AREVA NP, Inc. Topical Report ANP-2637, Revision 1,
Boiling Water Reactor Licensing Methodology Compendium,
June 2007

ANP-2637
Revision 1

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Abstract

This report is a compendium of AREVA methodologies and design criteria that 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 AREVA 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 FSAR.

Nature of Changes

Item	Paragraph or Page(s)	Description and Justification
1.	2-22, 2-25	Revised "Observations" discussion to include application to ATRIUM-9 and ATRIUM-10 designs.
2.	2-25, 3-7, 5-21, and 5-22	Revised "Clarifications" discussion to add a description of the issue being clarified.
3.	2-27	Removed "Clarifications" discussion associated with References 27, 28, and 29 since it is already discussed in Implementation of SER Restrictions Item 4.
4.	2-30	Added "Clarifications" associated with exposure limits for full-length and part-length rods.
5.	4-7	Revised "Observations" to include discussion of Reference 2-10.
6.	4-8	Added "Observations" note for SER restriction 2.
7.	7-2	Removed Reference 28 and adjusted other references accordingly.
8.	7-3	Added References 37, 38, and 39.

Contents

Tables

This document contains a total of 101 pages.

Nomenclature

ANF	Advanced Nuclear Fuels
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BOC	Beginning-of-Cycle
BWR	Boiling Water Reactors
CHF	Critical Heat Flux
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRDA	Control Rod Drop Accident
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
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
$LHGRFAC_f$	Flow Dependent LHGR Multiplier
$LHGRFAC_p$	Power Dependent LHGR Multiplier
LOCA	Loss-of-Coolant Accident
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
$MCPR_f$	Flow-Dependent Minimum Critical Power Ratio
$MCPR_p$	Power-Dependent Minimum Critical Power Ratio
MEOD	Maximum Extended Operating Domain
MSIV	Main Steam Isolation Valve
MWR	Metal-Water Reaction
NRC	Nuclear Regulatory Commission, U.S.
OLMCPR	Operating Limit MCPR

PA	Postulated Accident
PAPT	Protection Against the Power Transient
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
RIA	Reactivity Initiated Accident
RPS	Recirculation Pump Seizure
SAFDL	Specified Acceptable Fuel Design Limit
SER	Safety Evaluation Report
SLMCPR	Safety Limit Minimum Critical Power Ratio
SRP	Standard Review Plan
TER	Technical Evaluation Report

1.0 Introduction

This report is a compendium of AREVA NP Inc.* (AREVA) 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. Compliance with the SER restrictions is assured by implementing them within the engineering guidelines or by incorporating them into the computer codes.

The methods and topical reports addressed herein are designed to give BWR licensees using AREVA 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 are used to demonstrate that AREVA fuel is compatible with co-resident fuel.

The organization of this report parallels the major sections of the Standard Review Plan (SRP) (Reference 1) 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 AREVA BWR methodologies. Table 1-2 provides a list of topical reports that are used by AREVA to support operation of BWRs. Table 1-2 also 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.

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There are two styles for 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 (see Table 1-2 for a list of References).” Other supporting references found in Section 7.0 are cited by the reference number.

Table 1-1 SRP No. Addressed by AREVA 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 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow (AOO)
15.2.1 – 15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed) (AOO)
15.2.7	Loss of Normal Feedwater Flow (AOO)
15.3.1 – 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions (AOO)
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (Postulated Accident (PA))
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power (AOO)
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate (AOO)
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (PA)
15.4.9	Spectrum of Rod Drop Accidents (BWR) (PA)
15.4.9a	Radiological Consequences or Rod Drop Accident (BWR) (PA)
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory (AOO)
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve (AOO)

Table 1-1 SRP No. Addressed by AREVA Methodologies *(Continued)*

SRP No.	Chapter 15 Accident Analysis *(Continued)*
15.6.5	Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary (PA)
15.7.4	Radiological Consequences of Fuel Handling Accidents (PA)
15.8	Anticipated Transients without Scram

Table 1-2 Reference Index

Reference No.	Methodology	Page No.(s)	Referenced in Core Operating Limits Report
2-1	XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981.	2-17	
2-2	XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)	2-18	
2-3	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.	2-19	yes
2-4	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.	2-20	
2-5	XN-NF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Exxon Nuclear Company, August 1986.	2-21	
2-6	XN-NF-85-67(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.	2-22	yes
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.	2-23	
2-8	XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.	2-24	
2-9	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.	2-25	
2-10	ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.	2-26	yes
2-11	ANF-90-82(P)(A) Revision 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995.	2-28	
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.	2-29	yes
2-13	EMF-93-177(P)(A) Revision 1, "Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005.	2-31	

Table 1-2 Reference Index *(Continued)*

Reference No.	Methodology	Page No.(s)	Referenced in Core Operating Limits Report
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.	3-5	yes
3-2	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.	3-7	yes
3-3	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.	3-8	
3-4	EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation, October 1999.	3-10	yes
3-5	EMF-CC-074(P)(A) Volume 4, Revision 0, "BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.	3-12	yes
4-1	XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983.	4-6	
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.	4-7	yes
4-3	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.	4-8	yes
4-4	EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.	4-9	yes
4-5	EMF-2209(P)(A) Revision 2, "SPCB Critical Power Correlation," Framatome ANP, September 2003.	4-10	yes

Table 1-2 Reference Index *(Continued)*

Reference No.(s)	Methodology	Page No.(s)	References in Core Operating Limits Report
5-1	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.	5-14	
5-2	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.	5-16	
5-3	XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.	5-17	
5-4	XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$," Exxon Nuclear Company, May 1986.	5-18	
5-5	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.	5-19	yes, for BWR/6
5-6	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.	5-20	yes
5-7	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.	5-22	yes
5-8	ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993.	5-24	
5-9	ANF-91-048(P)(A) Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997.	5-25	
5-10	EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.	5-26	yes
5-11	EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.	5-27	yes
5-12	ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.	5-28	yes

2.0 Fuel System Design

AREVA 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 (AOOs). 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 AREVA 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 2-10, are satisfied by AREVA design criteria approved by the NRC, which include:

- Preparing controlled documentation of the fuel system description and fuel assembly design drawings.
- Performing analyses with NRC-approved and accepted models and methods for AREVA 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 AREVA's approved QA procedures, QC inspection program, and design control requirements identified in FQM Revision 2 (Reference 2).

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 operation 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 (Reference 3) (GDC) 10, 27, and 35; 10 CFR Part 100, (Reference 4) and 10 CFR Part 50 (Reference 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 operation and 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 based on the fuel assembly weight multiplied by a load factor.

- 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, channels). The stress categories and strength theory for austenitic stainless steel presented in the ASME Boiler and Pressure Vessel Code, Section III (Reference 6) are used as a general guide.

- Steady state stress design limits are given in Table 3-1 of Reference 2-10. 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 and Pressure Vessel Code, Section III, Article III-2000 (Reference 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 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 2-10 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 (Reference 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 AREVA analysis methods for calculating fuel rod cladding and assembly steady-state stresses are discussed and approved in References 2-6 and 2-9. The methods for calculating fuel channel stresses are discussed and approved in Reference 2-13.

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. The strain limit is reduced at higher exposures to account for lower ductility.

Bases

The design criteria for cladding strain are intended to preclude excessive cladding deformation and failure from normal operations and AOOs. AREVA uses the NRC-approved RODEX2A code (References 2-5 and 2-12) to calculate steady-state cladding strain during normal operation. Transient cladding strain is calculated as described in Supplement 1 of Reference 2-3.

2.2.3 Strain Fatigue

Design Criteria

The design criteria for strain fatigue limits the cumulative fatigue usage factor based on a defined design fatigue life.

Bases

Cycle loading associated with relatively large changes in power can cause cumulative damage, which may eventually lead to fatigue failure. Therefore, AREVA requires that the cladding not exceed the fatigue usage design life as reduced by a proprietary factor. 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 AREVA methodology for determining fuel assembly strain fatigue is based on Supplement 1 of Reference 2-3 and the O'Donnell and Langer fatigue design curves (Reference 8). The fatigue curves have been adjusted to incorporate the recommended safety factor of two on stress amplitude or 20 on number of cycles, whichever is more conservative. The RODEX2 code is used to provide fuel rod stress conditions for AREVA fatigue analysis.

