet column is compact at the burnup of maximum densification [ ] The internal plenum spring provides an axial load on the fuel stack that is sufficient to assist in the closure of any gaps caused by handling, shipping, and densification. Evaluation of cladding creep stability in the unsupported condition is performed considering the compressive load on the cladding due to the difference between primary system pressure and the fuel rod internal pressure. SPC fuel is designed to minimize the potential for the formation of axial gaps in the fuel and to minimize clad creepdown which would prevent the closure of axial gaps or allow creep collapse.

The RODEX2A code (Reference 2-4) is used to provide initial in-reactor fuel rod conditions to the COLAPX[9] method described in Reference 2-7 which is used to predict creep collapse, e.g., radial fuel-cladding gap size, fill gas pressure, and cladding temperatures. COLAPX calculates ovality changes and creep deformation of the cladding as a function of time.

2.2.12.3 Overheating of Cladding

Design Criteria. The design basis to preclude fuel rod cladding overheating is 99.9% of the fuel rods shall not experience transition boiling.

Bases. It has been traditional practice to assume that fuel failures will occur if the thermal margin criteria is violated. Thermal margin is stated in terms of the minimum value of the critical power ratio (CPR) for the most limiting fuel assembly in the core. Prevention of potential fuel failure from overheating of the cladding is accomplished by minimizing the probability of exceeding thermal margin limits on limiting fuel rods during normal operation and anticipated operational occurrences. Compliance with this criterion as part of the reload thermal hydraulics analysis is discussed in Section 4.2 of this report.

2.2.12.4 Overheating of Fuel Pellets

Design Criteria. Fuel failure from overheating of the fuel pellets is not allowed. The centerline temperature of the fuel pellets must remain below melting during normal operations and AOOs.

Bases. Steady state and transient design linear heat generation rate (LHGR) limits are established for each fuel system to protect against centerline melting. Operation within these LHGR limits prevents centerline melting during normal operation and anticipated operational occurrences throughout the design lifetime of the fuel.

A correlation is used for the fuel melting point that accounts for the effect of burnup and gadolinia content. This fuel melting limit has been reviewed and approved (Reference 2-7) with respect to the extended burnup of fuel and gadolinia bearing fuel.

SPC uses the RODEX2A computer code (Reference 2-4) to calculate the maximum possible fuel centerline temperature for normal operations. Conservative LHGR power histories are used to perform the centerline temperature calculations. For AOOs, SPC uses the RODEX2A code to calculate maximum possible fuel centerline temperatures at LHGRs which are higher than the steady-state LHGR history used for normal operations.

2.2.12.5 Pellet/Cladding Interaction

Design Criteria. The Standard Review Plan[1] does not contain an explicit criterion for pellet/cladding interaction. However, it does present two related criteria. The first is that transient-induced deformations must be less than 1% uniform cladding strain. The second is that fuel melting cannot occur.

Bases. The cladding strain requirement is addressed in Section 2.2.2. The centerline temperature requirement is addressed in Section 2.2.12.4.

2.2.12.6 Cladding Rupture

Design Criteria. 10 CFR 50 Appendix K[5] requires that cladding rupture must not be underestimated when analyzing a loss of coolant accident.

Bases. Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure conditions during a LOCA. Since there are no specific design criteria in the Standard Review Plan[1] associated with cladding rupture, SPC has established a rupture temperature correlation to be used during the LOCA emergency core cooling system (ECCS) analysis.

The effects of cladding rupture are an integral part of the SPC ECCS evaluation model (See Reference 5-13). The cladding ballooning and rupture models used are those presented in NUREG-0630[10] for cladding rupture evaluation. These models are described in XN-NF-82-07(P)(A) Revision 1 (See Reference 5-15).

2.2.12.7 Fuel Rod Mechanical Fracture

Design Criteria. SPC limits the combined stresses from postulated accidents to the stresses given in the ASME Code, Section III, Appendix F[6] for faulted conditions.

Bases. A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force, such as a hydraulic load or a load derived from core plate motion induced by a seismic or LOCA event. The design bases and criteria for mechanical fracturing of SPC BWR reload fuel are presented in Reference 2-8, which describes SPC's LOCA-seismic structural response analysis. The application of the LOCA-seismic structural response analysis to SPC's 9x9 fuel designs is presented in Reference 2-9. The design basis is that the [ ] the fuel rod cladding. The stresses, due to postulated accidents in combination with normal steady-state fuel rod stresses, should not exceed the stress limits given in References 2-8 and 2-9. The allowable stresses are derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix F, for faulted conditions.

The mechanical fracture analysis is done as part of the plant specific seismic-LOCA loading analysis, [ ] For BWR fuel, the response is dominated by the channel. If the channel design is unchanged and the assembly behaviors are similar, [ ] SPC verifies the assembly characteristics for new designs to ascertain that these characteristics (assembly weight and vibration mode) are similar to the co-resident fuel.

2.2.12.8 Fuel Densification and Swelling

Design Criteria. Fuel densification and swelling are limited by the design criteria specified for fuel temperature, cladding strain, cladding collapse, and internal pressure criteria.

Bases. SPC uses the NRC reviewed and accepted densification and swelling models in the fuel performance codes, RODEX2 (Reference 2-5) and RODEX2A (Reference 2-4).

2.2.13 BWR Fuel Coolability

For accidents in which severe fuel damage might occur, core coolability and the capability to insert control blades are essential. Normal operation or anticipated operational occurrences must remain within the thermal margin criteria. Chapter 4.2 of the Standard Review Plan[1] provides several specific areas important to the coolability and the capability of control blade insertion. The sections below discuss these areas.

2.2.13.1 Fragmentation of Embrittled Cladding

Design Criteria. ECCS evaluations meet the 10 CFR 50.46[5] limits of 2200 °F peak cladding temperature, local and core-wide oxidation, and long term coolability.

Bases. The requirements on cladding embrittlement relate to the LOCA requirements of 10 CFR 50.46. The principal cause of cladding embrittlement is the high cladding temperatures that result in severe cladding oxidation.

The models to compute the temperatures and oxidation are those prescribed by Appendix K of 10 CFR 50[5] (See Reference 5-14). LOCA analyses are performed on a plant specific basis (See References 5-10 and 5-11).

2.2.13.2 Violent Expulsion of Fuel

Design Criteria. SPC limits the radially-averaged enthalpy deposition at the hottest axial location to 280 cal/gm for severe reactivity initiated accidents.

Bases. In a severe reactivity initiated accident (RIA), large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of the fuel. The SPC methodology complies with the fission product source term guideline in Regulatory Guide 1.77[11] and the Standard Review Plan[1] that restricts the radially-averaged energy deposition.

The limiting RIA for SPC fuel in a BWR is the control rod drop accident (CRDA). SPC calculates the maximum radially averaged enthalpy for the CRDA for each reload core in order to assure that the maximum calculated enthalpy is below the 280 cal/gm limit. The control rod drop calculation methodology approved by the NRC is described in Reference 5-7. The parameterized SPC control rod drop methodology determines maximum deposited enthalpy as a function of dropped rod worth, effective delayed neutron fraction, Doppler coefficient, and four-bundle local peaking factor.

The CRDA analysis is not part of the normal fuel assembly mechanical analysis but is part of the cycle specific safety analysis performed for each BWR.

2.2.13.3 Cladding Ballooning

Design Criteria. There are no specific design limits associated with cladding ballooning, other than a requirement in Appendix K to 10 CFR 50[5] that the degree of swelling not be underestimated.

Bases. Zircaloy cladding will balloon (swell) under certain combinations of temperature, heat rate, and stress during a LOCA. Cladding ballooning can result in flow blockage; therefore, the LOCA analysis must consider the cladding ballooning impacts on the flow.

The effects of cladding ballooning are an integral part of the SPC ECCS evaluation model (See Reference 5-13). The cladding ballooning and rupture models used are those presented in

NUREG-0630[10] for cladding rupture evaluation. These models are described in XN-NF-82-07(P)(A) Revision 1 (See Reference 5-15). These models have been approved by the NRC for extended burnup levels.[15]

The RODEX2 fuel performance code (Reference 2-5) is used to provide burnup dependent input to the LOCA analysis, e.g., stored energy and rod pressures, that are a function of the initial steady-state operation of the fuel. This initial steady-state fuel condition is also important to cladding ballooning.

2.2.13.4 Fuel Assembly Structural Damage from External Forces

Design Criteria. The SPC design criteria for fuel assembly structural damage from external forces are discussed in Sections 2.2.1-Stress, 2.2.9-Assembly Liftoff, and 2.2.12.7-Fuel Rod Mechanical Fracture.

Bases. Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. The Standard Review Plan[1] states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control blade insertion when required during these accidents. The SPC design basis is that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case accident and that system damage is never so severe as to prevent control blade insertion. SPC ensures these design bases are met by placing ASME design limits on the stresses that critical fuel assembly components can experience. These limits have been approved for SPC fuel assemblies in References 2-8 and 2-9.

2.3 *NRC-Accepted Topical Report References*

The NRC has approved the following licensing topical reports that describe the methods and assumptions used by SPC to demonstrate the adequacy of its BWR fuel system design. These reports address mechanical design criteria and required mechanical and thermal conditions. The purpose of each topical report and the restrictions that have been placed on the methods presented are described below.

2-1: ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

- Purpose: Establish a set of design criteria which assures that BWR fuel will perform satisfactorily throughout its lifetime.

- SER Conclusions/Restrictions: Burnup shall not be increased beyond 60,000 MWd/MTU unless axial growth and fretting wear data have been collected from lead test assemblies of the modified design. {Note: The NRC approved higher burnup values as presented in EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A) described below.}

  Exposure beyond 60,000 MWd/MTU peak pellet must be approved by the NRC. {Note: The exposure limit was extended to a rod-average burnup of 62 GWd/MTU by the approval of EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A).}

  Approval does not extend to the development of additive constants for ANFB to co-resident fuel.

For each application of the mechanical design criteria, SPC must document the design evaluation and submit a summary of the evaluation to the NRC.

- Observations: The application of the processes and criteria described in this topical report do not require prior NRC approval. These processes and criteria are those that have been evolving for many years for different SPC fuel designs. [12-16]

  The mechanical design of BWR assembly fuel channel is performed using the criteria and methods described and approved in Reference 2-11.

  The design methodology for the reconstitution of a BWR fuel assembly complies with Reference 2-12.

2-2: XN-NF-85-67(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.

- Purpose: Demonstrate that mechanical design criteria are not violated when fuel is operated at the LHGR limits for both 8x8 fuel and 9x9 fuel with maximum assembly discharge exposures of 35,000 MWd/MTU and 40,000MWd/MTU, respectively.

- SER Conclusions/Restrictions:

  1. LHGR limit curves (Fig. 3.1, 3.2, and 3.3) are to be used for the fuel described.
  2. Discharge exposure is limited to previously approved 30,000 MWd/MTU batch average exposure pending approval of Reference 2-3.
  3. Additional rod bow data are required for burnup extensions beyond 30,000 MWd/MTU for 8x8 fuel and 23,000 MWd/MTU for 9x9 fuel.

- Observations: This topical report includes an initial description of the process used to develop linear heat generation rates for fuel designs. (Note: This process is currently still in use.)

2-3: XN-NF-82-06(P)(A) Supplement 1 Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.

- Purpose: Provide the design bases, analyses, and test results in support of the qualification of BWR fuel (9x9) for burnup extension to 40,000 MWd/MTU peak assembly exposure and to obtain approval of the rod bow method for extended burnup.

- SER Conclusions/Restrictions: The LHGR limit curves (Figures 3.1, 3.2, and 3.3) in XN-NF-85-67(P)(A) Revision 1 continue to be applicable as bounding LHGR limits.

- Observations: The rod bow model approved in XN-NF-75-32(P)(A) was approved for application to 9x9 fuel for assembly exposures to 40,000 MWd/MTU. The extended burnup data to confirm the rod bow model indicated that rod bow at extended burnup does not affect thermal margins due to the lower rod powers at high exposure. The rod bow model is currently used for ATRIUM-9 and ATRIUM-10 fuel designs.

2-4: EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.

- Purpose: Extend the exposure limits of the RODEX2A[17] code, which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs.
- SER Conclusions/Restrictions: RODEX2A is acceptable for steady state licensing applications to 62,000 MWd/MTU rod-average burnup and the fuel rod growth, fuel assembly growth, and fuel channel growth models and analytical methods are acceptable for ATRIUM-9 and -10 fuel designs up to 54,000 MWd/MTU assembly-average burnup.
- Observations: The RODEX2A code, which is used for BWR fuel design applications, is a derivative of SPC's base fuel performance code RODEX2.

