BSEP 07-0108
Enclosure 6

AREVA Presentation Slides
for Brunswick Fuel Transition
License Amendment Request
(Non-proprietary Version)

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Brunswick Fuel Transition LAR NRC Meeting

Richland Washington
July 31 – August 2, 2007

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Brunswick Fuel Transition LAR
NRC Meeting

> Meeting purpose

- Provide an overview of the AREVA safety analysis methodology used to support the Brunswick transition
- Describe applicability of the AREVA methodology for Brunswick with operation at EPU conditions
- NRC review of calculation packages documenting Brunswick transition cycle analyses

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Brunswick Fuel Transition LAR Support Plan

Requirement	Product	Product Details	Delivery to NRC
Submit LARs	**Letters (2)**	**Identify TS changes**	**January 22, 2007**
Establish LAR Support Plan	**Table**	**Identify submittals** **Identify schedule**	**April 25 Call** **May 15 Meeting**
Identify SER restrictions and demonstrate compliance	**Document**	**Package existing documentation**	**June 21**
Demonstrate methodology applicable for EPU	**Document**	**Address mixed core and EPU** **Incorporate BF EPU RAI responses**	**July 25 Document** **July 31 Meeting**
Submit plant-specific analyses and demonstrate compliance with design criteria	**Fuel Transition Report Package**	**Package Outline** **TH Report** **LOCA Reports (2)** **Neutronic Report** **Transient Report**	**July 18** **July 18** **July 18** **July 25** **September**
Support NRC Audits			**August 1-2** **September**

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Agenda

- Introduction
- Safety Analysis Methodology
  - Reload Core Design and Analysis Process
  - Safety Analysis Methodology Overview
  - Thermal-Hydraulic Methodology
  - Neutronic Analysis Methodology
  - Stability Analysis Methodology
  - Transient Analysis Methodology
  - LOCA Analysis Methodology
- EPU and Non-EPU Analysis Conditions
- Methodology Applicability for Brunswick

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Safety Analysis Methodology
Presentation Goal

> Provide background information to facilitate discussions with NRC during LAR review
  - General licensing approach for AREVA fuel
  - Reload core design and analysis process
  - Overview of safety analysis methodology
    - Major codes
    - Calculation process
    - Typical cycle-specific calculations

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Reload Core Design and Analysis Process

Mike Garrett
Manager, BWR Safety Analysis

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Reload Core Licensing Approach
Transition Cycle

> AREVA currently is not an NSSS vendor (OEM) for any U.S. BWR

> AREVA currently is the fuel vendor for several U.S. BWRs

> Introduction of AREVA fuel requires confirmation that fuel-related and plant-related design and licensing criteria continue to be satisfied

> AREVA licensing approach and analysis methodology was developed to support the introduction of AREVA fuel into a BWR already licensed for operation in the U.S.

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Reload Core Licensing Approach
Transition Cycle (continued)

- Maintain current plant licensing basis when possible
- Evaluate the introduction of AREVA fuel per the requirements of 10 CFR 50.59
  - Similar to approach used for any plant change
  - Similar to approach used for each reload core design (except for scope)
- Identify plant safety analyses potentially affected by a fuel or core design change
- Assess impact on potentially affected safety analyses and repeat analyses as required

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Reload Core Licensing Approach
Transition Cycle *(continued)*

- Technical Specification changes generally limited to
  - References to NRC-approved methods used to determine thermal limits specified in the COLR
  - MCPR safety limit based on AREVA methods
  - Fuel design description
- COLR thermal limits are determined for the transition core based on analyses using NRC-approved methods

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Major Steps in a Fuel Design Transition

> Data collection
> Develop plant and core models
> Core follow and benchmark analyses (4–5 cycles)
> Compatibility analyses
> **Establish current licensing basis**
> **Disposition of events**
> Plant transition safety analysis
> LOCA analysis
> Core monitoring system parallel operation
> Criticality analyses
> Transition cycle neutronic design
> Initial cycle reload licensing analyses
> Startup and core monitoring data
> Licensing support
> Training

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Reload Core Licensing Approach

> Two steps performed as part of the transition process implement the reload core licensing approach and establish safety analysis methodology requirements
  - Establish current licensing basis
  - Disposition of events

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Reload Core Licensing Approach
Establish Current Licensing Basis