Fuel channel fatigue is evaluated with finite element calculations to evaluate channel stresses due to pressure variations in the channel as a function of bundle power and flow (Reference 2-13). The same O'Donnell and Langer fatigue design curve is used as for the fuel rod evaluations.

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

AREVA 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. AREVA performs fretting tests to verify consistent fretting performance for new spacer designs. Examination of a large number of irradiated BWR rods, fuel assemblies, and channels has substantiated the absence of fretting in AREVA designs.

2.2.5 Oxidation and Crud Buildup

Design Criteria

There is no specific limit for oxide thickness or crud buildup. The effects of oxidation and crud buildup are considered in the fuel rod thermal and internal gas pressure analyses.

Bases

The AREVA 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 no specific limit for crud buildup. However, the BWR fuel performance code RODEX2A (Reference 2-12) 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. AREVA 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 (EOL) oxidation thickness that is equivalent to the maximum peak oxidation observed for AREVA BWR fuel.

AREVA 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 insignificant since cladding deformation is strain 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 EOL stress analysis, the wall thickness is reduced consistent with the peak oxide thickness.

2.2.6 Rod Bowing

Design Criteria

The AREVA design criteria for rod bowing is that lateral displacement of the 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. AREVA uses an approved methodology (Reference 2-9) 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 2-2) to calculate minimum EOL rod to rod spacing. The potential effect of this rod bow on 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

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

Bases

AREVA 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 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, where the water channel connects the bottom and top tie plates, the goal is to ensure adequate clearance for growth of the fuel rods.

The analysis method (Reference 2-9) for evaluating rod-to-tie plate engagement is based on a statistical upper bound of measured differential rod-to-tie plate growth data (Reference 2-12) for 9x9 and 10x10 designs. The correlation predicts differential growth that bounds the differential growth data with a given statistical tolerance. This analysis uses fabrication tolerances in order to maintain conservatism in the calculated initial engagements and clearances.

2.2.8 Rod Internal Pressure

Design Criteria

AREVA limits maximum fuel rod internal pressure relative to system pressure. In addition, AREVA requires that when fuel rod pressure exceeds system pressure, the pellet-clad gap has to remain closed if it is already closed or that it should not tend to open for steady state or increasing 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 AREVA in Reference 2-7.

2.2.9 Fuel Assembly Liftoff

Design Criteria

AREVA 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 criterion 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

AREVA uses either a stress analysis or testing to demonstrate compliance. The analysis or test uses an axial load factor on 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 buckling load in the case of the 9x9 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 criterion 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 ATRIUM™-10* design as there is only one large spring 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

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between the channel and the fuel assembly. 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, AOOs, and postulated accidents. When the fuel rod failure criteria are applied in normal operation and AOOs, they are used as limits (SAFDLs) since fuel failure under those conditions must not occur according to GDC 10 (Reference 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 (Reference 4).

2.2.12.1 Internal Hydriding

Design Criteria

AREVA limits internal hydriding by imposing a fabrication limit for total hydrogen in the fuel pellets.

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 are 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 AREVA 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 AREVA fuel system design by eliminating the formation of significant axial gaps. The evaluation must show that the pellet column is compact at a specified burnup. 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. AREVA fuel is designed to minimize the potential for the formation of axial gaps in the fuel and to minimize clad creepdown that would prevent the closure of axial gaps or allow creep collapse.

The RODEX2A code (Reference 2-12) is used to provide initial in-reactor fuel rod conditions to the COLAPX (Reference 9) method described in Reference 2-7 which is used to predict creep collapse. 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 criterion 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 operation 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.

AREVA uses the RODEX2A computer code (Reference 2-12) 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 and accidents, AREVA also uses the RODEX2A code to calculate maximum possible fuel centerline temperatures at LHGRs that are higher than the steady-state LHGR history used for normal operation.

2.2.12.5 Pellet/Cladding Interaction

Design Criteria

The Standard Review Plan (Reference 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 (Reference 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 loss-of-coolant accident (LOCA). Since there are no specific design criteria in the Standard Review Plan (Reference 1) associated with cladding rupture, AREVA 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 AREVA ECCS evaluation model. The cladding ballooning and rupture models used are those presented in NUREG-0630 (Reference 10) for cladding rupture evaluation. These models are described in XN-NF-82-07(P)(A) Revision 1 (see Reference 5-3).

2.2.12.7 Fuel Rod Mechanical Fracture

Design Criteria

AREVA limits the combined stresses from postulated accidents to the stresses given in the ASME Code, Section III, Appendix F (Reference 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 AREVA BWR reload fuel are presented in Reference 2-4, which describes AREVA's LOCA-seismic structural response analysis. The design basis is that the channeled fuel assemblies must withstand external loads due to earthquake and postulated pipe breaks without fracturing 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 Reference 2-4. 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. Consideration can be given to the fuel assembly dynamic properties in determining the need for reanalysis when the fuel design is changed. AREVA 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

AREVA uses the NRC reviewed and accepted densification and swelling models in the fuel performance code, RODEX2A (Reference 2-12) and RODEX2 (Reference 2-3).

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 (Reference 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 (Reference 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 10 CFR 50 Appendix K (Reference 5) (see Reference 5-1). LOCA analyses are performed on a plant specific basis.

2.2.13.2 Violent Expulsion of Fuel

Design Criteria

AREVA 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 AREVA methodology complies with the fission product source term guideline in Regulatory Guide 1.77 (Reference 11) and the Standard Review Plan (Reference 1) that restricts the radially-averaged energy deposition.

The limiting RIA for AREVA fuel in a BWR is the control rod drop accident (CRDA). AREVA 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 3-1. The parameterized AREVA 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 10 CFR 50 Appendix K (Reference 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 AREVA ECCS evaluation model. The cladding ballooning and rupture models used are those presented in NUREG-0630 (Reference 10) for cladding rupture evaluation. These models are described in XN-NF-82-07(P)(A) Revision 1 (see Reference 5-3).

The RODEX2 fuel performance code (Reference 2-3) 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 AREVA design criteria for fuel assembly structural damage from external forces are discussed in Sections 2.2.1, 2.2.9, and 2.2.12.7.

Bases

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. The Standard Review Plan (Reference 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 AREVA 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. AREVA ensures these design bases are met by placing ASME design limits on the stresses that the fuel channel and critical fuel assembly components can experience. These limits have been approved for AREVA fuel assemblies in References 2-4 and 2-13.

2.3 *NRC-Accepted Topical Report References*

The NRC has approved the following licensing topical reports that describe the methods and assumptions used by AREVA 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 in the following sections.

2-1 : XN-NF-79-56(P)(A) Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981.

- Purpose: Justify gadolinia fuel properties for up to 5 wt% gadolinia loading in uranium dioxide fuel.
- SER Restrictions:
  1. The concentration of gadolinia is limited to 5 wt%.
  2. The report is acceptable based on a commitment to acquire more data for gadolinia bearing rods.
- Implementation of SER Restrictions:
  1. This SER restriction is no longer applicable. The limit on gadolinia concentration was increased to 8 wt% in Reference 2-8.
  2. The additional data was gathered and was provided to the NRC in Reference 14.
- Observations: The limitation on the concentration of gadolinia was raised to 8 wt% by the topical report XN-NF-85-92(P)(A). Additional data was gathered on gadolinia from Prairie Island, Tihange, and other reactors.

2-2 : XN-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 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 continue fuel surveillance to ensure confidence in the assumptions and bases.
- Implementation of SER Restrictions: The application of the rod bow model to higher burnup and other fuel designs was approved in Reference 2-9.
- Observations: AREVA has continued to gather data from fuel surveillance and CPR experiments.

2-3 : 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 Restrictions:

  1. The NRC concluded that the RODEX2 fission gas release model was acceptable to burnups up to 60 MWd/KgU. This implies a burnup limit of 60 MWd/KgU (nodal basis).
  2. The creep correlation accepted by the NRC is the one with the designation MTYPE = 0.

- Implementation of SER Restrictions:

  1. This restriction no longer applies. The exposure limits for BWR fuel were increased to 54 MWd/kgU for an assembly and to 62 MWd/kgU for a rod in Reference 2-12. These exposure limits are reflected in engineering guidelines.
  2. This restriction is implemented in the engineering guidelines and through computer code controls (defaults, override warning messages).