  In the approved topical report the NRC acknowledges the following observations as correct:

  - Steady state analyses of maximum wall thinning from oxidation for end of life conditions will be performed.
  - The growth correlations reviewed are applicable to all SPC 9x9 fuel designs.
  - Transient strain is to be calculated with the version of RODEX referenced in XN-NF-81-58(P)(A) Revision 2 Supplement 1 (Reference 2-5) and strain is limited to 1.0% up to 60 MWd/kgU and 0.75% for greater than 60 MWd/kgU.
  - Steady state strain is to be calculated with RODEX2A and is limited to 1%.
  - RODEX2A is to be used to calculate fuel temperatures for fuel melt analyses.
  - RODEX2 shall be used as the base fuel performance code to interface with the SPC LOCA and transient thermal-hydraulic methodologies.
  - RODEX2A can be used to model fuel with up to 8% gadolinia loading (See Reference 2-10).

2-5: XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.

- Purpose: Provide an analytical capability to predict BWR and PWR fuel thermal and mechanical conditions for normal core operation and to establish initial conditions for power ramping, non-LOCA and LOCA analyses.
- SER Conclusions/Restrictions: A physically-based gas release model shall be used.
- Observations: RODEX2 is the fuel performance code that provides input to BWR LOCA and transient thermal-hydraulic methodologies.

  RODEX2 was approved for use up to an exposure limit of 62,000 MWd/MTU rod-average burnup for PWR applications (ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," June 1990).

  RODEX2 may be used to model fuel with up to 8% gadolinia loading (See Reference 2-10).

2-6: XN-NF-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)

- Purpose: Develop an empirical method for determining fuel rod bow.

- SER Conclusions/Restrictions: The technical evaluation of the methodology was limited to the fuel designs, exposures, and conditions stated in the topical report and, in part, on assumptions made in formulating the methodology. It was recommended that Exxon [SPC] continue fuel surveillance to ensure confidence in the assumptions and bases.

- Observations: SPC has continued to gather data from fuel surveillance and CPR experiments.

2-7: XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.

- Purpose: Provide the design bases, analyses and test results in support of the qualification of BWR fuel (8x8 and 9x9) for burnup extension to 35,000 MWd/MTU assembly batch exposure. (Note: This topical report also addressed burnup extension to 45,000 MWd/MTU for PWR fuel.)

- SER Conclusions/Restrictions: If fuel at extended burnup levels experiences a plant depressurization accident, the licensee must address possible cladding hydride reorientation prior to further irradiation of the fuel.

- Observations: The TER for ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995 references this topical report as the approved method for setting a fuel pressure limit above system pressure and a criterion which requires that a radial fuel-cladding gap be maintained during constant and increasing power operation under normal reactor operating conditions.

2-8: XN-NF-81-51(P)(A), "LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, May 1986.

- Purpose: Develop a methodology for performing LOCA-Seismic structural analyses of BWR jet pump assemblies.

- SER Conclusions/Restrictions: The allowable stress values reported for BWR jet pump fuel channel and assembly components are acceptable and licensees referencing the topical report for other non-GE manufactured channels are required to show that the calculated allowable stresses for seismic and LOCA loading conditions are bounded by those in the topical report.

- Observations: The analyses reported were for an 8x8 fuel assembly. The channeled fuel assembly seismic analysis was performed using the response spectrum method of dynamic analysis in the NASTRAN finite element program. [18]

2-9: XN-NF-84-97(P)(A), "LOCA - Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, August 1986.

- Purpose: Demonstrate the adequacy of the methodology approved in XN-NF-81-51(P)(A), Reference 2-8 above, to perform LOCA-Seismic calculations for 9X9 fuel.

- SER Conclusions/Restrictions: "We have completed our review of the structural response of the ENC 9x9 JP-BWR reload fuel design to LOCA-seismic loading as described in the topical report XN-NF-84-97(P). We conclude that this topical report is acceptable for referencing in licensing applications. We note, however, that the analyses in XN-NF-84-97(P) were performed for the Dresden 3 reactor. For other reactors, conformance to the acceptance criteria or the SRP Section 4.2, Appendix A, can be demonstrated by referencing XN-NF-84-97(P) and submitting justification that the analyses presented in the topical report bounds the particular application under review."

- Observations: The channeled fuel assembly seismic analysis was performed using the response spectrum method of dynamic analysis in the NASTRAN finite element program[18] noted above in XN-NF-81-51(P)(A).

  First time applications of the methodology described in both XN-NF-81-51(P)(A) and XN-NF-84-97(P)(A) to other reactors and fuel designs not included in those reports are to be justified and submitted to the NRC, as required by the SER restriction above.

  As described in Section 2.2.12.7, the use of existing LOCA-seismic event fuel assembly loading analyses may be justified provided the [ ] for the new fuel design. For LOCA–seismic analyses performed to the generic design criteria (Reference 2-1), prior NRC approval is not required.

2-10: XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.

- Purpose: Justify gadolinia fuel properties for up to 8 wt % gadolinia loading in uranium dioxide fuel to be used in BWR fuel designs.

- SER Conclusions/Restrictions: Based on a commitment to confirm the fission gas release model with in-reactor data, the gadolinia fuel properties are acceptable for licensing applications up to 8 wt% gadolinia concentration.

- Observations: In-reactor fission gas release test results[19] were provided to the NRC. The thermal conductivity model supercedes the previously approved model[19].

2-11: EMF-93-177(P)(A) and Supplement 1, "Mechanical Design for BWR Fuel Channels," Siemens Power Corporation, August 1995.

- Purpose: Demonstrate that analytical methods are adequate to perform evaluations which ensure that fuel channels perform as designed for normal operations and during anticipated operational occurrences and that for postulated loadings channel damage does not prevent control blade insertion and assembly coolability is maintained.

- SER Conclusions/Restrictions: Subject to certain conditions, the analyses conducted by SPC are acceptable for licensing applications. For specific plant applications the following conditions are to be met:

  - The [reported] allowable differential pressure loads and accident loads should bound those of the specific plant.

  - Lattice dimensions should be compatible to those used in the analyses reported such that the minimum clearances with control blades continue to be acceptable.

  - Maximum equivalent exposure and residence time should not exceed the values used in the analyses.

- Observations: The methodology approved is appropriate for exposures and minor dimensional changes beyond those evaluated and reported in the topical. Use of the methodology to extended exposure must be validated against the original design criteria.

2-12: ANF-90-82(P)(A) Revision 1 and Revision 1 and Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.

- Purpose: Develop a methodology to justify reinsertion of irradiated fuel assemblies, which have been reconstituted with replacement rods, into a reactor core. Replacement rods can be fuel rods containing natural uranium pellets, water rods, and inert rods containing zircaloy or stainless steel inserts.

- SER Conclusions/Restrictions: The reconstitution methodology is acceptable for reload licensing applications with the following conditions:

  - BWR reconstituted assemblies are limited to 9 rods per assembly.

  - PWR reconstituted assemblies are limited to two inert rods or one inert rod and a guide tube per subchannel, with a total of no more than 26 percent inert replacement rods per assembly.

  - Reconstituted assemblies located at the outer edge of the core are limited to 26 percent inert rods per assembly, but are not limited to two inert rods or one inert rod and a guide tube per subchannel.

- Observations: The reconstitution methodology is applicable to all fuel designs.

3.0 Nuclear Design

Nuclear design analyses are used for nuclear fuel assembly design and core design. The core design analysis demonstrates operating margins for minimum critical power ratio (MCPR), maximum average planar linear heat generation rate (MAPLHGR), and linear heat generation rate (LHGR). Two LHGR limits are established for each fuel design. One is a steady state operating fuel design limit (FDL), and the other provides protection against the power transient (PAPT) limit.

An exposure dependent LHGR limit is established for each fuel assembly design as part of the mechanical design analysis. The LHGR limit bounds the power history established to perform the mechanical analyses. Hence, operation of the fuel assembly within the steady state LHGR limit ensures that the power history assumption used in the mechanical design analyses remains valid.

3.1 *Regulatory Requirements*

SRP Section 4.3 Nuclear Design discusses GDC 10-13, 20, and 25-28 which pertain to nuclear design. Many of the GDCs relate to mechanical properties of the fuel assembly that are to be satisfied by meeting appropriate thermal and reactivity margin limits during the residence of the fuel in the reactor core. SPC standard design practice is to define these limits and demonstrate that the fuel maintains appropriate margin to these limits by calculating the expected margins in simulated projections of the cycle prior to the fuel being loaded in the reactor core. In addition, by demonstrating that appropriate licensing criteria are met when certain postulated accidents are modeled to occur during the cycle in which the fuel is loaded, the safety aspects of the fuel are assured.

Of the GDCs mentioned in 4.3 Nuclear Design, only GDC 11 is principally related to the neutronic response of the fuel. GDC 11 requires that "in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity."

3.2 *Nuclear Design Analyses*

The nuclear design analyses demonstrate operating margin to design limits, including MCPR, MAPLHGR, and LHGR. The approved nuclear design codes and methodologies are described in References 3-1, 3-2, and 3-3.

3.2.1 Fuel Rod Power History

Design Criteria. The nuclear design analysis must be consistent with the exposure dependent LHGR limit established during the mechanical design analysis for each fuel assembly design.

Two LHGR limits are established for each fuel design. One is a steady state limit, the other a PAPT limit. Both limits are a function of fuel planar burnup. The transient LHGR design limit satisfies the strain and fuel overheating design criteria discussed in Sections 2.2.2-Strain and 2.2.12.4-Overheating of Fuel Pellets. The design margin between the steady state and transient LHGR limits is sufficient to account for increases in the LHGR during transients.

Bases. An exposure dependent LHGR limit is established for each fuel assembly design as part of the mechanical design analysis (Reference 2-2 and 2-3). The LHGR limit bounds the power history established to perform the mechanical analyses. Therefore, operation of the fuel assembly within the LHGR limit is necessary to ensure that the power history assumption used in the mechanical design analyses remains valid. The specific mechanical design criteria are provided in Reference 2-1.

Exposure dependent LHGR limits are established for each fuel design. These limits are established based on nuclear design analyses and the fuel design criteria established in Reference 2-1. Specifically, conservative power histories are generated based on proposed LHGR limits. This is accomplished by assuming an average axial peaking factor [ ] with extreme values [ ] during approximately equal exposure intervals of [ ] GWd/MTU. This assumption leads to conservative estimates of fission gas release. Since the maximum fuel rod nodal power is the LHGR limit, the low axial peaking factor means that the average fuel rod power is relatively high, with high fission gas release along the fuel length. When the rod is assumed to be at the maximum LHGR, a high value of the axial peaking factor means a lower average fuel rod power, which is not conservative in the fission gas release calculation.

3.2.2 Kinetics Parameters

Design Criteria. The design criteria for the core reactivity coefficients are as follows:

- Void reactivity coefficient due to boiling in the active channel shall be negative;
- Doppler coefficient shall be negative at all operating conditions;
- Power coefficient shall be negative at all operating conditions.

Bases. Fuel assembly designs in which less moderation and/or higher temperatures reduce the core reactivity will therefore act as an automatic shutdown mechanism. Thus, prompt reactivity insertion events such as the control rod drop accident have an inherent shutdown mechanism. SPC calculates the reactivity coefficients on a plant and cycle specific basis through application of the standard neutronics design and analysis methodology (References 3-1 and 3-2).

3.2.3 Stability

Design Criteria. New fuel designs and new fuel design features must be stable (core decay ratio <1.0) and should exhibit channel decay ratio characteristics equivalent to existing SPC fuel designs.

Bases. Determination of the effect of all fuel designs and design features on core stability is made on a cycle-specific basis. Associated with these calculations is confirmation of existing power flow range exclusion regions or redefinition of the regions, as necessary.

SPC uses the NRC-approved STAIF code (Reference 3-4) for stability evaluations. STAIF is a frequency domain code that simulates the dynamics of a BWR. SPC confirms that the stability performance of a new BWR fuel design is equivalent to or better than that of an approved SPC fuel design. As the stability performance of the fuel is dominated by the power distribution, and to a lesser extent by fuel design variations, it is necessary to establish consistent boundary conditions when comparing different variations in the fuel design. Therefore, SPC uses the same power distribution and full core equilibrium cycle loading pattern for these fuel design comparisons. The neutronic characteristics will remain unchanged with the exception of the void coefficient, which is directly dependent on the fuel design.

3.2.4 Core Reactivity Control

Design Criteria. The design of the assembly shall be such that the technical specification shutdown margin will be maintained. Specifically, the assemblies and the core must be designed to remain subcritical by the technical specification margin with the highest reactivity worth control rod fully withdrawn and the remaining control rods fully inserted. Calculated shutdown margin is verified using startup critical data. At a minimum, this verification is performed at beginning-of-cycle (BOC) for each reactor.