- Licensing basis consists of all analyses performed to demonstrate that regulatory requirements are met
- Licensing basis is defined in documents such as
  - FSAR
  - Technical Specifications
  - Core Operating Limits Reports (COLRs)
  - Technical Requirements Manual
  - Cycle Reload Licensing Reports
  - Extended Operating Domain (EOD) Reports (e.g. increased core flow operation)
  - Equipment Out-of-Service (EOOS) Reports (e.g. feedwater heaters OOS)
  - LOCA Analysis Reports

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Reload Core Licensing Approach
Disposition of Events

> Review all event analyses in the current licensing basis

> Analyses are dispositioned as
  - Not impacted by the change in fuel or core design
  - Bounded by the consequences of another event
  - Potentially limiting — reanalyze using AREVA methodology

> Rated and off-rated conditions considered

> Results from the disposition of events define the safety analyses required for the transition cycle to address the change in fuel and core design

> Disposition of events is documented in calculation notebook and QA reviewed per AREVA procedures

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Transition Cycle Analyses
Typical Disposition Conclusions

> Mechanical design

> Nuclear design
  - Stability
  - Shutdown margins

> Thermal-hydraulic design
  - Hydraulic compatibility
  - MCPR safety limit
  - $MCPR_f$ (slow flow excursion)

> ASME overpressurization

> ATWS
  - Overpressurization
  - Standby liquid control system

Transition Cycle Analyses
Typical Disposition Conclusions

> Criticality analyses
- New fuel storage
- Spent fuel storage

> Anticipated operational occurrences
- Load rejection no bypass
- Turbine trip no bypass
- Loss of feedwater heating
- Control rod withdrawal error
- Feedwater controller failure

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Transition Cycle Analyses
Typical Disposition Conclusions

> Design basis accidents
- Control rod drop accident
- Loss-of-coolant accident
- Fuel handling accident
- Fuel assembly mislocation
- Fuel assembly misorientation

> Emergency operating procedures
- Fuel-dependent input parameters

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Reload Core Licensing Approach
Follow-On Cycle

- Similar to transition core approach but with a reduced scope
- Disposition of events for transition cycle provides basis for analyses typically performed for follow-on reload cores
- All potentially limiting events are reanalyzed or justification provided for continued applicability of previous analysis
- If plant configuration or operational changes are planned during the refueling outage, a cycle-specific disposition of events is performed and additional analyses may be required

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Reload Core Licensing Approach
Summary

> A fuel transition is addressed as a change in the plant design basis that is evaluated relative to the current plant licensing basis

> A systematic approach (disposition of events) is used to identify the impact of the change on the plant safety analyses that constitute the current plant licensing basis

> Potentially impacted safety analyses are reanalyzed with appropriate fuel and core characteristics to ensure that all design and licensing criteria continue to be satisfied

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Reload Core Design and Analysis Process
Key Steps

> Several steps in the core design and analysis process are directed towards ensuring that the planned scope, analysis methods, and input assumptions for the cycle safety analysis are valid

- **Project Initialization (initial reload)**
- Fuel Mechanical Design (initial reload or design change)
- Preliminary Core Design
- **Plant Parameters Document**
- **Fuel Design Analysis Review**
- **Calculation Plan**
- Licensing Basis Core Design
- Safety Analyses
- Design and Licensing Reports
- Fuel Delivery
- Startup Support

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Reload Core Design and Analysis Process
Project Initialization

> A Project Initialization meeting is conducted following finalization of a new or major revision to a contact

> Purpose
  - Inform Engineering and Manufacturing of contractual provisions and schedule
  - Identify any unique product, material, or commercial requirements
  - Establish the need for any qualification or proof-of-fabrication activities

> Any unique engineering methodology, analysis, or reporting requirements should be identified

Reload Core Design and Analysis Process

Plant Parameters Document

- > Defines plant configuration, operating conditions, and equipment performance characteristics used in AREVA safety analyses
- > Provides mechanism for utility to:
  - Review and approve plant parameters used in safety analysis
  - Determine when plant changes will impact safety analyses
  - Notify AREVA of planned plant changes during the next refueling outage
- > AREVA requests PPD updates for upcoming cycle (generally, a draft PPD with known changes is provided)
- > Utility confirms or identifies PPD changes for upcoming cycle
- > AREVA reviews PPD changes and performs a disposition to identify any additional analyses required
- > Ensures that AREVA and utility have a mutual agreement on the plant configuration and operation basis used in safety analyses