- Observations: The computer code that is used to perform analyses is now called RODEX2-2A. The NRC approved models, RODEX2 or RODEX2A, are chosen by input. A single code is maintained in order to assure that the NRC approved models are implemented correctly. RODEX2 is the fuel performance code that provides input to BWR LOCA and transient thermal-hydraulic methodologies.

  RODEX2 and RODEX2A may be used to model fuel with up to 8% gadolinia loading (See Reference 2-8).

2-4 : 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 fuel assemblies.

- SER 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.

- Implementation of SER Restrictions: This restriction is no longer applicable. The requirements for fuel channels are now described in Reference 2-13.

- 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 (Reference 13). Current analyses make use of the KWUSTOSS dynamic analysis code for fuel channels (with fuel assembly) as described in Reference 2-13. The LOCA seismic criteria are specified in Reference 2-10.

2-5 : XN-NF-85-74(P)(A) Revision 0, "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model" Exxon Nuclear Company, August 1986.

- Purpose: The purpose of this topical report was to obtain NRC approval of a modification of the RODEX2 (Reference 2-3) fission gas release model for application to BWRs. This code version was named RODEX2A.

- SER Conclusions / Restrictions:

  1. The code RODEX2A is acceptable for mechanical analyses but RODEX2 must continue to be used for LOCA and transient analysis input generation.

  2. The RODEX2A calculation of fuel rod pressure must be performed to a minimum burnup of 50 MWd/kgU using the approved power history.

- Implementation of SER Restrictions:

  1. This SER restriction is implemented in engineering guidelines.

  2. The code RODEX2A was approved to a rod average burnup of 62 MWd/kgU in Reference 2-12. The analyzed burnup for all current designs is greater than 58 MWd/kgU.

- Observations: The RODEX2A code was approved to a maximum rod average burnup of 62 MWd/kgU in Reference 2-12.

2-6 : 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 40,000 MWd/MTU, respectively.

- SER Restrictions:

  1. LHGR limit curves (Figures 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 2-9.

  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.

- Implementation of SER Restrictions:

  1. This restriction no longer applies since the 8x8 and 9x9 fuel addressed by this report are no longer being supplied.

  2. and 3. These restrictions no longer apply. The exposure limits for BWR fuel were increased to 54 MWd/kgU for an assembly and to 62 MWd/kgU for a rod in Reference 2-12. These exposure limits are reflected in engineering guidelines.

- Observations: Although Reference 2-6 only discusses applications to 8x8 and 9x9 fuel types, the report includes a description of the process used to develop linear heat generation rates for fuel designs. Subsequent to the approval of this topical report, AREVA developed and the NRC approved the use of generic design criteria for new fuel designs (Reference 2-10). Reference 2-12 describes the use of the same LHGR methodology for application to the ATRIUM-9 and ATRIUM-10 designs.

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 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.

- Implementation of SER Restrictions: This and other issues would be addressed in response to a request from a licensee to justify continued operation of BWR fuel following an accident.

- Observations: Reference 2-10 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-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 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.

- Implementation of SER Restrictions: The SER restriction on 8 wt% gadolinia is implemented in engineering guidelines.

- Observations: In-reactor fission gas release test results (Reference 14) were provided to the NRC. The thermal conductivity model supersedes the previously approved model (Reference 2-1).

- Clarifications: NRC concurrence with a clarification related to the topical report was requested in Reference 33. The NRC concurrence with the clarification was provided in Reference 34. The clarification was with respect to the use of one conductivity equation for UO_2-only fuel and a separate gadolinia-bearing fuel conductivity equation for all gadolinia concentrations greater than zero wt%.

2-9 : 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 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.

- Implementation of SER Restrictions: This restriction no longer applies. LHGR limit curves can be established as allowed in Reference 2-10.

- Observations: The rod bow model approved in XN-75-32(P)(A) was approved for application to 9x9 fuel for assembly exposures to 40,000 MWd/MTU. The extended burnup data used 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 use of the same rod bow model up to 54,000 MWd/MTU for the ATRIUM-9 and ATRIUM-10 designs is described in Reference 2-12.

- Clarifications: NRC concurrence with a clarification related to the topical report was requested in References 27 and 28. The NRC concurrence with this clarification was provided in Reference 29. The clarification is that Reference 2-10 removes the need for a specific LHGR limit curve for BWR fuel designs and allows for LHGR limits to be established in accordance with the approved mechanical design criteria.

2-10 : 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 Restrictions:

1. Peak pellet 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.
2. Exposure beyond 60,000 MWd/MTU peak pellet must be approved by the NRC.
3. Approval does not extend to the development of additive constants for ANFB to co-resident fuel.
4. For each application of the mechanical design criteria, AREVA must document the design evaluation and provide a summary of the evaluation for the NRC.

- Implementation of SER Restrictions:

The revised SER restrictions on burnup are implemented in engineering guidelines.

1. The NRC approved higher burnup values as presented in Reference 2-12.
2. The exposure limit was extended to a rod-average burnup of 62 GWd/MTU by the approval of Reference 2-12.
3. The ANFB correlation is no longer used.
4. It was clarified in References 27 and 28 that this requirement applies to generic evaluations that are independent of plant specific evaluations. The NRC concurred with this in Reference 29.

- Observations: The application of the processes and criteria described in this topical report do not require prior NRC approval.

  The mechanical design of the fuel channel is performed using the criteria and methods described and approved in Reference 2-13.

  The design methodology for the reconstitution of a BWR fuel assembly complies with Reference 2-11.

2-11: ANF-90-82(P)(A) Revision 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 Restrictions: The reconstitution methodology is acceptable for reload licensing applications with the following conditions:

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

  2. The seismic LOCA analysis will be reassessed if the reconstructed weight drops below a proprietary value.

- Implementation of SER Restrictions: The SER restrictions are implemented in engineering guidelines.

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

  The SER restrictions on the number of replacement rods apply only to inert rods containing Zircaloy or stainless steel inserts.

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.

- Purpose: Extend the exposure limits of the RODEX2A (Reference 2-5) code, which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs.

- SER 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.

- Implementation of SER Restrictions: The SER restrictions on burnup are implemented in engineering guidelines.

- Observations: The RODEX2A code, which is used for BWR fuel design applications, is a derivative of AREVA's base fuel performance code RODEX2.

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

  1. Steady state analyses of maximum wall thinning from oxidation for end of life conditions will be performed.
  2. The growth correlations reviewed are applicable to all AREVA 9x9 fuel designs.
  3. Transient strain is to be calculated with the version of RODEX referenced in XN-NF-81-58(P)(A) Revision 2 Supplement 1 (Reference 2-3). Strain is limited to 1.0% and the limit is reduced at high exposures.
  4. Steady state strain is to be calculated with RODEX2A and is limited to 1%.
  5. RODEX2A is to be used to calculate fuel temperatures for fuel melt analyses.
  6. RODEX2 shall be used as the base fuel performance code to interface with the AREVA LOCA and transient thermal-hydraulic methodologies. The RODEX2 code was also approved for BWR analyses to 62 GWd/MTU rod average burnup.

- Clarifications: NRC concurrence with clarifications related to this topical report was requested in References 37 and 38. The NRC concurrence with these clarifications was provided in Reference 39. The clarification was associated with applying the exposure limits to only the full length fuel rods and not the part length fuel rods.

2-13: EMF-93-177(P)(A) Revision 1, "Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005.

- 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 accident loadings channel damage does not prevent control blade insertion and assembly coolability is maintained.

- SER Restrictions: Subject to certain conditions, the analyses conducted by AREVA are acceptable for licensing applications.

  1. The fuel channel TR (Technical Report) methods and criteria may be applied to fuel channel designs similar to the configuration of a square box with radiused corners open at the top and bottom ends. The wall thickness shall fall within the range of current designs. The channels shall be fabricated from either Zircaloy-2 or Zircaloy-4. AREVA will not use Zircaloy material for channels which has less strength than specified in the TR, and if the strength of material is greater than that in the TR, AREVA will not take credit for the additional strength without staff review.