Bases. Shutdown margin is calculated on a cycle specific basis using NRC-approved methodology (References 3-1 and 3-2). If necessary, shutdown margin is calculated at several cycle exposure points in order to determine the minimum shutdown margin for a cycle. The calculated shutdown margin is reported on a plant and cycle specific basis as required in Reference 3-3. SPC also confirms the worth of the standby liquid control system on a cycle specific basis using the technical specification values of boron concentration.

3.3 *NRC-Accepted Topical Report References*

The NRC has approved the following licensing topical reports that describe the methods and assumptions used by SPC to demonstrate the adequacy of its fuel system nuclear design. These reports address nuclear design criteria and required fuel and thermal conditions used in licensing analyses. The purpose of each topical report and restrictions on the methods presented are described below.

3-1: XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983.

- Purpose: Development of BWR core analysis methodology which comprises codes for fuel neutronic parameters and assembly burnup calculations, reactor core simulation, diffusion theory calculations, core and channel hydrodynamic stability predictions, and producing input for nuclear plant transient analysis. Procedures for applying the codes for control rod drop, control rod withdrawal and fuel misloading events have been established.

- SER Conclusions/Restrictions: No specific restrictions were given in the SER, but a recommendation was "...made that the analytical models be continuously verified to insure their applicability."

- Observations: Portions of this topical report have been superceded by subsequently approved codes or methodologies. Superceded and currently applicable portions are identified below:

  Superceded Portions:

  Fuel Assembly Depletion Model - XFYRE replaced with CASMO-3G (See Reference 3-2.)

  Core Simulator - XTGBWR replaced with MICROBURN-B (See Reference 3-2.)

Diffusion Theory Model - XDT replaced with CASMO-3G (See Reference 3-2.)

Stability Analysis - COTRAN replaced with STAIF (See Reference 3-4.)

Control Rod Withdrawal - XTGBWR replaced with MICROBURN-B (See Reference 3-2.)

Fuel Misloading Analysis – XFYRE replaced with CASMO-3G and XTGBWR replaced with MICROBURN-B. These analyses are now performed to verify that the offsite dose due to such events does not exceed a small fraction of 10 CFR 100 guidelines as described and approved in Reference 3-3.

Applicable Portions:

Control Rod Drop Accident - This analysis is performed using COTRAN.

Control Rod Withdrawal – This analysis is the same as that used to determine the change in CPR (ΔCPR) for error rod patterns, but with an additional procedure. The additional procedure evaluates the number of fuel rods in boiling transition (BT) to determine that a specific error rod pattern does not challenge the criterion that < 0.1% rods are in BT at the MCPR safety limit, assuming failure of the Rod Block.

Neutronic Reactivity Parameters - These parameters are determined as described in the topical report but using the most recently approved codes.

Void Reactivity Coefficient - Method used to calculate core reactivity coefficient is the same but MICROBURN-B is used instead of XTGBWR.

Doppler Reactivity Coefficient - Method used to calculate the core average Doppler coefficient is the same but CASMO-3G is used instead of XFYRE.

Scram Reactivity - Method used is the same except MICROBURN-B is used instead of XTGBWR.

Delayed Neutron Fraction - Calculated using CASMO-3G instead of XFYRE.

Prompt Neutron Lifetime - Calculated using CASMO-3G instead of XFYRE.

3-2: XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, November 1990.

- Purpose: Replace the XFYRE bundle depletion and XTGBWR simulator codes with the updated codes MICBURN-3/CASMO-3G and MICROBURN-B, respectively.

- SER Conclusions/Restrictions:
  1. The currently approved traversing incore probe (TIP) asymmetry uncertainty value of 6.0 percent (See Reference 3-1) should be used in determining the radial bundle power uncertainty.
  2. The application of CASMO-3G/MICROBURN-B to fuel designs that differ significantly from those included in the [report] data base should be supported by additional code validation to ensure that the methodology and uncertainties are applicable.

- Observations: CASMO-3G and MICROBURN-B were incorporated into the methodologies described in Reference 3-1.

  Application to fuel designs that differ significantly from those in the Supplement 3 data base to be supported by additional code validation to ensure uncertainties remain applicable. (This is addressed generically when new design types are introduced - e.g., 11x11 fuel, etc.)

3-3: XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

- Purpose: Summarize the types of BWR licensing analyses performed, identify each with approved computer codes and methodologies, and develop a reload reporting format.

- SER Conclusions/Restrictions: Conditions imposed were based on pending approvals of outstanding topical reports which have been subsequently approved. Currently, this topical report is applicable to fuel loading error analyses, combining safety limits in conjunction with procedural controls based on POWERPLEX®* monitoring, and the reload reporting format.

- Observations: This report may be used contextually only since many of the codes and methodologies referenced have changed or have been replaced since the report was approved.

  This topical report included a modified procedure for performing the consequence analysis for fuel misloading as mentioned in Observations for Reference 3-1. [

  ]

3-4: EMF-CC-074(P)(A) Volume 1, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994.

- Purpose: Provide a methodology for the determination of the thermal-hydraulic stability of BWRs, including reactivity feedback effects.

* POWERPLEX is a trademark of Siemens registered in the United States and various other countries.

- SER Conclusions/Restrictions:

  1. The core model must be divided into a minimum of 24 axial nodes.

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

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

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

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

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

  4. The collapsed 1-D cross sections must represent the actual conditions being analyzed as closely as possible, including control rod positions.

  5. The STAIF calculation must use the "shifted Nyquist" or complex pole search feature to minimize the error at low decay ratio conditions.

- Observations: Stability analyses procedures described in XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983 (See Section 3.2, Reference 3-1) were superceded by the approval of the STAIF code.

3-5: ANF-90-82(P)(A) Revision 1 and Revision 1 and Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.

- Purpose: Develop a methodology to justify reinsertion of irradiated fuel assemblies, which have been reconstituted with replacement rods, into a reactor core. Replacement rods can be fuel rods containing natural uranium pellets, water rods, and inert rods containing zircaloy or stainless steel inserts.

- SER Conclusions/Restrictions: The reconstitution methodology is acceptable for reload licensing applications with the following conditions:

  - BWR reconstituted assemblies are limited to 9 rods per assembly.

  - PWR reconstituted assemblies are limited to two inert rods or one inert rod and a guide tube per subchannel, with a total of no more than 26 percent inert replacement rods per assembly. (Note: This report is relevant to both BWRs and PWRs.)

    - Reconstituted assemblies located at the outer edge of the core are limited to 26 percent inert rods per assembly, but are not limited to two inert rods or one inert rod and a guide tube per subchannel.

- Observations: The reconstitution methodology is applicable to all fuel designs.

4.0 Thermal and Hydraulic Design

Thermal-hydraulic analyses of the fuel and core are performed to verify that design criteria are satisfied and to establish an appropriate value for the MCPR fuel cladding integrity safety limit.

4.1 *Regulatory Requirements*

The acceptance criteria of SRP Section 4.4 Thermal Hydraulic Design are based on meeting the relevant requirements of General Design Criterion 10, as it relates to the reactor core design, with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or AOOs.

4.2 *Thermal and Hydraulic Design Analyses*

4.2.1 Hydraulic Compatibility

Design Criteria. The hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to existing fuel in the reactor that there is no significant degradation in total core flow or maldistribution of the flow among assemblies in the core.

Bases. The Standard Review Plan(1) does not contain an explicit criterion for fuel assembly hydraulic compatibility. However, flow differences between assembly types in a mixed core need to be accounted for in assuring that all design criteria are satisfied.

The component hydraulic resistances in the reactor core are determined by a combination of both analytical techniques and experimental data. For example, the single-phase flow resistances of the inlet region, bare rod region, spacers, and upper tie plate of the SPC fuel designs and representative co-resident designs are generally determined in single phase flow tests with full scale assemblies. The two-phase flow resistances of appropriate components are determined from the single-phase loss coefficients and two-phase flow models. The prediction of pressure drop by a combination of single-phase loss coefficients and two-phase flow models has been experimentally verified.

The SPC thermal-hydraulic methodology implicitly includes the impact of assembly differences on the individual assembly flow. The overall criterion for acceptability is that individual fuel types must be in compliance with the thermal hydraulic limits. To assure this, for reload analyses, if there is more than an **[ ]** difference in assembly orifice flow at rated conditions (i.e., full flow and full power), additional core stability evaluation will be performed with the STAIF methodology (Reference 3-4). The purpose of these evaluations is to better define the core stability behavior with this mismatch in flow. The MCPR performance remains protected by compliance with the safety and operating limits.

4.2.2 Thermal Margin Performance

Design Criteria. The fuel design shall fall within the limits of applicability of the approved critical heat flux (CHF) correlation. New fuel assembly designs and/or changes in existing assembly designs shall minimize the likelihood of boiling transition during normal reactor operation and AOOs. The applicable critical power correlation will be used to determine the operating limits and for consistency will be used to monitor the core.

Bases. SPC fuel and reload cores are designed so that operation within the technical specification limits ensures that 99.9% of the fuel rods are expected to avoid boiling transition during AOOs. An NRC-approved CHF correlation is used by SPC to determine operating and safety limits during the design of a reload core, and, for consistency, the same CHF correlation is used to monitor the core during operation.

Operation of a BWR requires protection against fuel damage during normal reactor operation and AOOs. A rapid decrease in heat removal capacity associated with boiling transition could result in high temperatures in the cladding, which may cause cladding degradation and a loss of fuel rod integrity. Protection of the fuel against boiling transition assures that such degradation is avoided. This protection is accomplished by determining the operating limit minimum critical power ratio (OLMCPR) each cycle.

The SPC thermal limits analysis methodology, THERMEX, is described in Reference 4-2. The thermal limits methodology in THERMEX consists of a series of related analyses which establish an OLMCPR. The OLMCPR is determined from two calculated values, the safety limit MCPR (SLMCPR) and the limiting transient ΔCPR. The overall methodology is comprised of four major segments: 1) reactor core hydraulic methodology, 2) a critical power correlation, 3) plant transient simulation methodology, and 4) critical power methodology.

SPC fuel assembly pressure drop methodology is presented in Reference 4-3. This methodology is part of the calculational method used by SPC to determine the assembly pressure drop that is used to calculate assembly flows for a BWR core. The pressure drop methodology determines the void fraction and the two-phase pressure losses, which are in turn used as input to the calculation of the assembly pressure drop using the XCOBRA computer code described in Reference 4-2.

The SPC fuel assembly critical power performance is established by means of an empirical correlation based on results of boiling transition test programs (see References 4-4, 4-5, and 4-6). The critical power performance of co-resident fuel, which is not in the SPC correlation development data base, is determined using the methodology described in Reference 4-7.

The methodology and computer codes for SPC BWR plant transient analyses are the XCOBRA-T code (Reference 4-8) and the COTRANSA2 code (Reference 4-9). The COTRANSA2 code is used to calculate BWR system behavior for steady-state and transient conditions. This behavior is then used to provide input to the XCOBRA-T and XCOBRA codes, from which critical power ratios are determined for limiting transients.

The critical power ratio methodology is the approach used by SPC to determine thermal margin for BWRs. The SPC critical power methodology for BWRs is presented in Reference 4-10.

Reference 4-10 provides the basis for the SPC methodology for determining the operating safety limit for minimum critical power (SLMCPR) which ensures that 99.9% of the fuel rods are expected to avoid boiling transition. The safety limit is determined by statistically convoluting hydraulic and thermal calculational uncertainties with plant measurement uncertainties associated with the calculation of MCPR. This determination is carried out by a series of Monte Carlo calculations in which the variables affecting of boiling transition are varied randomly and the total number of rods experiencing boiling transition is determined for each Monte Carlo trial. The SPC CPR correlation depends on the core coolant pressure, channel mass velocity, planar enthalpy, a local peaking function, radial and axial power, and channel geometry (channel bow).

Power distribution uncertainties used in the calculation are those associated with the core monitoring system and are obtained from references such as Reference 3-2. The CPR correlation uncertainty is accounted for through the additive constant uncertainty. The additive constant uncertainties for specific fuel designs used in the determination of the SLMCPR are determined using the methodologies and values provided in References 4-4, 4-5, 4-7, and 4-11.

Plant measurement uncertainties (such as pressure, core flow, feedwater temperature, etc.) are plant dependent and are obtained from the utility.

4.2.3 Fuel Centerline Temperature

Design Criteria. Fuel design and operation shall be such that fuel centerline melting is not predicted for normal operation and AOOs.

Bases. This design criteria is addressed during the specific mechanical design analysis performed for each fuel type. The bases is discussed in Section 2.2.12.4 of this document.

4.2.4 Rod Bowing

Design Criteria. The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margins requirements.