Reload Core Design and Analysis Process
Fuel Design Analysis Review

> Primary purpose of the Fuel Design Analysis Review is to ensure that all analyses required to demonstrate compliance with design and licensing criteria are identified in the Calculation Plan

> Review includes
  - Review design and identify appropriate criteria
  - Review open issues in Correspondence Activity Tracking System
  - Identify analyses required to demonstrate compliance with criteria
  - Review methodology applicability and SER restrictions

> Preliminary Calculation Plan should be available prior to Review

> For initial reload, Review should be performed after completion of licensing basis determination and disposition of events

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Reload Core Design and Analysis Process
Calculation Plan

> Defines the scope of the safety analyses to be performed for a specific reload including any additional analyses required due to PPD changes

> Provides cycle-specific reference identifying analyses to be performed, associated methodology, and key assumptions

> AREVA provides draft calculation plan identifying all analyses to be performed for the cycle

> Following utility review and comment, final calculation plan is issued by AREVA

> Assures that the work scope and analysis bases are understood and acceptable to all parties

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Reload Core Design and Analysis Process
Summary

> The AREVA core design and analysis process has procedurally controlled steps to ensure that the scope of safety analyses and applied methodology are appropriate to demonstrate that all design and licensing criteria are satisfied for the reload core design

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Safety Analysis Methodology Overview

Mike Garrett

Manager, BWR Safety Analysis

Safety Analysis Methodology
Goals

> Perform analyses of anticipated operational occurrences (AOOs) to confirm or establish operating limits that:
  - Adequately protect all fuel design criteria
  - Ensure all licensing criteria are satisfied
  - Promote economically efficient fuel cycles
  - Provide operational flexibility

> Perform analyses of design basis accidents to confirm that results are within regulatory acceptable limits

> Perform analyses of special events to ensure regulatory requirements or industry codes are satisfied

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Safety Analysis Methodology

> Safety analyses include
  - Anticipated operational occurrence (AOO) analyses
  - Accident analyses
  - Special event analyses

> Safety analysis methodology includes
  - Thermal-hydraulic analysis methodology
  - Neutronic analysis methodology
  - Stability analysis methodology
  - Transient analysis methodology
  - LOCA analysis methodology

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AOO Analyses
Typical Events and Applied Methodology

Events	Methodology
• Control rod withdrawal error • Loss of feedwater heating	Neutronic Methodology
• Load rejection without bypass • Turbine trip without bypass • Feedwater controller failure	System Transient Methodology
• Recirculation flow runup • Safety limit MCPR	Thermal-Hydraulic Methodology

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Accident Analyses
Typical Events and Applied Methodology

Events	Methodology
• Loss-of-coolant accident	LOCA Methodology
• Control rod drop accident • Fuel assembly loading accident • Fuel handling accident	Neutronic Methodology

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Special Analyses
Typical Events and Applied Methodology

Events	Methodology
• Shutdown margin analysis • Standby liquid control analysis • Stability	Neutronics Methodology
• ASME overpressurization analysis • ATWS overpressurization analysis	System Transient Methodology