  2. Updates to channel bulge and bow data are permitted without review by the NRC staff; however, AREVA shall resubmit the channel bulge and bow data statistics if the two-sigma upper and lower bounds change by more than one standard deviation

  3. This TR is approved using the ABAQUS or ANSYS codes in the deformation analysis. The use of other codes in the deformation analysis, i.e., NASTRAN, is beyond the current approval.

  The following restrictions are carried over from EMF-93-177(P)(A) Revisicn 0; for specific plant applications the following conditions are to be met:

  4. The allowable differential pressure loads and accident loads should bound those of the specific plant.

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

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

- Implementation of SER Restrictions: The SER restrictions are implemented in engineering guidelines.

- 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.

  The Reference 26 letter was provided to the NRC to inform them that Revision 0 of the topical report had been used to confirm the fuel channel design met the design criteria at an approved assembly exposure for which results had not been previously provided. No NRC response was requested.

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 is the 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 is consistent with 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 that pertain to nuclear design. Many of the GDCs relate to mechanical properties of the fuel assembly that are satisfied by meeting appropriate thermal and reactivity margin limits while the fuel resides in the reactor core. AREVA 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 3-4.

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 burnup. The transient LHGR design limit satisfies the strain and fuel overheating design criteria discussed in Section 2.2.2 and Section 2.2.12.4. 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 2-6 and 2-9). The LHGR limit is consistent with 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 2-10.

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. AREVA calculates the reactivity coefficients on a plant and cycle specific basis through application of the standard neutronics design and analysis methodology (References 3-1, 3-2, and 3-4).

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 NRC-approved AREVA 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.

AREVA uses the NRC-approved STAIF code (References 3-3 and 3-5) for stability evaluations. STAIF is a frequency domain code that simulates the dynamics of a BWR. AREVA performs cycle-specific analyses in order to establish reactor operating parameters that ensure stable operation throughout the cycle.

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, 3-2, and 3-4). It is calculated at exposure points throughout the cycle 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 3-2. AREVA 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 AREVA 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 in the following sections.

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 Restrictions: No restrictions

- Implementation of SER Restrictions:

  None

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

  Superseded Portions:

  Fuel Assembly Depletion Model - XFYRE replaced with CASMO-4 (see Reference 3-4).

  Core Simulator - XTGBWR replaced with MICROBURN-B2 (see Reference 3-4).

  Diffusion Theory Model - XDT replaced with CASMO-4 (see Reference 3-4).

  Stability Analysis - COTRAN replaced with STAIF (see Reference 3-5).

  Control Rod Withdrawal - XTGBWR replaced with MICROBURN-B2 (see Reference 3-4).

  Fuel Misloading Analysis – XFYRE replaced with CASMO-4 and XTGBWR replaced with MICROBURN-B2. These analyses are 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 3-2.

Applicable Portions:

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

Control Rod Withdrawal – This analysis determines the change in CPR (ΔCPR) for error rod patterns. In addition a check is made that the LHGR does not exceed the transient (PAPT) LHGR limit.

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 void reactivity coefficient is the same but MICROBURN-B2 is used instead of XTGBWR.

Doppler Reactivity Coefficient – Method used to calculate the core average Doppler coefficient is the same but CASMO-4 is used instead of XFYRE.

Scram Reactivity – Method used is the same except MICROBURN-B2 is used instead of XTGBWR.

Delayed Neutron Fraction – Calculated using CASMO-4 instead of XFYRE.

Prompt Neutron Lifetime – Calculated using CASMO-4 instead of XFYRE.

3-2 : 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 Restrictions: Conditions imposed were based on pending approvals of outstanding topical reports which have been subsequently approved.

- Implementation of SER Restrictions: This restriction is no longer applicable (because of subsequent approvals).

- Observations: Many of the codes and methodologies referenced have changed or have been replaced since the report was approved.

- Clarifications: AREVA provided a clarification related to the topical report in References 27 and 28. The clarification was associated with the use of power and flow dependent LHGR multipliers to establish LHGR limits that provide adequate margin during events initiated from off-rated conditions.

3-3 : 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 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.

- Implementation of SER Restrictions: The SER restrictions are implemented in the code and the users manual for STAIF. The requirements will automatically be satisfied if the code defaults are used and the MICROBURN-B2 STAIF guideline is followed.

- Observations: Stability analysis procedures described in Reference 3-1 were superseded by the approval of the STAIF code (References 3-3 and 3-5).

3-4 : EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation, October 1999.

- Purpose: Replace the MICBURN-3/CASMO-3G bundle depletion codes and the MICROBURN-B simulator code with the codes CASMO-4 and MICROBURN-B2, respectively.

- SER Restrictions:

  1. The CASMO-4/MICROBURN-B2 code systems shall be applied in a manner that predicted results are within the range of the validation criteria (Tables 2.1 and 2.2) and measurement uncertainties (Table 2.3) presented in EMF-2158(P).

  2. The CASMO-4/MICROBURN-B2 code system shall be validated for analyses of any new fuel design which departs from current orthogonal lattice designs and/or exceed gadolinia and U-235 enrichment limits.

  3. The CASMO-4/MICROBURN-B2 code system shall only be used for BWR licensing analyses and BWR core monitoring applications.

  4. The review of the CASMO-4/MICROBURN-B2 code system should not be construed as a generic review of the CASMO-4 or MICROBURN-B2 computer codes.

  5. The CASMO-4/MICROBURN-B2 code system is approved as a replacement for the CASMO-3G/MICROBURN-B code system used in NRC-approved AREVA BWR licensing methodology and in AREVA BWR core monitoring applications. Such replacements shall be evaluated to ensure that each affected methodology continues to comply with its SER restrictions and/or conditions.

  6. AREVA shall notify any customer who proposes to use the CASMO-4/MICROBURN-B2 code system independent of any AREVA fuel contract that conditions 1 through 4 above must be met. AREVA's notification shall provide positive evidence to the NRC that each customer has been informed by AREVA of the applicable conditions for using the code system.

- Implementation of SER Restrictions: The SER restrictions relevant to methodology used by AREVA are implemented in engineering guidelines.

- Observations: None.

3-5 : EMF-CC-074(P)(A) Volume 4, Revision 0, "BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000.

- Purpose: Document and justify enhancements to the STAIF code including the capability to accept input from the code MICROBURN-B2. Justify a modification to the approved stability criteria for STAIF in conjunction with input from both MICROBURN-B and MICROBURN-B2. The STAIF code is used to perform stability analysis for BWRs.

- SER Restrictions:

  The SER concludes that the STAIF code is acceptable for best-estimate decay ratio calculations. This conclusion applies to the three types of instabilities relevant to BWR operation, which are quantified by the hot-channel, core-wide, and out-of-phase decay ratios. The staff estimates that STAIF decay ratio calculations for the decay ratio range of 0.0 to 1.1 are accurate within:

  +/- .2 for the hot-channel decay ratio
  +/- .15 for the core-wide decay ratio
  +/- .2 for the out-of-phase decay ratio

  The staff concludes that the proposed modification of the E1A acceptance criteria for region-validation calculations is acceptable because it provides the intended protection against instabilities outside the E1A regions. The following E1A region-validation criteria are acceptable for the STAIF code:

  The calculated hot-channel decay ratio must be lower than .8.
  The calculated core-wide decay ratio must be lower than .85.
  The calculated out-of-phase decay ratio must be less than .8.

- Implementation of SER Restrictions: The SER restrictions are implemented in engineering guidelines.

- Observations: The NRC stated in Reference 35, that the revised stability criteria is applicable to calculations with the STAIF code with input from either MICROBURN-B or MICROBURN-B2.

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 and 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 SAFDLs are not exceeded during normal operation and AOOs. Specific criteria are identified in Reference 2-10 and discussed below.

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 such 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 (Reference 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 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 AREVA fuel designs and 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 AREVA 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 a specified difference in assembly orifice flow for a given (or specified) assembly power at rated conditions (i.e., full flow and full power), additional core stability evaluations will be performed with the STAIF methodology (Reference 3-5). 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

AREVA 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 AREVA 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 AREVA 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.

AREVA fuel assembly pressure drop methodology is presented in Reference 4-1. This methodology is part of the calculational method used by AREVA 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 AREVA fuel assembly critical power performance is established by means of an empirical correlation based on results of boiling transition test programs (see Reference 4-5). The critical power performance of co-resident fuel, which is not in the AREVA correlation development data base, is determined using the methodology described in Reference 4-4.