Bases. The bases for rod bow are discussed in Section 2.2.6. Rod bow magnitude is determined during the mechanical design analyses done for each fuel type. The need for a thermal margin rod bow penalty is evaluated on a plant and cycle specific basis. Post-irradiation examinations of BWR fuel fabricated by SPC show that the magnitude of fuel rod bowing is small and the potential effect of this bow on thermal margins is negligible. Rod bow at extended burnups does [ ] of the lower powers experienced by high exposure assemblies.

4.2.5 Bypass Flow

Design Criteria. The bypass flow characteristics of the reload fuel assemblies shall not differ significantly from the existing fuel in order to provide adequate flow in the bypass region.

Bases. The Standard Review Plan(1) does not contain an explicit criterion for fuel assembly bypass flow characteristics. However, significant changes in bypass region flow may alter the response characteristics of the incore neutron detectors. In order to avoid altering the incore neutron detector response characteristics, SPC evaluates bypass flow fraction on a plant and cycle specific basis to assure that the bypass flow characteristics are not significantly altered.

4.3 *NRC-Accepted Topical Report References*

The NRC has approved the following licensing topical reports that describe the methods and assumptions used by SPC to demonstrate the adequacy of its thermal and hydraulic fuel system design analyses. These reports address thermal and hydraulic criteria and thermal conditions used in steady-state and transient licensing analyses. The purpose of each topical report and restrictions on the methods presented are described below.

4-1: EMF-CC-074(P)(A) Volume 1, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994.

- Purpose: Provide a methodology for the determination of the thermal-hydraulic stability of BWRs, including reactivity feedback effects.

- SER Conclusions/Restrictions:

  1. The core model must be divided into a minimum of 24 axial nodes.

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

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

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

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

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

  4. The collapsed 1-D cross sections must represent the actual conditions being analyzed as closely as possible, including control rod positions.

  5. The STAIF calculation must use the "shifted Nyquist" or complex pole search feature to minimize the error at low decay ratio conditions.

- Observations: Stability analyses procedures described in XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983 (See Section 3.2, Reference 3-1) were superceded by the approval of the STAIF code.

4-2: XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.

- Purpose: Provide an overall methodology for determining a MCPR operating limit. The methodology comprises CHF correlations, fuel hydraulic characteristics, safety limit analyses, AOO analyses, and statistical convolution of uncertainties.

- SER Conclusions/Restrictions: The core monitoring system used (POWERPLEX or other) should be specifically identified in plant submittals referencing the THERMEX methodology.

Hot channel calculations with COTRANSA were not approved pending review of XN-NF-86-113.

ENC submitted methodology for application to ENC 8x8 and 9x9 fuel types.

- Observations: Although this topical report references only applications to ENC 8x8 and 9x9 fuel types, the overall methodology is still applicable to SPC 10x10 and new fuel types when appropriate CHF correlations and fuel hydraulic characteristics are implemented within the methodology.

  The review of XN-NF-86-113 was deferred to the review of the topical report for the currently used COTRANSA2 code. (See Reference 4-9)

  Some of the computer codes referenced in the topical report have been superceded by other NRC-approved codes (e.g., COTRANSA with COTRANSA2, XTGBWR with MICROBURN-B) and the XN-3 CHF correlation has been supplemented with the NRC-approved ANFB and ANFB-10 CHF correlations. (See References 4-4 and 4-5)

  The SER states "Based on the similarity of the computational models of the two codes (XCOBRA and XCOBRA-T) and the NRC approval of the XCOBRA-T code (Reference 13), we find the use of the steady-state code [XCOBRA] acceptable in this context." XCOBRA continues to be applied for steady-state analyses.

4-3: XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983.

- Purpose: Develop a methodology for determining the BWR assembly pressure drop which determines the assembly coolant flow and which varies with total recirculating flow and reactor power.

- SER Conclusions/Restrictions: Favorable comparisons of calculated pressure drops with data from CHF tests and the use of widely accepted methods that are found in the open literature serve as the bases for finding this methodology acceptable.

- Observations: This methodology continues to be used with some adaptation to experimental pressure drop data for new fuel and spacer designs.

4-4: ANF-1125(P)(A) and Supplement 1, "ANFB Critical Power Correlation," Advanced Nuclear Fuels Corporation, April 1990.

- Purpose: Implement an improved critical power correlation for SPC fuel designs.

- SER Conclusions/Restrictions:

  1. Range of correlation limited to thermal and coolant conditions in Tables 1.1 and 1.2 of Supplement 1.

  2. Departure from nucleate boiling (DNB) to be assumed if conditions occur outside the conditions identified above.

3. Additive constants for new fuel designs must be justified.

4. Correlation applicable to Chapter 15 transients excluding LOCA.

5. The application of the correlation for the determination of rods in boiling transition must use the uncertainties incorporated in the plant safety limit methodology.

- Observations: To assure that results from licensing analyses and core monitoring are within the range of applicability for the correlation required in SER Restriction 1, SPC developed programming changes for licensing and core monitoring codes to check and to take appropriate actions based on these limits.

  SPC developed a process to assure boiling transition does not occur in the lower part of a fuel assembly. This process, which uses the Hench-Levy limit line, was accepted by the NRC. [20]

4-5: EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," Siemens Power Corporation, July 1998.

- Purpose: Develop a critical power correlation specifically for ATRIUM -10 fuel.

- SER Conclusions/Restrictions:

  1. The ANFB-10 critical power correlation is applicable to ATRIUM-10 fuel with a design local peaking of ≤ 1.5.

  2. If the local peaking factor of 1.5 is exceeded during the process of calculating the MCPR safety limit, an additional uncertainty of 0.018 will be imposed on a rod by rod basis.

  3. The ANFB-10 correlation acceptable range of parameters are:

Pressure (psia)	571 to 1415
Mass Flow Rate (Mlb/hr-ft^2)	0.115 to 1.5
Inlet Subcooling (Btu/lbm)	5 to 149

- Observations: This topical report is only applicable to ATRIUM-10 fuel.

4-6: EMF-1997 Supplement 1(P)(A) Revision 0, "ANFB-10 Critical Power Correlation: High Local Peaking Results," Siemens Power Corporation, July 1998.

- Purpose: Demonstrate that the set of additive constants for ATRIUM-10 fuel in Reference 4-5 requires no additional uncertainties for fuel rods at higher local peaking.

- SER Conclusions/Restrictions: Same as listed in Reference 4-5.

- Observations: When local peaking factors exceed 1.5, an additional uncertainty of 0.018 will be imposed on a rod by rod basis. This is expected to only affect non-limiting controlled bundles.

4-7: EMF-1125(P)(A) Supplement 1 Appendix C, "ANFB Critical Power Correlation Application for Co-Resident Fuel," Siemens Power Corporation, August 1997.

- Purpose: Develop a method to determine additive constants and uncertainties for co-resident fuel not within the ANFB correlation data base.

- SER Conclusions/Restrictions:

  1. This methodology is applicable to once burned co-resident fuel. Lead assemblies are excluded.

  2. A table comparing MCPR data throughout the first reload exposure must be submitted to justify each plant application.

- Observations: The table mentioned in 2. above is required to be provided to the NRC by the licensee for the first reload exposure with co-resident fuel.

4-8: XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.

- Purpose: Provide a capability to perform analyses of transient heat transfer behavior in BWR assemblies.

- SER Conclusions/Restrictions:

  1. XCOBRA-T was found acceptable for the analysis of only the following licensing basis transients:

     a) Load rejection without bypass

     b) Turbine trip without bypass

     c) Feedwater controller failure

     d) Steam isolation valve closure without direct scram

     e) Loss of feedwater heating or inadvertent high pressure coolant injection (HPCI) actuation

     f) Flow increase transients from low-power and low-flow operation

  2. XCOBRA-T analyses that result in any calculated downflow in the bypass region will not be considered valid. (This restriction applies "only to those transients for which modeling of the bypass flow is part of the calculation or negative bypass flow can be shown to be a significant contributing factor in the calculation of the critical heat flux.")

3. "A concern was expressed regarding the comparison of void-profile results calculated with the XCOBRA-T code with experimental bundle data..." This concern was remedied with the submittal and approval of the topical report XN-NF-84-105(P)(A), Volume 1 Supplement 4, "XCOBRA-T: A computer code for BWR Transient Thermal-Hydraulic Core Analysis Void Fraction Model Comparison to Experimental Data." [21]

- Observations: The methodology has been approved for plant specific applications[22-23] to transients other than those listed in the SER.

4-9: ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

- Purpose: Develop an improved computer program for analyzing BWR system transients.

- SER Conclusions/Restrictions:

The use of COTRANSA2 is subject to the limitations set forth in the safety evaluations for the methodologies described and approved for XCOBRA-T and COTRAN. The staff reviewed the subject safety evaluations and identified the following limitations that apply to COTRANSA2:

1. XCOBRA-T was found acceptable for the analysis of only the following licensing basis transients:

    a) Load rejection without bypass

    b) Turbine trip without bypass

    c) Feedwater controller failure

    d) Steam isolation valve closure without direct scram

    e) Loss of feedwater heating or inadvertent HPCI actuation

    f) Flow increase transients from low-power and low-flow operation

2. Based on a similar limitation on XCOBRA-T thermal-hydraulic modeling, the COTRANSA2 code is not applicable to the analysis of any transient for which lateral flow in a bundle is significant and non-conservative in the calculation of system response.

3. Based on a similar limitation on XCOBRA-T modeling, for those analyses in which core bypass is modeled, the effect of a computed negative flow in the core bypass region should be shown to make no significant non-conservative contribution in the calculation of system response.

4. Licensing applications referencing the COTRANSA2 methodology must include confirmation that sensitivity to the time step selection has been considered in the analysis.

Observations: The COTRANSA2 SER restrictions are similar to those for XCOBRA-T. (See Reference 4-8) As with XCOBRA-T, COTRANSA2 has been approved for transients for specific plant applications[22-23] not listed in the SER.

4-10: ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.

- Purpose: Provide a methodology for the determination of thermal margins.
- SER Conclusions/Restrictions:
  1. NRC-approved MICROBURN-B power distribution uncertainties should be used to determine SLMCPR.
  2. ANFB additive constant uncertainties should be verified for each plant-specific application.
  3. Conservative channel bowing penalty estimates for non-SPC fuel should be used.
  4. Channel bowing methodology is not applicable to second-lifetime channels.
- Observations: The critical power methodology is a general methodology which may be used with all SPC developed CHF correlations that include additive constants and additive constant uncertainties.

  Power distribution uncertainties for MICROBURN-B and other SPC core simulator codes approved by the NRC will be used in the CPR methodology.

  As additive constants and additive constant uncertainties are fuel type specific, they do not change for each plant specific application, as noted in SER restriction 2.

4-11: ANF-1125(P)(A) Supplement 1 Appendix E, "ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, " Siemens Power Corporation, September 1998.

- Purpose: Determine the additive constant uncertainty for ATRIUM-9B fuel from an extended ATRIUM-9B CHF data base.
- SER Conclusions/Restrictions:
  1. The additive constant uncertainty of 0.027 is applicable to SPC ATRIUM-9B fuel with a local peaking factor $\leq$ 1.22.
  2. For ATRIUM-9B fuel with a local peaking factor > 1.22, with a maximum peaking design limit of 1.3, an uncertainty of 0.029 will be imposed on a rod by rod basis.
  3. The additive constant and additive constant uncertainties for ATRIUM-9B fuel are applicable for the following parameter ranges.

Pressure (psia)	600 to 1400
Mass Flow Rate (lb/s)	4.8 to 41.7
Inlet Subcooling (Btu/lbm)	8 to 82

- Observations: This topical report is only applicable to ATRIUM-9B fuel.

4-12: ANF-90-82(P)(A) Revision 1 and Revision 1 and Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.

- Purpose: Develop a methodology to justify reinsertion of irradiated fuel assemblies, which have been reconstituted with replacement rods, into a reactor core. Replacement rods can be fuel rods containing natural uranium pellets, water rods, and inert rods containing zircaloy or stainless steel inserts.

- SER Conclusions/Restrictions: The reconstitution methodology is acceptable for reload licensing applications with the following conditions:

  - BWR reconstituted assemblies are limited to 9 rods per assembly.

  - PWR reconstituted assemblies are limited to two inert rods or one inert rod and a guide tube per subchannel, with a total of no more than 26 percent inert replacement rods per assembly.

  - Reconstituted assemblies located at the outer edge of the core are limited to 26 percent inert rods per assembly, but are not limited to two inert rods or one inert rod and a guide tube per subchannel.

- Observations: The reconstitution methodology is applicable to all fuel designs.