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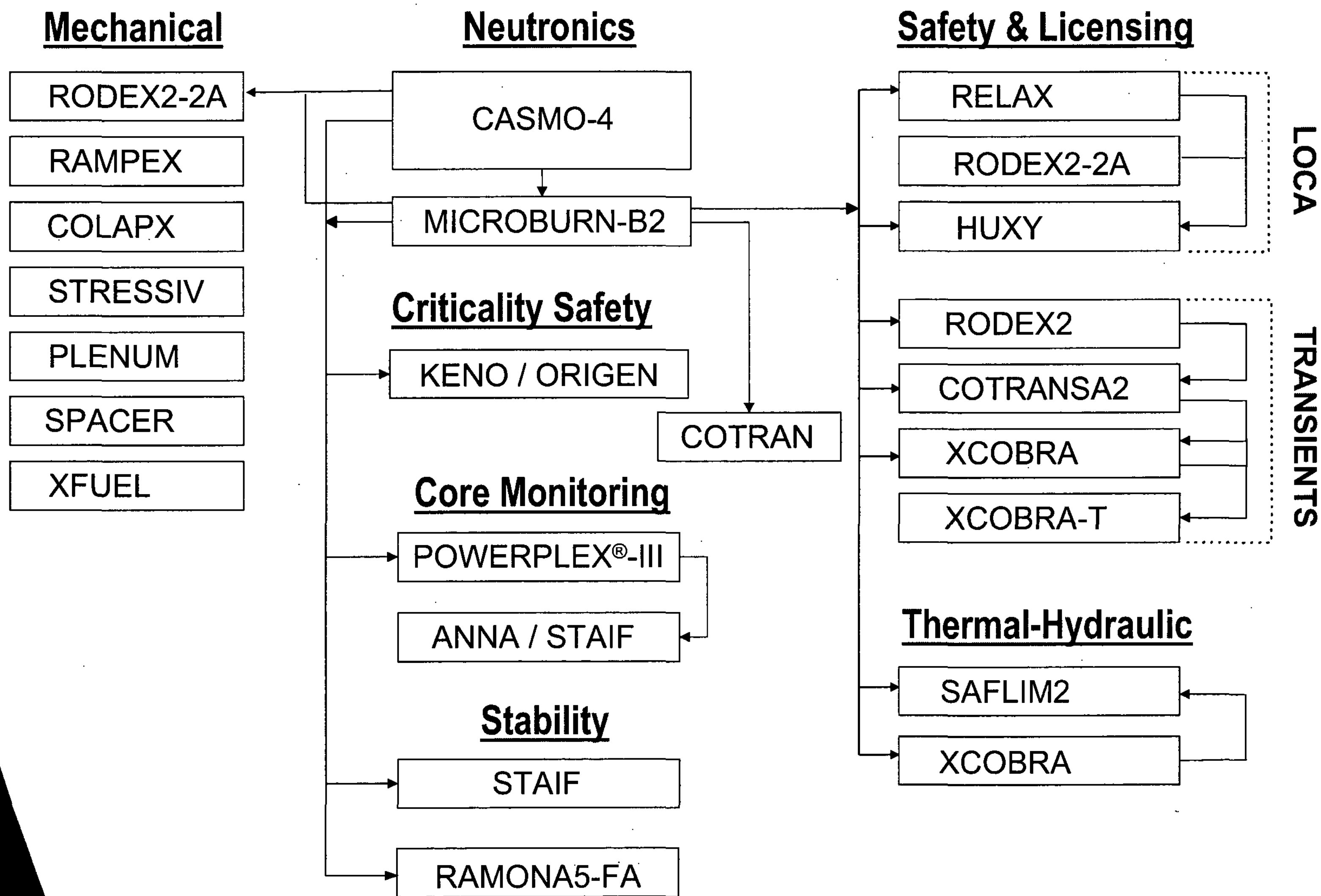
Safety Analysis Methodology



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Thermal-Hydraulic Methodology

Darrell Carr
Team Leader, BWR Safety Analysis

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Thermal-Hydraulic Analysis Methodology

> Thermal-hydraulic analysis methodology description

> Thermal-hydraulic compatibility

> Flow-dependent MCPR analysis

> Critical power correlation

> Safety Limit MCPR Methodology

Thermal-Hydraulic Analysis Methodology
XCOBRA Computer Code

Description	XCOBRA predicts the steady-state thermal-hydraulic performance of BWR cores at various operating conditions and power distributions
Use	Evaluate the hydraulic compatibility of fuel designs. Evaluate core thermal-hydraulic performance (core pressure drop, core flow distribution, bypass flow, MCPR, etc.)
Documentation	XN-NF-CC-43(P), *XCOBRA Code Theory and User's Manual*
Acceptability	XN-NF-80-19(P)(A) Volume 3 Rev 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description,* January 1987
	NRC accepts the use of XCOBRA based on the similarity of the computational models to those used in the approved code XCOBRA-T

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XCOBRA Computer Code
Major Features

- Represents the core as a collection of parallel flow channels
- Each flow channel can represent single or multiple fuel assemblies as well as the core bypass region
- Core flow distribution is calculated to equalize the pressure drop across each flow channel
- Pressure drop in each channel is determined through the use of the AREVA thermal-hydraulic methodology
- Input includes fuel assembly geometry, pressure drop coefficients, and core operating conditions
- Water rods (or channels) can be explicitly modeled
- Calculates the flow and local fluid conditions at axial locations in each channel for use in evaluating MCPR