The methodology and computer codes for AREVA BWR plant transient analyses are the XCOBRA-T code (Reference 5-6) and the COTRANSA2 code (Reference 5-7). 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.

Reference 4-3 provides the basis for the AREVA methodology for determining the SLMCPR which ensures that 99.9% of the fuel rods are expected to avoid boiling transition. The SLMCPR is determined by statistically combining calculational uncertainties and 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 boiling transition are varied randomly and the total number of rods experiencing boiling transition is determined for each Monte Carlo trial. The AREVA CPR correlations depend on the core 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 3-4. 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 Reference 4-5.

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 criterion is addressed during the specific mechanical design analysis performed for each fuel type. The bases are 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 AREVA 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 not affect thermal margins because 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 (Reference 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, AREVA 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 AREVA 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 in the following sections.

4-1 : 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 Restrictions: No restrictions.
- Implementation of SER Restrictions: None.
- Observations: This methodology continues to be used and incorporates experimental pressure drop data for new fuel and spacer designs.

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 combination of uncertainties.

- SER Restriction: Monitoring systems other than POWERPLEX®* CMSS may be used provided that the associated power distribution uncertainties are identified and appropriate operating parameters compatible with ENC transient safety analyses are monitored. Whatever monitoring system is used should be specifically identified in plant submittals.

- Implementation of SER Restriction: The SER restriction is implemented in engineering guidelines.

- Observations: Although Reference 4-2 only discusses applications to ENC 8x8 and 9x9 fuel types, the overall methodology is applicable to other AREVA fuel designs when appropriate CHF correlations are implemented. Subsequent to the approval of this topical report, AREVA developed and the NRC approved the use of generic design criteria for new fuel designs (Reference 2-10). In the SER/TER for Reference 2-10, the NRC concurred with the continued applicability of the methodology in Reference 4-2 (with the exception of the CHF correlation) for demonstrating compliance with thermal hydraulic design criteria.

- Some of the computer codes referenced in the topical report have been superseded by other NRC-approved codes (e.g., COTRANSA with COTRANSA2, XTGBWR with MICROBURN-B2) and the XN-3 CHF correlation has been supplemented with the NRC-approved SPCB CHF correlation (see Reference 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 5-6), we find the use of the steady-state code [XCOBRA] acceptable in this context." XCOBRA continues to be applied for steady-state analyses.

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* POWRPLEX is a trademark registered in the U.S. and various other countries.

4-3 : 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 the SLMCPR.

- SER Restrictions:

    1. The NRC approved MICROBURN-B power distribution uncertainties should be used in the SLMCPR determination.
    2. Since the ANFB correlation uncertainties depend on fuel design, in plant-specific applications the uncertainty value used for the ANFB additive constants should be verified. (Note, ANFB was subsequently replaced in the methodology by the SPCB correlation, Reference 4-5.)
    3. The CPR channel bowing penalty for non-ANF fuel should be made using conservative estimates of the sensitivity of local power peaking to channel bow.
    4. The methodology for evaluating the effect of fuel channel bowing is not applicable to reused second-lifetime fuel channels.

- Implementation of SER Restrictions: SER restrictions 1 and 2 are implemented in engineering guidelines and automation tools. Restrictions 3 and 4 are implemented in engineering guidelines.

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

  Power distribution uncertainties for MICROBURN-B2 and other AREVA 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.

4-4 : EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.

- Purpose: Present and justify the use of AREVA critical power correlations to co-resident fuel (non-AREVA manufactured).

- SER Restrictions: Technology transfer to licensees who may be responsible for using these processes will be accomplished through AREVA and licensee procedures consistent with the requirements of GL 83-11, Supplement 1. This process includes the performance of an independent benchmarking calculation by AREVA for comparison to licensee-generated results to verify that the application of AREVA CHF correlations is properly applied for the first application by a licensee.

- Implementation of SER Restrictions: The SER restriction is implemented in engineering work practices.

- Observations: None.

4-5 : EMF-2209(P)(A) Revision 2, "SPCB Critical Power Correlation", Framatome ANP, September 2003.

- Purpose: Present and justify a critical power correlation applicable for the ATRIUM-9B and ATRIUM-10 fuel designs.

- SER Restrictions:

  1. The SPCB correlation is applicable to Framatome ANP, Inc. ATRIUM-9B and ATRIUM-10 fuel designs with a local peaking factor no greater than 1.5.

  2. If in the process of calculating the MCPR safety limit, the local peaking factor exceeds 1.5, an additional uncertainty of 0.026 for ATRIUM-9B and 0.021 for ATRIUM-10 will be imposed on a rod by rod basis.

  3. The SPCB correlation range of applicability is 571.4 to 1432.2 psia for pressure, 0.087 to 1.5 Mlb/hr-ft^2 for inlet mass velocity and 5.55 to 148.67 Btu/lbm for inlet subcooling.

  4. Technology transfer will be accomplished only through the process described in Reference 12, which includes the performance of an independent bench-marking calculation by FANP for comparison to the licensee-generated results to verify that the new CHF correlation (SPCB) is properly applied for the first application by the licensee.

  5. Application of this correlation and the proposed revisions to fuel designs other than the ATRIUM-9B and ATRIUM-10 designs require prior staff approval.

  Note, restrictions 1 – 4 are from Revision 1.

- Implementation of SER Restrictions: SER restrictions 1 and 5 are implemented in engineering guidelines. Restriction 2 is implemented in engineering guidelines and automation tools. Restriction 3 is directly implemented in engineering computer codes. Restriction 4 is implemented in engineering work practices.

- Observations: The purpose of Revision 2 was to modify the SPCB critical power correlation in the region of the uranium blanket at the top of the fuel.

- Clarifications: NRC concurrence with a clarification related to this topical report (Revision 1) was requested in References 30 and 31. The NRC concurrence with the clarification was provided in Reference 32. The clarification discusses the actions taken when the calculation values fall outside the correlation bounds.

5.0 Accident Analysis

This section addresses the methodologies used to perform the analyses of AOOs and postulated accidents in SRP Chapter 15 that are related to core reloads.

5.1 *Anticipated Operational Occurrences*

AOOs are evaluated to determine thermal operating limits to ensure applicable event acceptance criteria are met. Table 5-1 lists those AOOs analyzed with AREVA's approved methodologies.

Table 5-1 Anticipated Operational Occurrence Analyses

SRP No.	Chapter 15 AOO Analysis
15.1.1 - 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow
15.2.1 - 15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1 -15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a BWR Pressure Relief Valve

5.1.1 <u>Regulatory Requirements</u>

The specific criteria necessary to meet the requirements of the relevant GDCs 10, 15, and 26 for the 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) Cladding strain does not exceed 1%.

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

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.

b) Uniform cladding strain does not exceed 1%.

Analyses are performed to demonstrate that the fuel performs within design criteria during AOOs and to establish appropriate operating limits for the reactor. To protect the established safety limit MCPR, evaluations of AOOs are performed which produce the limiting transient ΔCPR, which when added to the safety limit MCPR, defines the operating limit MCPR. The methodologies used for the analysis of these events are found in References 3-1, 3-2, 4-2, 4-4, 4-5, 5-4, 5-5, 5-6, 5-7, and 5-12.

5.1.2 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 conditions for plant operations are established to assure that acceptable thermal operating margins are maintained during all anticipated operations. Application of AREVA's methodology provides a basis for the determination that plant operation will meet appropriate safety criteria.

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

- Decrease in core coolant temperature
- Increase in reactor pressure
- Decrease in reactor coolant flow rate
- Reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Decrease in reactor coolant inventory

- Increase in reactor coolant flow
- Increase in reactor core coolant temperature.

Primarily because of the strong void reactivity feedback characteristic of a boiling water reactor, AOOs involving a decrease in reactor coolant inventory, a decrease in core flow, or an increase in core coolant temperature do not result in a limiting ΔCPR.

A decrease in core coolant temperature may result in a gradual core heatup until the high neutron flux scram setpoint is exceeded. Since the power excursion is slow and the fuel thermal response does not significantly lag the neutronic response, this event can be evaluated with either a transient code or a steady-state code.

Rapid reactor pressure increases may result in 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 is considered in the determination of cycle limiting conditions for operation.