5.0 Accident Analysis

This section addresses the methodologies used to perform the analyses of those AOOs and postulated accidents in SRP Chapter 15 that are related to core reloads, and how the analytical results meet defined acceptance criteria.

5.1 *Regulatory Requirements - AOOs*

AOOs involving the entire core and the recirculation system are evaluated at full power and flow conditions to determine the nominal MCPR limit. Some events are also analyzed at off-rated conditions. Table 5.1 lists those AOOs analyzed with SPC-approved methodologies.

Table 5.1 Anticipated Operational Occurrences Requiring Analyses

SRP No.	Chapter 15 Accident Analysis
15.1.1 - 15.1.2	Decrease in Feedwater Temperature and Increase in Feedwater Flow
15.2.1 - 15.2.2	Loss of External Load and Turbine Trip
15.2.4	Closure of Main Steam Isolation Valve (BWR
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power
15.4.5	Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory

The specific criteria necessary to meet the requirements of the relevant GDCs 10, 15, and 26 for AOOs listed in Table 5.1 (except SRP No. 15.4.2) are:

a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

b) Fuel cladding integrity shall be maintained by ensuring that the CPR remains above the MCPR safety limit for BWRs based on acceptable [CHF] correlations. (See SRP Section 4.4)

c) The positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints" are used.

d) The events should not generate a more serious plant condition without other faults occurring independently.

e) The most limiting single failures shall satisfy Regulatory Guide 1.53.

The criteria necessary to meet GDCs 10, 20, and 25 for SRP 15.4.2 AOO are:

a) The thermal margin limits (MCPR) specified in SRP Section 4.4, II.1 are met, and

b) Uniform cladding strain does not exceed 1%.

5.1.1 Anticipated Operational Occurrences Analyses

Analyses are performed to demonstrate that the fuel performs within design criteria for boiling transition during infrequent and moderately frequent AOOs and to establish appropriate operating limits for the reactor. The methodology used for the analysis of these anticipated events is found in References 5-1 through 5-9.

To protect the established MCPR safety limit, evaluations of AOOs are performed which produce the limiting transient ΔCPR, which when added to the safety limit, defines the MCPR operating limit.

5.1.1.1 Limiting Transient Events

The loading of fresh fuel, regardless of design, into a reactor core may alter the characteristics of both steady state core performance and plant transient response throughout each subsequent cycle of operation. Limiting criteria for plant operations are established to assure that acceptable thermal operating margins are maintained during all anticipated operations. Application of SPC's methodology as described in Reference 5-1 provides a basis for the determination that plant operation will meet appropriate safety criteria.

The evaluation of anticipated operation occurrences considers events identified in the FSAR. These events are generally classified as:

- Rapid vessel pressurization
- Decrease in recirculation flow rate
- Increase in recirculation flow rate
- Decrease in core inlet subcooling
- Increase in core inlet subcooling
- Decrease in vessel coolant inventory
- Increase in vessel coolant inventory

Prior to the initial cycle that SPC provides reload fuel for, a disposition of events is performed to identify the FSAR events that may be affected by a change in fuel or core design. From the affected events, the potentially limiting events relative to thermal margins are identified and analyzed. The following events are generally identified as being potentially limiting:

- turbine/generator trip without bypass
- loss of feedwater heating
- recirculating flow increase events
- feedwater flow increase

- inadvertent ECCS high-pressure subsystem startup
- turbine or generator trip without bypass and main steam isolation valve (MSIV) closure without direct scram*

The events described above were also found to be the most limiting of their type due to the severity of the initiating cause. Other events including loss of recirculating coolant flow or increase of recirculating coolant enthalpy are inherently non-limiting due to the characteristic negative void reactivity feedback of a BWR.

Primarily because of the strong void reactivity feedback characteristic of a boiling water reactor, incidents involving decrease in vessel coolant inventory and flow and events involving a decrease in vessel coolant subcooling are not expected to result in a limiting ΔCPR. Events involving either an increase in core inlet subcooling or rapid pressure increases are considered potentially limiting transients for all BWR designs.

A decrease in feedwater enthalpy may result in a gradual core heatup until the high neutron flux scram setpoint is exceeded. Since the gradual nature of the power excursion assures that the fuel thermal response will not significantly lag the neutronic response, this event can be evaluated with either a transient code or a steady-state code. The possible mitigation of this event with an effective flow control system would not normally be assumed in the analysis.

Rapid pressure increases may be a thermal margin limiting event for some designs and conditions. The severity of the event is strongly dependent upon the reactivity state of the core, the valve closure characteristics initiating the event, and the performance of the scram shutdown system. Thus, specific event sequences at some reactor conditions may emerge as consistently most limiting in nature. Each potentially limiting event will be considered in the determination of cycle limiting conditions for operation.

The remaining two single event categories which involve increases in either coolant flow rate or inventory are dependent upon plant design and conditions. Both involve potentially limiting conditions at partial power and flow conditions, where the augmentation of flow (either recirculatory or feed) to the maximum physical capacity of equipment is greatest. Effective designs and/or reactor protection systems may substantially mitigate the rate and potential acceleration of power production in the core or terminate the transient prior to serious degradation of thermal margin. Current technical specifications for Jet Pump-BWRs provide an augmentation of the CPR operating limit to protect against potentially greater transient reduction of CPR at partial flow conditions. Existing augmentation procedures (i.e., MCPR curves) for a plant are verified to ensure they provide adequate thermal margin for the cycle design at applicable conditions.

Once the applicable set of limiting transients for thermal margin has been identified for a specific reactor, the evaluation of each event at limiting reactor conditions will provide the basis for determining the MCPR operating limit, which is applicable to all other anticipated operating conditions.

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*Evaluated for compliance with the provisions of the ASME code.

5.1.1.2 Analysis of Plant Transients at Rated Conditions

Potentially limiting anticipated are evaluated at full power and flow conditions to determine the rated conditions MCPR limit. Some events are also analyzed at off-rated conditions as discussed in Sections 5.1.1.3 and 5.1.1.4. The limiting transient event (or events) is (are) evaluated using the plant transient methodology described in Reference 5-1.

5.1.1.3 Analyses for Reduced Flow Operation

Protection of the MCPR fuel cladding integrity safety limit is assured during reduced flow operation through application of a flow dependent MCPR operating limit which is established independently of the full flow MCPR limits through analyses of the flow dependent transients from reduced power and flow settings.

Flow dependent MCPR limits are typically established to support the manual flow control (MFC) mode of operation. A flow excursion event during the MFC mode of operation is considered an anticipated operational occurrence and the flow dependent MCPR limit is set to ensure that the safety limit MCPR is supported if the recirculation flow is inadvertently increased to the maximum attainable based on equipment limitations. Limits may also be established to support operation in the Automatic Flow Control (AFC) mode. AFC operation is typically used in load follow applications where the power level is changed by "automatically" adjusting the recirculation flow rate. Since this is considered normal operation, the AFC flow dependent MCPR limits are set to protect the operating limit MCPR during the flow excursion.

Analyses are performed from various points on the power-flow operating map to demonstrate that the flow-dependent MCPR limits provide the necessary protection.

A special case of operation at less than rated power and flow is operation with a single recirculation loop out of service. It may be desirable to operate the reactor with a single loop if one component requires extensive maintenance. Analysis of single loop operation is performed on a plant specific basis, where needed.

5.1.1.4 Analysis for Reduced Power Operation

Reactor operation at less than full power is evaluated to determine the adequacy of the operating limits to protect against fuel failures during events initiated from low power conditions. If a need for reduced power operating limit augmentation is shown, results of these analyses are used to establish a power dependent MCPR limit function which protects the MCPR fuel cladding integrity safety limit during the occurrence of anticipated events from power-flow states less than nominal rated power.

5.1.1.5 ASME Overpressurization Analysis

An overpressurization analysis is performed to assure that the vessel pressure requirements of the ASME Code are satisfied. This analysis, which presumes failure of all non-safety grade components, does not contribute to the determination of thermal margin requirements.

The turbine trip or generator load rejection transient is generally more limiting in regard to thermal margin requirements than the containment isolation event. However, the MSIV closure event with the assumption of failure of the direct position scram may result in a more severe calculated overpressurization. [ ]

The ASME overpressurization event is analyzed using COTRANSA2 (Reference 5-1).

5.1.1.6 Generic Loss of Feedwater Heating Methodology

The NRC has approved a generic SPC methodology for evaluating the loss of feedwater heating (LFWH) transient in BWRs (Reference 5-5). The generic methodology is a parametric description of the critical power ratio response that was developed using the results of many applications of the currently approved plant and cycle specific methodology (References 5-6 and 5-7). SPC developed a critical power [ ] caused by the event. Applying this function yields a conservative MCPR operating limit for the LFWH event.

5.1.1.7 Control Rod Withdrawal Error

Withdrawal of the highest worth control rod in the core (in error) until its movement is blocked by the control system is evaluated with MICROBURN-B as described in Reference 5-6.

For this analysis, the reactor is assumed to be in a normal mode of operation with the control rods being withdrawn in the proper sequence and all reactor parameters within technical specification limits and requirements. The most limiting case is when the reactor is operating at power with a high reactivity worth control rod fully inserted. To maximize the worth of the control rod, the reactor is assumed to be xenon free and the control rod with the maximum rod worth is selected as the error rod. When necessary, the partially withdrawn control rods in the core are adjusted to place the fuel near the inserted control rod near thermal limits.

During the control rod withdrawal transient, the reactor operator is assumed to ignore the local power range monitor (LPRM) alarms and the rod block monitor (RBM) alarms and continue to withdraw the control rod until the control rod motion is stopped by the Control Rod Block. In addition, a limiting combination of LPRM failures is assumed.

The results are determined parametrically by RBM setting. The RBM setting which allows the greatest operational flexibility without unnecessarily restricting thermal margins is selected for implementation on a cycle specific basis.

Results for the control rod withdrawal error analysis include maximum control rod withdrawal distance and change in thermal margins as a function of RBM setting. For reactors utilizing reduced power augmentation to MCPR limits, the existing reduced power limits are revised as necessary. A detailed description of the SPC control rod withdrawal error evaluation methodology is given in Reference 5-7.

For BWR/6 reactors, the NRC approved SPC's generic control rod withdrawal error analysis (Reference 5-8). The generic analysis has been extended to cover maximum extended operating domain (MEOD) operation (Reference 5-9). SPC demonstrated that at a [ ] of the rod withdrawal events.

The SPC generic analysis of the rod withdrawal error event assumes operation at the Technical Specification rod withdrawal limits as a function of power as established by the reactor designer. These limits are one foot for core powers greater than 70% of rated power and two feet for core powers between 20% and 70% of rated. The generic analyses were performed to establish the values of operating limit MCPR as a function of core power which are required to assure that the Safety Limit MCPR will not be violated for the rod withdrawal event.

The calculational methods and procedures used to simulate the rod withdrawal event are the approved methods and procedures described in Reference 5-8.

5.1.1.8 Determination of Thermal Margins

The results of the evaluation of the anticipated operational occurrences are compared for the greatest change in MCPR at full power operation. For reactors using a rod block monitor the limiting transient ΔCPR which is used to define the MCPR operating limit is used to select the rod block monitor setting used from the tabulated results of the control rod withdrawal error analysis. Observance of the operating MCPR limit and rod block monitor settings determined in this fashion provides protection of the MCPR fuel cladding integrity safety limit during operation at rated conditions.

The results of the reduced flow and reduced power analyses are used to establish the proper values for the MCPR limit functions required for operation at less than rated power and flow conditions. Reactor operation within the power- and flow-dependent limits defined in this fashion assures adequate protection of MCPR limits throughout the power-flow operating map.

The scram insertion time used for the transient analyses may be based on either the technical specifications or plant measurement data. If plant measurement data are used to determine the scram performance used to define any of the limiting safety system settings or limiting conditions for operation, surveillance procedures are specified to determine the continued applicability of the data base and to modify limits to assure applicability of the analysis.

The results of reduced power and reduced flow analyses are used to ensure that the 1% strain and centerline melt criteria are met during anticipated operational occurrences. If adjustments to operating limits are needed, power and flow dependent [

] to determine the applicable LHGR operating limit.

The core power and exposure distributions are monitored by the licensee throughout the cycle to assure that the end-of-cycle (EOC) axial power shape assumed in the licensing analysis will bound the actual EOC axial power shape.

5.2 *Regulatory Requirements – Postulated Accidents*

Postulated accidents for BWRs evaluated for compliance with relevant GDCs are listed in Table 5.2 below.