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XCOBRA Computer Code

Physical phenomena modeled

- Friction pressure drop
- Elevation pressure drop
- Two-phase pressure drop mult. factors
- Void/quality relationship
- Local (irreversible) pressure drop
- Acceleration pressure drop
- Subcooled boiling
- Core leakage (bypass)

Local losses

- Inlet orifice
- Lower tie plate
- Spacers
- Upper tie plate

Leakage (core bypass) paths

- Core support plate
- Lower tie plate flow holes
- Channel - lower tie plate seal
- Fuel support - nozzle interface

XCOBRA Computer Code

> Empirically derived hydraulic characteristics

> Reynolds number dependent hydraulic characteristics for a number of fuel assembly pressure losses must be empirically determined

- Bare rod friction loss
- Spacer loss coefficient
- Upper tie plate loss coefficient
- Orifice/lower tie plate loss coefficient

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XCOBRA Computer Code
Applications

- Thermal-hydraulic compatibility evaluations
- Flow-dependent MCPR limits
- Flow distribution
  - Pressurization transient analysis
  - Safety limit MCPR

Thermal-Hydraulic Compatibility

- Approved thermal-hydraulic criteria described in ANF-89-98 (P)(A) Rev 1
- Hydraulic compatibility
- Thermal margin performance
- Rod bow
- Bypass flow

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Thermal-Hydraulic Compatibility
Criteria

> Hydraulic compatibility

- The hydraulic resistance of the reload fuel assemblies shall be sufficiently similar to the existing fuel in the reactor such that there is no significant impact on the core flow or the flow distribution among assemblies in the core
- (For example, the flow resistance of the AREVA fuel should not be so low as to produce an unwarranted flow penalty and associated thermal margin reduction for existing fuel in the reactor)

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Thermal-Hydraulic Compatibility
Criteria

> Thermal margin performance

- Fuel assembly geometry, including spacer design and rod-to-rod local peaking, should minimize the likelihood of boiling transition during normal reactor operation as well as during AOOs. The fuel design shall fall within the bounds of the applicable empirically based boiling transition correlation approved for AREVA reload fuel and coresident fuel. Within other applicable mechanical, neutronic, and fuel performance constraints, the fuel design should achieve good thermal margin performance.

Thermal-Hydraulic Compatibility
Criteria

> Rod bow

- The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margin requirements

> Bypass flow

- 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.

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Brunswick Thermal-Hydraulic Compatibility Analysis

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Thermal-Hydraulic Compatibility
Representative Results

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Brunswick Thermal-Hydraulic Compatibility
Conclusion

> The approved design criteria associated with the thermal-hydraulic compatibility for Brunswick Unit 1 have been met

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Flow-Dependent MCPR ($MCPR_f$) Analysis

> $MCPR_f$ limit is established to provide protection against fuel failures during a slow core flow excursion (i.e., SLMCPR is not violated during the event)

> Analysis assumes core flow increases to the maximum physically attainable value

> Limit is a function of initial core flow; a larger core flow increase (and resulting power increase) occurs from reduced core flow

> XCOBRA computer code used to calculate change in CPR

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MCPR$_f$ Analysis Process

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MCPR$_f$ Analysis
Representative Results

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Critical Power Correlation

- Cycle-specific licensing calculations require calculation of MCPR limits
- Requirements include licensing and monitoring CPR of all assemblies in the core, including those supplied by other fuel vendors

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SPCB Critical Power Correlation
Description

> SPCB is an empirical correlation of measured critical heat flux in a fuel assembly

> The correlation predicts critical heat flux at the axial plane of interest

> The correlation is a function of:

- Pressure
- Flow
- Enthalpy
- Local peaking distribution
- Assembly geometry (flow area, surface area, and spacer design)

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Input Required to Calculate MCPR

- Bundle Power
- Bundle Flow
- Inlet Enthalpy
- Core Pressure
- Axial Power Shape
- F-eff (*)

→ SPCB → MCPR

(*) F-eff characterizes the local power peaking and flow distribution within the fuel assembly

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SPCB Critical Power Correlation
Development