Reactor and power distribution anomalies are localized reactivity additions that are usually initiated by operator error in selecting and withdrawing a control rod. While the event during refueling and reactor startup conditions are not limiting, the rod withdrawal error at power is potentially limiting and considered in the determination of the thermal operating limits.

The two event categories which involve increases in either core coolant flow rate or reactor coolant 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 recirculation 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.

Prior to the initial cycle that AREVA provides reload fuel, 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 AOOs are generally identified as being potentially limiting:

- Turbine/generator trip without bypass
- Feedwater controller failure to maximum demand
- Loss of feedwater heating
- Control rod withdrawal error
- Recirculating flow increase events

Once the applicable set of limiting transients for thermal margin has been identified for a specific reactor, the analysis of each event at reactor conditions at which it is potentially limiting provides the basis for determining the thermal operating limits.

5.1.3 Pressurization Transient Analysis

Events that result in significant reactor pressure increases are those that result in the closure of the steam isolation or turbine valves. There are several potential causes for the valve closure including loss of generator load, excessive turbine vibration and reaching a system set point (e.g. water level, low system pressure). The sudden reduction in steam flow causes a increase in reactor system pressure and core power. The event is usually terminated by reactor scram. In many cases, turbine bypass valves and safety relief valves operate to limit the system pressure rise. The turbine trip, generator load rejection and MSIV closure events are included in this classification. The feedwater controller failure event has many of the characteristics of these same events as it is a combination of a increase in coolant inventory and decrease in core coolant temperature event followed by a increase in reactor pressure event when the high water level trip setpoint is reached. The methodology used for the pressurization transient AOO analyses is presented in References 4-2, 5-6, and 5-7.

The plant transient AOO analysis methodology is also used in the overpressurization analyses to demonstrate compliance with the ASME pressure vessel code requirements.

5.1.4 Generic Loss of Feedwater Heating Methodology

The NRC has approved a generic AREVA methodology for evaluating the loss of feedwater heating (LFWH) transient in BWRs (Reference 5-12). The generic methodology is a parametric description of the critical power ratio response that was developed using the results of many applications of the previously approved plant and cycle specific methodology (Reference 3-1). Applying this methodology results in a conservative MCPR operating limit for the LFWH event.

5.1.5 Control Rod Withdrawal Error

During the control rod withdrawal error transient, the reactor operator is assumed to ignore the local power range monitor alarms and the rod block monitor alarms and continue to withdraw the control rod until the control rod motion is stopped by the control rod block. 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.

A detailed description of the AREVA control rod withdrawal error evaluation methodology is given in Reference 3-1. As noted in Reference 3-4, MICROBURN-B2 is approved for use in performing the analysis as a replacement to previously approved codes.

For BWR/6 reactors, the AREVA generic control rod withdrawal error analysis (Reference 5-4) is used. The generic analysis has been extended to cover maximum extended operating domain (MEOD) operation (Reference 5-5).

5.1.6 Recirculation Flow Increase

A slow flow excursion event assumes a failure of the recirculation flow control system such that the core flow increases slowly to the maximum flow physically attainable by the equipment. An uncontrolled increase in flow creates the potential for a significant increase in core power and heat flux. The analysis is performed using XCOBRA (Reference 4-2) to calculate the change in critical power ratio during the flow increase. Similar analyses are performed using MICROBURN-B2 (Reference 3-4) to determine the change in LHGR during a flow increase event.

The results of the slow flow excursion analyses are used to establish flow dependent MCPR ($MCPR_f$) limits and flow-dependent LHGR multipliers. The $MCPR_f$ limits ensure that the SLMCPR is protected if the recirculation flow is inadvertently increased to the maximum attainable value based on the plant equipment limitations.

5.1.7 Determination of Thermal Limits

The results of the evaluation of the anticipated operational occurrences at rated and off-rated power and flow conditions are used to establish power-dependent MCPR ($MCPR_p$) operating limits, including limits at rated power. As noted earlier, the results of the slow flow run-up event are used to establish the flow-dependent MCPR limits.

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 LHGR multipliers ($LHGRFAC_p$ and $LHGRFAC_f$) are established. The minimum of either the $LHGRFAC_p$ or $LHGRFAC_f$ multiplier is applied directly to the steady state LHGR limit to determine the applicable LHGR operating limit to ensure that the 1% strain and centerline melt criteria are not violated during an AOO.

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 assumed in the safety analyses, surveillance procedures are specified to determine the continued applicability of the data.

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 *Postulated Accidents*

Postulated accidents for BWRs evaluated for compliance with relevant GDCs are listed in Table 5-2 below.

Table 5-2 Postulated Accident Analyses

SRP No.	Chapter 15 Accident Analysis
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
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)
15.6.5	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

5.2.1 Regulatory Requirements

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 – 15.3.4; GDCs 27, 28, and 31

a) Pressure in the reactor coolant and main steam systems should be maintained below design limits.

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 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 (Reference 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 assessed in 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 (Reference 11).

SRP No. 15.6.5; GDC 35

a) Event-specific criteria are specified in: 10 CFR 50.46 and 10 CFR 50 Appendix K.

b) Regulatory Guide 1.3 (Reference 15) 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 (Reference 16), with the exception of the guidelines for the atmospheric dispersion factors (χ/Q values). The acceptability of the χ/Q values is determined under SRP Section 2.3.4.

The methodologies used to analyze the hypothetical LOCAs and other postulated accidents are discussed in the following sections.

5.2.2 Pump Seizure

Recirculation pump seizure (RPS) event is considered an accident where an operating recirculation pump suddenly stops rotating. There are three parts to the RPS analysis - the simulation of the reactor system response, the determination of the number of failed fuel rods, and the radiological dose assessment.

The first part of the analysis uses the COTRANSA2 (Reference 5-7) and XCOBRA-T (Reference 5-6) codes to simulate the system and limiting assembly response. The key parameter determined is the ΔCPR for the limiting assembly during the event. The second part is the determination of the number of failed rods. The minimum CPR for the event is determined from the OLMCPR and the calculated ΔCPR. The AREVA critical power methodology (Reference 4-3) is used to calculate the number of rods expected to experience boiling transition at the minimum CPR during the event. 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 are calculated to fail. If the minimum CPR during the event remains above the safety limit MCPR, the 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.

Depending on the specific FSAR 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 result must remain below a small fraction (10%) of the 10 CFR 100 limits. For a limiting fault/design basis accident, the dose calculation result must not exceed 10 CFR 100 limits. 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.

5.2.3 Fuel Loading Error

Two separate incidents are analyzed as part of the fuel misload analysis. 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 (Reference 4) as described in Reference 3-2.

The inadvertent misloading of a fuel assembly into an incorrect core location is analyzed with the MICROBURN-B2 methodology described in Reference 3-4. 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 fuel assembly. 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 fuel assembly. This prediction of the number of fuel rods in boiling transition is performed in accordance with the methodology reported in Reference 4-3.

The inadvertent rotation of a fuel assembly from its intended orientation is evaluated with the CASMO-4 methodology described in Reference 3-4. 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 fuel assembly. This prediction of the number of fuel rods in boiling transition is performed in accordance with the methodology reported in Reference 4-3. If an assessment of MCPR and LHGR show the potential for rod failures, a radiological evaluation may be needed to demonstrate that the off-site dose criterion (10 CFR 100) is met for both the fuel misload and fuel misorientation.

5.2.4 Control Rod Drop Accident Analysis

Analysis of the postulated CRDA is performed on a generic basis in Reference 3-1. Because the behavior of the fuel and core during such an event is not dependent upon system response, a generic CRDA parametric analysis can be applied to all BWR types.

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. For each cycle-specific application, values of each of the parameters are calculated and applied to the generic parametric analysis results and the resulting deposited fuel enthalpy is determined. The applicability of the generic analysis is verified for each application by comparison of the generic parameter range to the cycle-specific parameters, e.g., control rod worth, beta-eff and Doppler reactivity coefficient.

5.2.5 Loss of Coolant Accident Analysis

Plant specific ECCS analyses provide peak cladding temperature (PCT) and maximum local metal-water reaction (MWR) values and establish MAPLHGR limits for each fuel design. For the limiting single failure and limiting break, calculations are performed to determine the PCT and MWR values over the expected exposure lifetime of the fuel when operating at the MAPLHGR limit. The limiting break is determined by evaluating a spectrum of potential break locations, sizes, and single failures.