Table 5.2 Postulated Accidents Analyses

SRP No.	Chapter 15 Accident Analysis
15.3.3	Reactor Coolant Pump Rotor Seizure
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
15.4.9	Spectrum of Rod Drop Accidents (BWR)
15.4.9.A	Radiological Consequences of Control Rod Drop Accident
15.6.5	Loss-of-Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary
15.7.4	Radiological Consequences of Fuel Handling Accidents

The specific analytical criteria that are necessary to meet the requirements of the relevant GDCs for postulated accidents in Table 5.2 are:

SRP No. 15.3.3; GDCs 27, 28,and 31

a) Pressure in the reactor coolant and main steam systems should be maintained below design limits, considering potential brittle or well or ductile failures.

b) A small fraction of the fuel failures may occur, but these failures should not hinder the core coolability.

c) Radiological consequences should be a small fraction of 10 CFR 100 guidelines (generally < 10%).

d) The events should not generate a limiting fault or result in the consequential loss of the function of the reactor coolant system or containment barriers.

SRP No. 15.4.7; GDC 13

a) Offsite consequences due to fuel rod failure limits being exceeded during this postulated accident should be a small fraction of 10 CFR 100 limits.

SRP No. 15.4.9; GDC 28

a) Reactivity excursions should not exceed a radially averaged fuel rod enthalpy greater than 280 cal/g at any axial location in any fuel rod.

b) The maximum reactor pressure should be less than "Service Limit C" defined in the ASME code.[6]

c) The number of fuel rods predicted to reach assumed fuel failure thresholds and associated parameters such as the amount of fuel reaching melting conditions will be input to a radiological evaluation. The assumed failure thresholds are radially averaged fuel rod enthalpy greater than 170 cal/g at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

SRP No. 15.4.9A

a) Calculated exposure values should be less than 25% of the 10 CFR 100 exposure guideline values. The fission product source term used in the dose analysis is acceptable if it meets the guidelines of Regulatory Guide 1.77.[11]

SRP No. 15.6.5; GDC 35

a) Event-specific criteria are specified in:

    1) 10 CFR 50.46 and 10 CFR 50 Appendix K.

b) Regulatory Guide 1.3[24] establishes a set of fission gas release fractions to be applied for radiological assessments. Radiological consequences are within the guidelines of 10 CFR 100.

SRP No. 15.7.4; GDC 61

a) Calculated exposure values should be less than 25% of the 10 CFR 100 exposure guideline values.

b) The model for calculating the whole-body and thyroid doses is acceptable if it incorporates the appropriate conservative measurements in Regulatory Guide 1.25[25], with the exception of the guidelines for the atmospheric dispersion factors (X/Q values). The acceptability of the X/Q values is determined under SRP Section 2.3.4.

5.2.1 Postulated Accidents Analyses

The methodologies used to analyze the hypothetical Loss of Coolant Accidents (LOCAs) and other postulated accidents are discussed in the following sections. References 5-1 to 5-4 and 5-6 and 5-7 are used to analyze postulated accidents which may affect MCPR limits and the number of fuel rods in boiling transition; References 5-10 through 5-17 comprise the methodologies used to perform small break and large break LOCA analyses.

5.2.1.1 Pump Seizure

Recirculation pump seizure (RPS) event is considered an accident where an operating recirculation pump suddenly stops rotating. RPS event can be analyzed for both two-loop operation or single-loop operation. There are three parts to the RPS analysis - the event/system model, determination of the number of fail rods, and the radiological dose assessment.

The first part of the analysis uses the COTRANSA2 and XCOBRA-T codes to simulate the system and limiting assembly response. The key parameter determined is the minimum CPR during the event. The second part is the determination of the number of failed rods. It is assumed that all rods that experience boiling transition are assumed to fail. This is a very conservative assumption because the minimum CPR occurs for a short period of time. The third part determines the dose from the number of rods which were calculated to fail. If the minimum CPR during the event remains above the safety limit MCPR then dose calculation is not needed since operation at or above the safety limit MCPR meets the requirements of less than a small fraction of the 10 CFR 100 dose limits. If the minimum CPR is below the safety limit MCPR, dose calculations are performed for the number of rods calculated to fail.

Depending on the specific UFSAR licensing requirements for a given reactor, RPS is specified as either an infrequent event or a limiting fault/design basis accident. For an infrequent event, the dose calculation must remain below 10% (a small fraction) of the 10 CFR 100 limits. For a limiting fault/design basis accident, the dose calculation must not exceed 10 CFR 100. If RPS is defined as a limiting fault/design basis accident, it is generally qualitatively dispositioned as mild and non-limiting as compared to a LOCA accident.

5.2.1.2 Fuel Loading Error

Two separate incidents are analyzed as part of the fuel misloading analysis. The first incident, which is termed the fuel mislocation error, assumes a fuel assembly is placed in the wrong core location during refueling. The second incident, the fuel misorientation error, assumes that a fuel assembly is misoriented by rotation through 90° or 180° from the correct orientation when loaded into the reactor core. For both the fuel mislocation error and the fuel misorientation error, the assumption is made that the error is not discovered during the core verification and the reactor is operated during the cycle with a misloaded fuel assembly. Criteria for acceptability of the fuel misloading error analyses are that the off-site dose due to the event shall not exceed a small fraction of the 10 CFR 100 limits,[4] see Reference 3-3.

> Misloaded Fuel Bundle: The inadvertent misloading of a fuel bundle into an incorrect core location is analyzed with the MICROBURN-B methodology described in Reference 5-6. One approach to assuring that the 10 CFR 100 criteria are met is to calculate the minimum value of the MCPR in the misloaded core and the maximum LHGR in the mislocated bundle. If the resulting minimum CPR is lower than the MCPR safety limit, the core configuration and power distribution are used to verify that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition during full power operation with the misloaded bundle. This prediction of the number of fuel rods in boiling transition is performed in accordance with the methodology reported in Reference 5-4. The maximum LHGR is verified to be less than the transient LHGR limit.

> Misoriented Fuel Bundle: The inadvertent rotation of a fuel bundle from its intended orientation is evaluated with the CASMO-3G methodology described in Reference 5-6. Similar to the analysis for misloaded fuel above, a minimum value of MCPR and a maximum LHGR associated with the orientation error are calculated. If the resulting minimum CPR is lower than the MCPR safety limit, the core configuration and power distribution associated with the misorientation error are used to verify that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition during full power operation with the misoriented bundle. This prediction of the number of fuel rods in

boiling transition is performed in accordance with the methodology reported in Reference 5-4. The maximum LHGR is verified to be less than the transient LHGR limit.

5.2.1.3 Control Rod Drop Accident Analysis

Analysis of the postulated CRDA is performed on a generic basis in Reference 5-7. Because the behavior of the fuel and core during such an event is not dependent upon system response, a single generic CRDA analysis can be applied to all BWR types. The applicability of the generic CRDA analysis is verified for each application.

The results of the generic CRDA analysis consist of deposited fuel enthalpy values parameterized as a function of effective delayed neutron fraction, doppler coefficient, maximum (dropped) control rod worth, and four-bundle local peaking factor. Each cycle-specific application includes the values for each of the parameters, which are compared to the generic analysis and curves and the resulting deposited fuel enthalpy determined.

5.2.1.4 Loss of Coolant Accident Analysis

ECCS analyses provide peak cladding temperature (PCT) and maximum local metal-water reaction (MWR) values and establish MAPLHGR limits for each plant analyzed. For the limiting single failure and limiting break, calculations are undertaken to determine the MAPLHGR, PCT, and MWR values over the expected exposure lifetime of the fuel. The limiting break is determined by evaluating a spectrum of potential break locations and sizes. [

]

Worst Single Failure: SPC works closely with the licensee to identify the limiting (worse) single failure of ECCS equipment. The limiting single failure of ECCS equipment is that failure which results in the minimum margin to either PCT or MWR criterion.

Previous evaluations by the NSSS vendor identifying the worst case single failure of ECCS equipment are used in identifying potentially limiting single failures to be evaluated with SPC methodology. The applicability of the previous evaluations will be reviewed on a case-by-case basis. Additional calculations may be performed if plant modifications have occurred subsequent to previously reported worst case single failure ECCS analyses.

Break Location Spectrum: Evaluations and analyses to establish the location of the limiting break are performed on a generic basis. Analyses performed by the NSSS vendor may be used to narrow the scope of the analyses. While any piping connected to the reactor vessel could break during a LOCA, only breaks in the recirculation piping are analyzed since these are the largest pipes below the core water level. Analyses are performed for breaks on the suction and discharge sides of the recirculation pump. The determination of the limiting location is based on minimum margin to either PCT or MWR criterion calculated for consistent fuel exposure conditions at each of the break locations.

Break Size Spectrum: Analyses to establish the size of the limiting break are performed on a generic basis. Hypothetical split and guillotine piping system breaks are evaluated up to and including those with a break area equal to twice the cross-sectional area of the largest pipe in the recirculation system piping. As with the location spectrum, the determination of the limiting break size is based on the minimum margin either to PCT or MWR criterion.

<u>MAPLHGR Analyses</u>: After the worst single failure and the location and size of the limiting break have been determined, analyses are undertaken to characterize the maximum steady-state nodal power at which the fuel may be operated prior to the postulated design basis LOCA without exceeding the ECCS limits specified in 10 CFR 50.46.

The condition of the fuel during the blowdown and refill/reflood phases of the LOCA analysis is conservatively based on exposure conditions which assure that the highest value of fuel stored energy is used. The condition of the fuel during the heatup phase of the LOCA analysis is based on fuel conditions associated with planar average exposure. Conservatively high radial peaking factors are assumed in determining the peak axial planar conditions associated with each planar average exposure point.

The blowdown phase is evaluated with RELAX (References 5-10 and 5-11). Refill and reflood phases are evaluated with FLEX (References 5-10 and 5-11). Fuel heatup is analyzed with HUXY (Reference 5-15). Stored energy and fuel characteristics are determined with RODEX2 (Reference 5-13).

Although the determination of MAPLHGR may be an iterative procedure, the number of iterations may be minimized by choosing the initial MAPLHGR values at reasonable levels consistent with the limiting fuel mechanical design analysis. As fission gas release used in the heatup analysis depends on the power history of the fuel, conservative power histories are established for each plant and fuel design.

5.2.1.5 Fuel Handling Accident During Refueling

The introduction of a new mechanical fuel design into a reactor core must be supported by an evaluation of the fuel handling accident for the new fuel design. When required, SPC performs an [ ] of the impact of the new fuel design on the fuel handling accident scenario defined in the reactor's FSAR. Using the boundary conditions and conservative assumptions given in the FSAR and the relevant characteristics of the new fuel design, SPC calculates a conservative number of fuel rods expected to fail as a result of a fuel handling accident with the new fuel design.

The radiological consequences of a fuel handling accident for a new mechanical fuel design are assessed based on the same reactor power history assumed for the assessment of the existing fuel. [

]

5.3 *NRC-Accepted Topical Report References*

The NRC-accepted topical reports for AOO and accident analyses are listed below. Some of these reports were also described in Section 4.3.

5-1: ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

- Purpose: Develop an improved computer program for analyzing BWR system transients.
- SER Conclusions/Restrictions:

  The use of COTRANSA2 is subject to the limitations set forth in the safety evaluations for the methodologies described and approved for XCOBRA-T and COTRAN. The staff reviewed the subject safety evaluations and identified the following limitations that apply to COTRANSA2:

  1. XCOBRA-T was found acceptable for the analysis of only the following licensing basis transients:

     a) Load rejection without bypass

     b) Turbine trip without bypass

     c) Feedwater controller failure

     d) Steam isolation valve closure without direct scram

     e) Loss of feedwater heating or inadvertent HPCI actuation

     f) Flow increase transients from low-power and low-flow operation

  2. Based on a similar limitation on XCOBRA-T thermal-hydraulic modeling, the COTRANSA2 code is not applicable to the analysis of any transient for which lateral flow in a bundle is significant and non-conservative in the calculation of system response.

  3. Based on a similar limitation on XCOBRA-T modeling, for those analyses in which core bypass is modeled, the effect of a computed negative flow in the core bypass region should be shown to make no significant non-conservative contribution in the calculation of system response.

  4. Licensing applications referencing the COTRANSA2 methodology must include confirmation that sensitivity to the time step selection has been considered in the analysis.

Observations: The COTRANSA2 SER restrictions are similar to those for XCOBRA-T (see Reference 4-8). As with XCOBRA-T, COTRANSA2 has been approved for transients for specific plant applications not listed in the SER.

5-2: XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.

- Purpose: Provide an overall methodology for determining a MCPR operating limit. The methodology comprises CHF correlations, fuel hydraulic characteristics, safety limit analyses, AOO analyses, and statistical convolution of uncertainties.

- SER Conclusions/Restrictions: The core monitoring system used (POWERPLEX or other) should be specifically identified in plant submittals referencing the THERMEX methodology.

  Hot channel calculations with COTRANSA were not approved pending review of XN-NF-86-113.