- Base correlation initially derived from AREVA database for ATRIUM™-9B and ATRIUM-10
- Base correlation is a function of thermodynamic parameters
- Non-uniform axial tests support non-uniform axial correction factor
- Effects due to local peaking and spacer design were observed as separable and are accounted for by a correction factor

SPCB Critical Power Correlation
Correction Factor

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SPCB Critical Power Correlation
F-eff

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CPR Correlation for Coresident Fuel

> AREVA does not typically have access to CPR correlation for coresident fuel

> Approved methodology for applying AREVA critical power correlations to coresident fuel is described in EMF-2245(P)(A)
  - Process can be applied to any approved AREVA critical power correlation
  - Direct method
  - Indirect method

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CPR Correlation for Coresident Fuel
Brunswick Application - Indirect Method

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MCPR Safety Limit Methodology

> NRC-Approved Topical Report

- ANF-524(P)(A) Rev 2 and Supplements 1 and 2, *ANF Critical Power Methodology for Boiling Water Reactors*, Advanced Nuclear Fuels Corporation, November 1990

- This report includes the MCPR calculational procedure, identifies the fuel and non-fuel related uncertainties and the statistical process used to determine a MCPR safety limit that protects 99.9% of the fuel rods in the core from boiling transition

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MCPR Safety Limit Methodology

> The purpose of the safety limit MCPR (SLMCPR) is to protect the core from boiling transition (BT) during both normal operation and anticipated operational occurrences (AOOs)

> At least 99.9% of the fuel rods in the core are expected to avoid BT when the minimum CPR during the transient is greater than the SLMCPR

> The SLMCPR is determined by a statistical convolution of uncertainties associated with the calculation of MCPR

> The SLMCPR analysis is performed each cycle using core and fuel design specific characteristics

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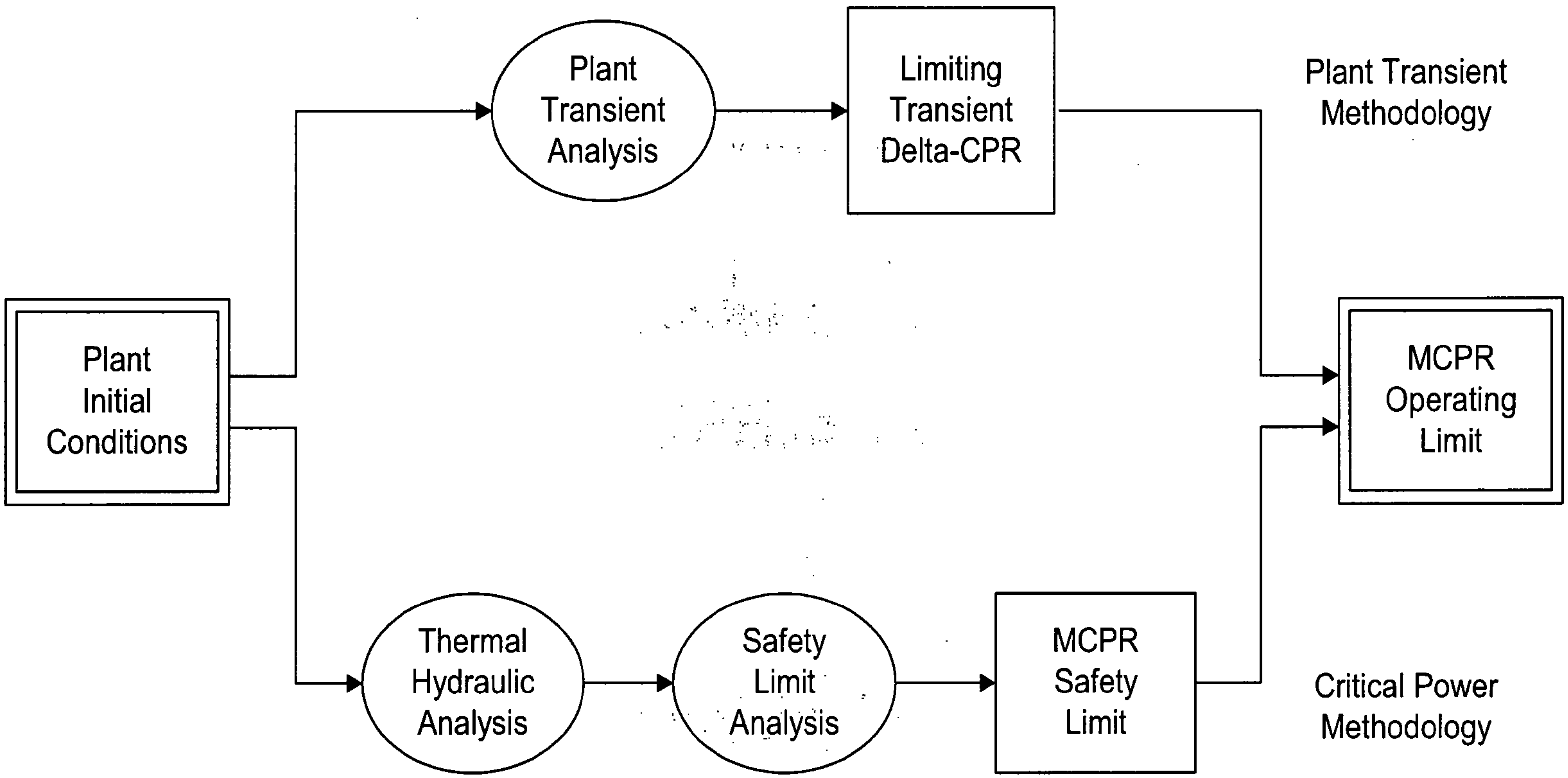
Thermal Limits Process