The limiting single failure of ECCS equipment is that failure which results in the minimum margin to the PCT criterion. The plant FSAR identifies potentially limiting ECCS single failures. AREVA analyzes those potentially limiting failures and identifies the worst single failure for the AREVA fuel design.

Evaluations and analyses to establish the location of the limiting break are performed. Analyses are performed for breaks on the suction and discharge sides of the recirculation pump. Non-recirculation line breaks are also evaluated but are generally non-limiting. The determination of the limiting location is based on minimum margin to the PCT criterion calculated for consistent fuel exposure conditions at each of the break locations. The MWR criterion is typically not challenged if the PCT limit is met, and is normally reported for the highest PCT case.

Analyses to establish the size of the limiting break are performed. Hypothetical split and guillotine piping system breaks are evaluated up to and including those with a break area equal to 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 to the PCT criterion.

The condition of the fuel during 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 is based on fuel conditions associated with planar average exposure.

The AREVA Appendix K LOCA methodology is referred to as the EXEM BWR-2000 Evaluation Model (Reference 5-11). The reactor system and hot channel response is evaluated with RELAX (References 5-2, 5-8, and 5-9). Fuel assembly heatup during the LOCA is analyzed with HUXY (Reference 5-1) which incorporates approved cladding swelling and rupture models (Reference 5-3). Stored energy and fuel characteristics are determined with RODEX2 (Reference 2-3).

The use of Appendix K spray heat transfer coefficients for the ATRIUM-10 fuel design is justified in Reference 5-10.

5.2.6 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, AREVA performs an incremental evaluation of the impact of the new fuel design on the fuel handling accident scenario defined in the FSAR. Using the boundary conditions and conservative assumptions given in the FSAR and the relevant characteristics of the new fuel design, AREVA calculates a conservative number of fuel rods expected to fail as a result of a fuel handling accident.

The radiological consequences of a fuel handling accident for a new mechanical fuel design are assessed based on the same reactor power history assumed in the evaluation of the existing fuel. The plenum activity for the new fuel is calculated based on the relative number of fuel rods per fuel assembly and relative maximum rod LHGR for the new and existing fuel designs.

5.3 *NRC-Accepted Topical Report References*

The NRC-accepted topical reports for AOO and accident analyses are listed in the following sections.

5-1 : 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 fuel rod strains and ballooning.

- SER Restrictions:

  1. The staff, however, will require that a conservative reduction of 10% be made in the (spray heat transfer) coefficients specified in 10 CFR 50 Appendix K for 7x7 assemblies when applied to ENC 8x8 assemblies.

  2. In each individual plant submittal employing the Exxon model the applicant will be required to properly take rod bowing in account.

  3. Since GAPEX is not identical to HUXY in radial noding or solution scheme, it is required that the volumetric average fuel temperature for each rod be equal to or greater than that in the approved version of GAPEX. If it is not, the gap coefficient must be adjusted accordingly.

  4. It has been demonstrated that the (2DQ local quench velocity) correlation gives hot plane quench time results that are suitably conservative with respect to the available data when a coefficient behind the quench front of 14000 Btu/(hr-ft^2-°F) is used.

  5. It (Appendix K) requires that heat production from the decay of fission products shall be 1.2 times the value given by K. Shure as presented in ANS 5.1 and shall assume infinite operation time for the reactor.

  6. It is to be assumed for all these heat sources (fission heat, decay of actinides and fission product decay) that the reactor has operated continuously at 102% of licensed power at maximum peaking factors allowed by Technical Specifications.

  7. For small and intermediate size breaks, the applicability of the fission power curve used in the calculations will be justified on a case by case basis. This will include justification of the time of scram (beginning point in time of the fission power decrease) and the rate of fission power decrease due to voiding, if any.

8. The rate of (metal water) reaction must be calculated using the Baker-Just equation with no decrease in reaction rate due to the lack of steam. This rate equation must be used to calculate metal-water reactions both on the outside surface of the cladding, and if ruptured, on the inside surface of the cladding. The reaction zone must extend axially at least three inches.

9. The initial oxide thickness (that affects the zirconium-water reaction rate) used should be no larger than can be reasonably justified, including consideration of the effects of manufacturing processes, hot-functional testing and exposure.

10. Exxon has agreed to provide calculations on a plant by plant basis to demonstrate that the plane of interest assumed for each plant is the plane in which peak cladding temperatures occur for that plant.

- Implementation of SER Restrictions: SER restrictions 1, 2, 3, 4, 6, 7, 9, and 10 are implemented in engineering guidelines. Restrictions 5 and 8 are directly implemented in engineering computer codes.

  Observations: None.

5-2 : 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 criteria and 10 CFR 50 Appendix K requirements.

- SER Restrictions: 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.

- Implementation of SER Restrictions: The FLEX computer code is no longer used. This was replaced in Reference 5-11.

- Observations: RELAX and FLEX, which are key computer codes in the methodology, have been subsequently modified as described in References 5-8 and 5-9, which documents the revised EXEM BWR Model, and in Reference 5-11 which documents EXEM BWR-2000 in which the RELAX code replaced FLEX. The EXEM BWR-2000 model supersedes the prior evaluation model.

5-3 : 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 (Reference 10) which comply with 10 CFR 50 Appendix K requirements into the HUXY code (Reference 5-1).
- SER Restrictions: No restrictions.
- Implementation of SER Restrictions: None.
- Observations: The swelling and rupture model is currently applicable.

5-4: 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 3-1) for application to BWR/6s or other BWRs with ganged control rods.

- SER Restrictions:

  1. 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.

  2. 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.

- Implementation of SER Restrictions:

  The SER restrictions are implemented in engineering guidelines.

- Observations: The original methodology, developed using the XTGBWR core simulator code which was superseded by MICROBURN-B2 (see Reference 3-4), is still applicable.

5-5 : 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 Reference 5-4 licensing topical report to control rod withdrawal error transients for BWR/6 plants within the extended operating domain.

- SER Restrictions:

  1. 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.

  2. 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.

- Implementation of SER Restrictions:

  The SER restrictions are implemented in engineering guidelines.

- Observations: The original methodology, developed using the XTGBWR core simulator code which was superseded with MICROBURN-B2 (see Reference 3-4), is still applicable.

5-6 : 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 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 for licensing purposes.

  3. XCOBRA-T licensing calculations must use NRC approved default options for void-quality relationship and two-phase multiplier correlations.

  4. The use of XCOBRA-T is conditional upon a commitment by ENC to a follow-up program to examine the XCOBRA-T void profile against experimental data from other sources.

- Implementation of SER Restrictions: SER restrictions 1, 2, and 3 are implemented in engineering guidelines. SER restriction 3 is also implemented through code controls (defaults, override warning messages). Restriction 4 was subsequently addressed in Reference 36 and no further action is required.

- Observations: None.

- Clarifications: NRC concurrence with an interpretation of the contents of the topical report was requested in References 23 and 24. The NRC concurrence with the interpretation was provided in Reference 25. The interpretation was with regard to a commitment to perform critical heat flux ratio evaluations at every node in the hot channel.

  NRC concurrence with clarifications related to SER and TER issues concerning the topical report was requested in References 27and 28. The NRC concurrence with these clarifications was provided in Reference 29. These references clarify that XCOBRA-T is approved for the analysis of the following events:

SRP Section	Chapter 15 Analysis
15.1.1 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Demand
15.2.1 – 15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressure Relief Valve and BWR Pressure Relief Valve
15.8	Anticipated Transients Without Scram (the Initial Pressurization Only)

5-7 : 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 Restrictions: The staff reviewed the subject safety evaluations and identified the following limitations that apply to COTRANSA2:

  1. Use of COTRANSA2 is subject to limitations set forth for methodologies described and approved for XCOBRA-T and COTRAN.

  2. The COTRANSA2 code is not applicable to the analysis of any transient for which lateral flow in a bundle is significant and nonconservative in the calculation of system response.

  3. 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 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.

- Implementation of SER Restrictions: SER restrictions 1, 2, and 4 are implemented in engineering guidelines. Restriction 3 is implemented in engineering guidelines and automation tools.

- Observations: The COTRANSA2 SER restrictions are similar to those for XCOBRA-T (Reference 5-6).