  ENC submitted methodology for application to ENC 8x8 and 9x9 fuel types.

- Observations: Although this topical report references only applications to ENC 8x8 and 9x9 fuel types, the overall methodology is still applicable to SPC 10x10 and new fuel types when appropriate CHF correlations and fuel hydraulic characteristics are implemented within the methodology.

  The review of XN-NF-86-113 was deferred to the review of the topical report for the currently used COTRANSA2 code. (See Reference 4-9)

  Some of the computer codes referenced in the topical report have been superceded by other NRC-approved codes (e.g., COTRANSA with COTRANSA2, XTGBWR with MICROBURN-B) and the XN-3 CHF correlation has been superceded by the NRC-approved ANFB and ANFB-10 CHF correlations. (See References 4-4 and 4-5)

  The SER states "Based on the similarity of the computational models of the two codes (XCOBRA and XCOBRA-T) and the NRC approval of the XCOBRA-T code (Reference 13), we find the use of the steady-state code [XCOBRA] acceptable in this context." XCOBRA continues to be applied for steady-state analyses.

5-3: XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.

- Purpose: Provide a capability to perform analyses of transient heat transfer behavior in BWR assemblies.

- SER Conclusions/Restrictions:

  1. XCOBRA-T was found acceptable for the analysis of only the following licensing basis transients:

     a) Load rejection without bypass

     b) Turbine trip without bypass

c) Feedwater controller failure

d) Steam isolation valve closure without direct scram

e) Loss of feedwater heating or inadvertent HPCI actuation

f) Flow increase transients from low-power and low-flow operation

2. XCOBRA-T analyses that result in any calculated downflow in the bypass region will not be considered valid. (This restriction applies "only to those transients for which modeling of the bypass flow is part of the calculation or negative bypass flow can be shown to be a significant contributing factor in the calculation of the critical heat flux.")

3. "A concern was expressed regarding the comparison of void-profile results calculated with the XCOBRA-T code with experimental bundle data..." This concern was remedied with the submittal and approval of the topical report XN-NF-84-105(P)(A), Volume 1 Supplement 4, "XCOBRA-T: A computer code for BWR Transient Thermal-Hydraulic Core Analysis Void Fraction Model Comparison to Experimental Data."[21]

- Observations: The methodology has been approved for plant specific applications[22-23] to transients other than those listed in the SER.

5-4: ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.

- Purpose: Provide a methodology for the determination of thermal margins.

- SER Conclusions/Restrictions:

1. NRC-approved MICROBURN-B power distribution uncertainties should be used to determine SLMCPR.

2. ANFB additive constant uncertainties should be verified for each plant-specific application.

3. Conservative channel bowing penalty estimates for non-SPC fuel should be used.

4. Channel bowing methodology is not applicable to second-lifetime channels.

- Observations: The critical power methodology is a general methodology which may be used with all SPC developed CHF correlations that include additive constants and additive constant uncertainties.

  Power distribution uncertainties for MICROBURN-B and other SPC core simulator codes approved by the NRC will be used in the CPR methodology.

  As additive constants and additive constant uncertainties are fuel type specific, they do not change for each plant specific application, as noted in SER restriction 2.

5-5: ANF-1358(P)(A) Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Siemens Power Corporation, September 1992.

- Purpose: Develop a generic methodology for evaluating the loss of feedwater heating event.

- SER Conclusions/Restrictions:

  1. The methodology applies to BWR/3, 4, 5, and 6 plant types if the exposure, steam production rate, rated final feedwater temperature, and change in feedwater temperature are within the range covered by the data points presented in the report.

  2. The methodology only applies to the MCPR operating limit for the loss of feed water heating (LWFH) event.

- Observations: The topical report includes results for GE and SPC 8x8 fuel and SPC 9x9 SPC fuel.

5-6: XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, November 1990.

- Purpose: Replace the XFYRE bundle depletion and XTGBWR simulator codes with the updated codes MICBURN-3/CASMO-3G and MICROBURN-B, respectively.

- SER Conclusions/Restrictions:

  1. The currently approved traversing incore probe (TIP) asymmetry uncertainty value of 6.0 percent (See Reference 3-1) should be used in determining the radial bundle power uncertainty.
  2. The application of CASMO-3G/MICROBURN-B to fuel designs that differ significantly from those included in the [report] data base should be supported by additional code validation to ensure that the methodology and uncertainties are applicable.

- Observations: CASMO-3G and MICROBURN-B were incorporated into the methodologies described in Reference 3-1.

  Nuclear designs are limited to the uncontrolled local peaking factors of the approved critical power correlation.

  Application to fuel designs that differ significantly from those in the Supplement 3 data base to be supported by additional code validation to ensure uncertainties remain applicable. (This is addressed generically when new design types are introduced - e.g., 11x11 fuel, etc.)

5-7: XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983.

- Purpose: Development of BWR core analysis methodology which comprises codes for fuel neutronic parameters and assembly burnup calculations, reactor core simulation, diffusion theory calculations, core and channel hydrodynamic stability predictions, and producing input for nuclear plant transient analysis. Procedures for applying the codes for control rod drop, control rod withdrawal and fuel misloading events have been established.

- SER Conclusions/Restrictions: No specific restrictions were given in the SER, but a recommendation was "...made that the analytical models be continuously verified to insure their applicability."

- Observations: Portions of this topical report have been superceded by subsequently approved codes or methodologies. Superceded and currently applicable portions are identified below:

  Superceded Portions:

  Fuel Assembly Depletion Model - XFYRE replaced with CASMO-3G (See Reference 3-2.)

  Core Simulator - XTGBWR replaced with MICROBURN-B (See Reference 3-2.)

  Diffusion Theory Model - XDT replaced with CASMO-3G (See Reference 3-2.)

  Stability Analysis - COTRAN replaced with STAIF (See Reference 3-4.)

  Control Rod Withdrawal - XTGBWR replaced with MICROBURN-B (See Reference 3-2.)

  Fuel Misloading Analysis – XFYRE is replaced with CASMO-3G and XTGBWR is replaced with MICROBURN-B. These analyses are now performed to verify that the offsite dose due to such events does not exceed a small fraction of 10 CFR 100 guidelines as described and approved in Reference 3-3.

  Applicable Portions:

  Control Rod Drop Accident - This analysis is performed using COTRAN.

  Control Rod Withdrawal – This analysis is the same as that used to determine the change in CPR (ΔCPR) for error rod patterns, but with an additional procedure. The additional procedure evaluates the number of fuel rods in boiling transition (BT) to determine that a specific error rod pattern does not challenge the criterion that < 0.1% rods are in BT at the MCPR safety limit, assuming failures of the rod block.

  Neutronic Reactivity Parameters - These parameters are determined as described in the topical report but using the most recently approved codes.

Void Reactivity Coefficient - Method used to calculate core reactivity coefficient is the same but MICROBURN-B is used instead of XTGBWR.

Doppler Reactivity Coefficient - Method used to calculate the core average Doppler coefficient is the same but CASMO-3G is used instead of XFYRE.

Scram Reactivity - Method used is the same MICROBURN-B is used instead of XTGBWR.

Delayed Neutron Fraction - Calculated using CASMO-3G instead of XFYRE.

Prompt Neutron Lifetime - Calculated using CASMO-3G instead of XFYRE.

5-8: XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$," Exxon Nuclear Company, May 1986.

- Purpose: Modify approved control rod withdrawal error transient methodology (Reference 5-7) for application to BWR/6s or other BWRs with ganged control rods.
- SER Conclusions/Restrictions:

  1. The methodology "assumes the presence of the technical specification on rod withdrawal limits as a function of power which have been established by the reactor designer (General Electric)."
  2. The methodology and results are valid for operation within the power flow domain illustrated in Figure 4.1 of the topical report and for the fuel management scheme used for determining the operating states of the data base. Use of other power-flow domains (e.g., the MEOD) or other fuel management schemes (e.g., the single rod sequence loading pattern) will require verification by analysis that the conclusions of this report are valid.
  3. Cycle specific analyses are not required if the operating power-flow region is bounded by that presented in the topical report and the core loading pattern and control rod patterns are consistent with the data base used.

- Observations: The original methodology was developed using the XTGBWR core simulator code which was superceded by MICROBURN-B (see References 3-2 and 5-6) is still applicable.

5-9: XN-NF-825(P)(A) Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain," Exxon Nuclear Company, October 1986.

- Purpose: Extend the applicability of the licensing topical report above (Reference 5-8) to control rod withdrawal error transients for BWR/6 plants within the extended operating domain.

- SER Conclusions/Restrictions:

  1. The methodology "assumes the presence of the technical specification on rod withdrawal limits as a function of power which have been established by the reactor designer (General Electric)."

  2. The methodology and results are valid for operation within the power flow domain illustrated in Figure 3.1 of the topical report and for the fuel management scheme used for determining the operating states of the data base for the MEOD. Other fuel management schemes will require verification by analysis that the conclusions of this report are valid.

  3. Cycle specific analyses are not required if the operating power-flow region is bounded by that presented in the topical report and the core loading pattern and control rod patterns are consistent with the data base used.

- Observations: The original methodology was developed using the XTGBWR core simulator code which was superceded with MICROBURN-B (see References 3-2 or 5-6) is still applicable.

5-10: XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.

- Purpose: Provide an evaluation model methodology for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology was developed to comply with 10 CFR 50.46 and 10 CFR 50 Appendix K criteria.

- SER Conclusions/Restrictions:

  1. Methodology conforms to 10 CFR 50 Appendix K.

  2. Follow up verification of jet pump model with new data is required and must be assessed by the NRC.

  3. Counter-current flow limit correlation coefficients used in FLEX for new fuel designs that vary from fuel cooling test facility (FCTF) measured test configurations must be justified.

  4. Methodology adequate for large and small break LOCA analyses.

- Observations: RELAX and FLEX, which are key computer codes in the methodology, have been subsequently modified as described in References 5-11 and 5-12.

5-11: ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993.

- Purpose: Update the RELAX system blowdown code and FLEX refill code by reducing code instabilities and improving their predictive capabilities.

- SER Conclusions/Restrictions:

  1. The revised model is valid within the range of applicability of the modified Dougall-Rohsenow heat transfer correlation.

  2. The phase separation models will be limited to the models used in the topical report.

  3. The revised evaluation model will be limited to jet pump plant applications.

- Observations: The RELAX, with the jet pump update from ANF-91-048(P)(A) Supplement 1 and 2, and FLEX models are currently applicable.

5-12: ANF-91-048(P)(A) Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997.

- Purpose: Modify the jet pump model in the RELAX blowdown code to better predict jet pump performance for all ranges of LBLOCA and SBLOCA conditions.

- SER Conclusions/Restrictions: No specific restrictions imposed.

- Observations: The jet pump model is currently applicable.

5-13: XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.

- Purpose: Provide an analytical capability to predict BWR and PWR fuel thermal and mechanical conditions for normal core operation and to establish initial conditions for power ramping, non-LOCA and LOCA analyses.

- SER Conclusions/Restrictions: A physically-based gas release model shall be used.

- Observations: RODEX2 is the fuel performance code that provides input to BWR LOCA and transient thermal-hydraulic methodologies.

  RODEX2 was approved for use up to an exposure limit of 62,000 MWd/MTU rod-average burnup for BWR LOCA and transient applications (see Reference 2-4).

  RODEX2 may be used to model fuel with up to 8% gadolinia loading (See Reference 2-10).

5-14: XN-CC-33(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975.

- Purpose: Develop a planar heat transfer model which includes rod-to-rod radiation. This code also includes the BULGEX model for the calculation of rod strains and ballooning.

- SER Conclusions/Restrictions:

  1. A value less than 1.0 for the fraction of locally generated gamma energy deposited in the fuel pin needs to be justified.

2. For zirconium-water reaction the initial oxide thickness used should be no larger than can be reasonably justified, including consideration of the effects of manufacturing processes, hot-functional testing, and exposure.

3. Appendix K spray heat transfer coefficients for 7x7 assemblies were accepted with a 10% reduction in value.

4. The fission power curve (i.e., scram time and decrease due to voiding, if any) for small and intermediate size break cases will be justified

5. Plant specific calculations are to demonstrate that the plane-of-interest (POI) assumed is the plane at which peak cladding temperature occurs.

- Observations:

  a) SER restriction 4 noted above was superceded by the cladding swelling and rupture methodology approved in the topical report (Reference 5-15).

  b) SPC maximizes peak cladding temperature by assuming no initial oxide thickness is present at the initiation of a LOCA for zirconium-water reaction calculations.

5-15: XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.

- Purpose: Incorporate the swelling and rupture models described in NUREG-0630[10] which comply with 10 CFR 50 Appendix K requirements into the HUXY code (Reference 5-14).

- SER Conclusions/Restrictions: No specific restrictions.

- Observations: The swelling and rupture model is currently applicable.

5-16: ANF-CC-33(P)(A) Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option," Advanced Nuclear Fuels Corporation, February 1991.

- Purpose: Justify conservative spray heat transfer coefficients for 9x9 fuel with internal water canister.

- SER Conclusions/Restrictions: The application of Appendix K 7x7 spray heat transfer coefficients is restricted to 9x9 fuel with an internal canister.

- Observations: Spray heat transfer coefficients are still applicable for 9x9 fuel with a water channel.

5-17: XN-NF-929(P)(A) and Supplements 1 through 4, "Spray Heat Transfer Coefficients for Jet Pump BWR Fuel Assemblies with Water Rods," Advanced Nuclear Fuels Corporation, March 1992.

- Purpose: Justify experimentally measured spray heat transfer coefficients for 9x9 fuel with water rods.

- SER Conclusions/Restrictions:

  1. The proposed convective heat transfer coefficients can be used only in the evaluation of the ANF-9x9 fuel rod array geometry with the upper tie plate configuration described in the topical report.

  2. Additional supporting information must be provided to justify the continued use of the proposed coefficients if applications occur such that the assumptions, or boundary conditions for the tests and supporting analytical computations described in the topical report do not bound the coolant conditions calculated by the ANF-approved emergency core cooling system model.

  3. The proposed convective heat transfer coefficients can be used only for those plants with rod power levels and axial power shapes bounded by the top-peaked and bottom-peaked power distributions presented in the topical report. Otherwise, additional justification is needed to support the continued use of the proposed heat transfer coefficients for BWR ECCS licensing analyses of the ANF 9x9 rod bundle array.

- Observations: Spray heat transfer coefficients are still applicable for 9x9 fuel with water channels.

6.0 Criticality Safety Analysis

In addition to reactor systems safety analyses, SPC performs criticality analyses for new and spent fuel storage. These analyses are described in this section.

SPC performs criticality safety analyses of new fuel storage vaults and spent fuel storage pools. Storage array k-eff calculations are performed with the KENO.Va Monte Carlo code, which is part of the SCALE 4.2 Modular Code System.[26] The CASMO-3G bundle depletion code (Reference 3-2) is used to calculate k_∞ values for fuel assemblies at beginning of life (new fuel storage) or as a function of exposure, void, and moderator temperature for both incore and in-rack (spent fuel storage) geometries.

The KENO.Va and the CASMO-3G computer codes are widely used throughout the nuclear industry. They are used primarily for criticality safety and core physics calculations, respectively. SPC has broad experience in the use of both of these codes. KENO.Va has been benchmarked by SPC against critical experiment data to define appropriate reactivity biases.

SPC uses criteria given in plant technical specifications and specific references[27-31] in Section 7.0 to assess the acceptability of the criticality safety analyses.

7.0 References

1. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

2. EMF-1(A), "Quality Assurance Program for Nuclear Fuels, Services, and Packaging and Transportation of Radioactive Materials," Siemens Power Corporation.

3. "General Design Criteria for Nuclear Power Plants," Code of Federal Regulations, Title 10 "Energy," Part 50, Appendix A.

4. "Reactor Site Criteria," Code of Federal Regulations, Title 10 "Energy," Part 100.

5. "Domestic Licensing of Production and Utilization Facilities," Code of Federal Regulations, Title 10 "Energy," Part 50.

6. "Rules for Construction of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code, Section III, 1977.

7. Swanson Analysis System, "ANSYS-Engineering Analysis System Theoretical Manual," 1977, and "ANSYS-User's Guide," 1979.

8. W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nuc. Sci. Eng., 1964, 20:1.

9. JN-72-23, Revision 1, "Cladding Collapse Calculation Procedure," Jersey Nuclear Company, Inc., November 1972.

10. NUREG-0630, Cladding Swelling and Rupture Models for LOCA Analysis, U.S. Nuclear Regulatory Commission, April 1980.

11. Regulatory Guide 1.77, Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, U.S. Atomic Energy Commission, Washington, D.C., May 1974.

12. XN-NF-81-21(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1982.

13. XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.

14. XN-NF-82-06(P)(A) Supplement 1 Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.

15. ANF-88-152(P)(A) and Amendment 1 and Supplement 1, "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel," Advanced Nuclear Fuels Corporation, November 1990.

16. ANF-89-014(P)(A) Revision 1 and Supplements 1 and 2, "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X Reload Fuel," Advanced Nuclear Fuels Corporation, October 1991.

17. XN-NF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Exxon Nuclear Company, August 1986.

18. NASA SP-221, The NASTRAN Theoretical Manual, 1969.

19. Fission Gas Release – Letter, R. A. Copeland (Siemens Nuclear Power) to R. C. Jones (NRC), "No Subject," RAC:050:91, May 13, 1992. Thermal Conductivity – XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981.

20. Letter, Cynthia A. Carpenter (NRC) to James F. Mallay (SPC), "Modifications to Procedures for Use of XCOBRA-T," June 10, 1999.

21. XN-NF-84-105(P)(A) Volume 1 Supplement 4, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis - Void Fraction Model Comparison to Experimental Data," Advanced Nuclear Fuels Corporation, June 1988.

22. SER for Amendment No. 102 to Facility Operating License No. NPF-14: Susquehanna Steam Electric Station Unit No. 1.

23. SER for Amendment No. 73 to Facility Operating License No. NPF-29: Grand Gulf Nuclear Station Unit 1.

24. Regulatory Guide 1.3, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," USNRC, June 1974.

25. "Regulatory Guide 1.25, "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," U.S. Nuclear Regulatory Commission, March 1972.

26. A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, SCALE 4.2, Oak Ridge National Laboratory, revised December 1993.

27. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 9.1.1 (New Fuel Storage), U.S. Nuclear Regulatory Commission, July 1981.

28. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 9.1.2 (Spent Fuel Storage), U.S. Nuclear Regulatory Commission, July 1981.

29. Spent Fuel Storage Facility Design Basis, Regulatory Guide 1.13, Proposed Revision 2, U.S. Nuclear Regulatory Commission, December 1981.

30. Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, ANSI/ANS American National Standard 57.2-1983, American Nuclear Society, October 1983.

31. Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors, ANSI/ANS American National Standard 8.17-1984, American Nuclear Society, January 1984.

Attachment A Table A-1 Major Methodology, Parameters, and Analyses Interfaces

Table A-1 Major Methodology, Parameters, and Analyses Interfaces

Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	$15.4.9A^D$	15.6.5	$15.7.4^D$
ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.															
• Mechanical, Nuclear, & Thermal-Hydraulic Criteria	A	A	A												
XN-NF-85-67(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.															
• Method for LHGR Limit Curves	A														
XN-NF-82-06(P)(A) Supplement 1 Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.															
• Thermal Limits Effects Due to Rod Bow	A		A												
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.															
• Gas Pressure	A														

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	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
• Cladding Oxidation	A														
• Steady State Cladding Strain	A														
• Fuel Melt Analyses	A														
• Fuel Rod, Assembly, Channel, and Water Channel Growth	A														
XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.															
• Transient Cladding Strain	A														
• Fuel Rod Characteristics (e.g. Gap Conductance)	A		A	A	A			A	A	A		A		A	
XN-NF-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)															
• Rod Bow Method	A		A												

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Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.															
• Method for Collapse, Pressure and Gap Limits	A														
XN-NF-81-51(P)(A), LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, May 1986.															
• Method for Determining Seismic Stresses	A														
XN-NF-84-97(P)(A), "LOCA-Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, August 1986.															
• Extension of Seismic Method to Other Fuel Designs	A														
XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.															
• Extended Fuel Properties to 8% Gadolinia	A		A	A	A	A	A	A	A	A	A			A	

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Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
EMF-93-177(P)(A) and Supplement 1, "Mechanical Design for BWR Fuel Channels," Siemens Power Corporation, August 1995.															
• Channel Bow and Bulge; Stress, Corrosion, Fatigue, and Seismic-LOCA Breaks	A														
ANF-90-82(P)(A) Revision 1 and Revision 1 and Supplement 1 , "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.															
• Method for Justifying Replacement Inert Rods	A	A	A												
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analyses," Exxon Nuclear Company, March 1983.															
• Margin to SLMCPR (<0.1% rods in BT)							A	A							
• Deposited Enthalpy, 280 cal/gm												A			

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	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, November 1990.															
• Power Distribution		A	A	A	A	A	A	A	A	A	A	A		A	
• Cross Section and Reactivity Input		A		A	A	A	A	A	A	A	A	A		A	
XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.															
• < 0.1% Rods in Boiling Transition											A				
EMF-CC-074(P)(A) Volume 1, "STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain – Code Qualification Report," Siemens Power Corporation, July 1994.															
• Thermal-Hydraulic – Reactivity Stability		A	A												

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Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.															
• Methodology for OLMCPR Determination			A	A	A	A	A	A	A						
XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983.															
• Assembly 2-Phase Pressure Drop			A	A	A	A	A	A	A	A				A	
ANF-1125(P)(A) and Supplements 1 and 2, "ANFB Critical Power Correlation," Advanced Nuclear Fuels Corporation, April 1990.															
• Critical Power Correlation			A	A	A	A	A	A	A	A	A	A			
EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," Siemens Power Corporation, July 1998.															
• Critical Power Correlation			A	A	A	A	A	A	A	A	A	A			

Table A-1 Major Methodology, Parameters, and Analyses Interfaces

Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
EMF-1997(P), Supplement 1 (P)(A), Revision 0, "ANFB-10 Critical Power Correlation: High Local Peaking Results," Siemens Power Corporation, July 1998.															
• Correlation Additive Constant Uncertainties			A	A	A	A	A	A	A	A	A	A			
EMF-1125(P)(A) Supplement 1 Appendix C, "ANFB Critical Power Correlation Application for Co-Resident Fuel, " Siemens Power Corporation, August 1997.															
• Co-resident MCPR			A	A	A	A	A	A	A	A	A	A			
XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.															
• Transient Assembly CPR Determination				A	A	A		A	A	P					
ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.															
• System Transient Response				A	A	A		A	A	P					

Table A-1 Major Methodology, Parameters, and Analyses Interfaces

Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation, November 1990.															
• Determination of SLMCPR				A	A	A	A	A	A						
• Rods in Boiling Transition										P	A				
ANF-1125(P)(A) Supplement 1, Appendix E, "ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties," Siemens Power Corporation, September 1998.															
• Correlation Additive Constant Uncertainties				A	A	A	A	A	A	A	A				
ANF-1358(P)(A) Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Siemens Power Corporation, September 1992.															
• Determination of OLMCPR				A											
XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, MCPRp," Exxon Nuclear Company, May 1986.															
• Margin to SLMCPR (<0.1% rods in BT)							A								

Table A-1 Major Methodology, Parameters, and Analyses Interfaces

Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1- 15.1.2	15.2.1- 15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
XN-NF-825(P)(A) Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, MCPRp for Plant Operations within the Extended Operating Domain," Exxon Nuclear Company, October 1986.															
• Margin to SLMCPR (<0.1% rods in BT)							A								
XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.															
• ECCS Evaluation Methodology														A	
ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993.															
• Update to ECCS Evaluation Methodology														A	
ANF-91-048(P)(A) Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997.															
• Improved Jet Pump Model														A	

Table A-1 Major Methodology, Parameters, and Analyses Interfaces

Methodology	STANDARD REVIEW PLAN														
	FUEL DESIGN	NUCLEAR DESIGN	THERMAL - HYDRAULIC DESIGN	Anticipated Operational Occurrences						Postulated Accidents					
	4.2	4.3	4.4	15.1.1-15.1.2	15.2.1-15.2.2	15.2.4	15.4.2	15.4.5	15.5.1	15.3.3	15.4.7	15.4.9	15.4.9A[D]	15.6.5	15.7.4[D]
XN-CC-33(P)(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975.															
• Prediction of 10 CFR 50.46 Criteria														A	
XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.															
• NUREG-0630 Rupture and Swelling Models														A	
ANF-CC-33(P)(A) Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option," Advanced Nuclear Fuels Corporation, February 1991.															
• Application of Appendix K Spray Heat Transfer Coeff.														A	
XN-NF-929(P)(A) and Supplements 1 through 4, "Spray Heat Transfer Coefficients for Jet Pump BWR Fuel Assemblies with Water Rods," Advanced Nuclear Fuels Corporation, March 1992.															
• 9x9 Spray Heat Transfer Coefficients														A	

A – Approved for generic applications.
P – Approved for plant-specific applications.
D – Dispositioned using previously reported results or analyses performed using NRC and industry accepted codes (e.g., ORIGEN)

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