- Clarifications: NRC concurrence with clarifications related to SER and TER issues concerning the topical report was requested in References 27 and 28. The NRC concurrence with these clarifications was provided in Reference 29. These references clarify that COTRANSA2 is approved for the analysis of the following events:

SRP Section	Chapter 15 Analysis
15.1.1 – 15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Demand
15.2.1 – 15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressure Relief Valve and BWR Pressure Relief Valve
15.8	Anticipated Transients Without Scram (the Initial Pressurization Only)

5-8 : 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 Restrictions:

  1. The modified Dugall-Rohsenow heat transfer correlation has been shown to yield conservative results for many experimental measurements. The applicant used a suitable multiplier in the comparison calculations. Licensees will use this multiplier in licensing calculations.

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

  3. The staff requires that the revised evaluation model be protected with appropriate quality assurance procedures.

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

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

- Implementation of SER Restrictions: SER restrictions 1 and 2 are directly implemented in engineering computer codes. Restriction 3 is implemented in engineering work practices. Restriction 4 is implemented in engineering guidelines and automation tools. Restriction 5 is implemented in engineering guidelines.

- Observations: The RELAX code, with the jet pump update from ANF-91-048(P)(A) Supplements 1 and 2, and FLEX models were approved. This evaluation model has subsequently been superseded by EXEM BWR-2000 (Reference 5-11).

5-9 : 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 LOCA conditions.
- SER Restrictions: No restrictions imposed.
- Implementation of SER Restrictions: None.
- Observations: The jet pump model was approved.

5-10: EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000.

- Purpose: Justify the use of 10 CFR 50 Appendix K convective heat transfer coefficients during loss of coolant accident spray cooling for the ATRIUM-10 fuel design.
- SER Restrictions: None.
- Implementation of SER Restrictions: None.
- Observations: None.

5-11: EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001.

- Purpose: Describes an evaluation model for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology complies with 10 CFR 50.46 and 10 CFR 50 Appendix K.
- SER Restrictions: The staff concluded that the EXEM BWR-2000 Evaluation Model was acceptable for referencing in BWR LOCA analysis, with the limitation that the application of the revised evaluation model be limited to jet pump applications.
- Implementation of SER Restrictions: The SER restriction is implemented in engineering guidelines.
- Observations: Replace the FLEX code by the code RELAX in the BWR LOCA methodology.

5-12: ANF-1358(P)(A) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005.

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

- SER Restrictions:

  1. The methodology applies to BWR/3, BWR/4, BWR/5, and BWR/6 plants, and the fuel types which were part of the database (GNF-8X8, 9/9B and 11; ANF-8X8 and 9/9; and ATRIUM-9B and 10), provided that the exposure, the ratio of rated power and rated steam generation rate, rated feedwater temperature, and change in feedwater temperature are within the range covered by the data points presented in ANF-1358(P)(A), Revision 3.

  2. To confirm applicability of the correlation to fuel types outside the database, AREVA will perform additional calculations using the methodology, as described in Section 3.0 of the SER. In addition, AREVA calculations will be consistent with the methodology described in EMF-2158(P)(A), Revision 0 and comply with the guidelines and conditions identified in the associated NRC staff SE.

  3. The methodology applies only to the MCPR operating limit and the LHGR for the LFWH event.

- Implementation of SER Restrictions:

  The SER restrictions are implemented in engineering guidelines.

- Observations: The topical report includes results for GNF-8X8, -9/9B and -11; ANF-8X8, -9/9; and ATRIUM-9B and -10 fuel. Application of the correlation to fuel types outside the database needs to be verified according to SER Restriction Item 2.

6.0 Criticality Safety Analysis

In addition to reactor systems, AREVA 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 (Reference 17). The CASMO bundle depletion code (Reference 3-4) is used to calculate k_∞ values for fuel assemblies at beginning of life (new fuel storage) and 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 computer codes are widely used throughout the nuclear industry. They are used primarily for criticality safety and core physics calculations, respectively. AREVA has broad experience in the use of both of these codes. KENO.Va has been benchmarked by AREVA against critical experiment data to define appropriate reactivity biases and uncertainties.

AREVA performs criticality safety analyses consistent with the guidance given in References 18 - 22. The acceptance criteria (k-eff limit) for specific analyses are as defined in the plant Technical Specifications or from Chapters 9.1.1 (New Fuel Storage) or 9.1.2 (Spent Fuel Storage) of the Standard Review Plan NUREG-0800, References 18 and 19, respectively.

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.

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Environmental Monitoring Emergency Kit No. 1

MONTH/YEAR _____

Minimum Quantity	Equipment/Supplies	Remarks	Verified (Initial)
N/A	Container Seals	Is seal present on door? Yes ___ No ___*	
1	Global Positioning System (GPS) unit	Full battery charge	
1	Air Sampler with combination filter holder	#_______. Does it run? Yes ___ No ___ Calib. Due Date _______	
1	Portable generator	Does it run? Yes ___ No ___ Is fuel available? Yes ___No ___ Oil level - SAT ___ UNSAT___	
2	Check 0-500 mR self-reading dosimeters.	Calib. Due Date _______**	
10	Check 0-5 R self-reading dosimeters.	Calib. Due Date _______**	
1	RO-2A or Equivalent	#_______. Calib. Due Date _______	
1	Bicron Micro R Meter	#_______. Calib. Due Date _______	
1	RM-14 with pancake type G-M probe or Equivalent	#_______. Calib. Due Date _______	
1	Teletector or Equivalent	#_______. Calib. Due Date _______	
1	Check source (approximately 8μCi Cs^{137})	# __________	

*Inventory of kit must be checked.

**All dosimeters of the same range should be due for recalibration in the same month.

Environmental Monitoring Emergency Kit No. 1

MONTH/YEAR _____

Minimum Quantity	Equipment/Supplies	Remarks	Verified (Initial)
27	TLDs (2 for team members) (25 for environmental monitoring)	Inventory and/or change out all TLDs in accordance with DOS-NGGC-0009, Thermoluminescent Dosimeter (TLD) Badge Exchange.	
2	Bottles of potassium iodide (KI) tablets.	Expiration Date _________. (If the expiration date is less than 8 months in the future, reorder KI using Attachment 10.)	
2	Copies of 0PEP-03.7.6, Emergency Exposure Controls, Attachments 3 and 4.	Current Revision No. ______.	
1	Check source (approximately 8μCi Cs-137)	# ___________	
20	Plastic petri dishes with covers		
20	Poly ziplock bags, small		
1	Box of surgeon's gloves		
1	Siren key		
10	Silver zeolite cartridges	Expiration date _________.	
2	Magic markers		
1	Box of pens		
1	Box of 47 mm air sample filters		
5	Air sample charcoal cartridges	Expiration date _________.	
1	Dosimeter charger with batteries		

Environmental Monitoring Emergency Kit No. 1

MONTH/YEAR

Minimum Quantity	Equipment/Supplies	Remarks	Verified (Initial)
1	Flashlight		
12	D-cell batteries	Expiration date __________.	
12	AA-cell batteries	Expiration date __________.	
12	9-volt transistor batteries	Expiration date __________.	
12	C-cell batteries	Expiration date __________.	
2	Rolls of duct tape		
2	Protective clothing packages		
1	Log book		
10	One-gallon collapsible sample bottles		
10	Shipping boxes for gallon sample bottles		
1	Funnel		
1	Hand shovel or trowel		
1	Large Tri-pour beaker (800 ml)		
1	Clipboard		
2	Pads paper		
50	Poly zip-lock bags, medium		
1	Portable 2 channel radio w/charger		
1	Pair of tweezers		
1	Map of local area		
1	Book - Brunswick County Maps		

Environmental Monitoring Emergency Kit No. 1

MONTH/YEAR

Minimum Quantity	Equipment/Supplies	Remarks	Verified (Initial)
6	Bottles of drinking water	Expiration date _________.	
N/A	* All instruments were left in the Off Position.		

* Ludlum model 177 must be "on" for charging.

Initials

Seal kit. _____

Submit data to update computer schedule. _____

Comments: ____

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____

____

Inventory Performed By: ____________________ Date: ____________________
E&RC Technician

Reviewed By: ____________________ Date: ____________________
E&RC Supervisor or Designee