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MCPR Safety Limit Methodology

> Application of Methodology

- The cycle-specific application of the NRC-approved methodology is controlled by
  - An implementing guideline which provides instructions to the engineers who perform and review the calculation
  - Automation which has been developed to perform all data manipulation between codes

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Safety Limit MCPR
Computer Codes

- MICROBURN-B2
  - EMF-2158(P)(A) Rev 0
  - Provides the radial peaking factor and exposure for each bundle in the core and the core average axial power shape
- CASMO-4
  - EMF-2158(P)(A) Rev 0
  - Provides the local peaking factor distribution for each fuel type
- XCOBRA
  - XN-NF-80-19(P)(A) Volume 3 Rev 2
  - Provides hydraulic demand curve for each fuel type
- SLPREP
  - Automation code which gathers neutronic data from MICROBURN-B2 and CASMO-4 and prepares SAFLIM2 input
- SAFLIM2
  - ANF-524(P)(A) Rev 2
  - EMF-2392(P), *SAFLIM2 Theory, Programmer's, and User's Manual*
  - Calculates the fraction of rods in boiling transition for a specified SLMCPR

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MCPR Safety Limit Methodology
SLMCPR Analysis Process

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MCPR Safety Limit Methodology
SLMCPR Analysis Process

SLMCPR Analysis Methodology

Monte Carlo Technique

> A Monte Carlo analysis is a statistical technique to determine the distribution function of a parameter that is a function of random variables

- Each random variable is characterized by a mean, standard deviation, and distribution function
- A random value for each input variable is selected
- The parameter of interest is calculated using the random values for the input variables
- The process is repeated a large number of times to create a probability distribution for the parameter of interest

SAFLIM2 Computer Code
Major Features

- Convolution of uncertainties via a Monte Carlo technique
- Consistent with POWERPLEX® CMSS calculation of MCPR
- Each nominal input is randomly perturbed based on its uncertainty
- Appropriate critical power correlation used directly to determine if a rod is in boiling transition (deterministic)
- BT rods for all bundles in the core are summed
- Non-parametric tolerance limits used to determine the number of BT rods with 95% confidence
- Explicitly accounts for channel bow
- New fuel designs easily accommodated

SAFLIM2 Computer Code
Calculation Procedure

> Initialization

> Monte Carlo Trials

- Core Calculations (Outer Loop)
- Fuel Assembly Calculations (Inner Loop)

> Rods in BT Calculation

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SAFLIM2 Computer Code
Initialization

> Establish initial (nominal) operating conditions at which the core MCPR equals the desired SLMCPR

> Initial conditions are required for the following parameters

- Core flow
- Core pressure
- Feedwater temperature
- Feedwater flow
- Core inlet enthalpy
- Core power
- Assembly power (radial peaking)
- Core average axial power shape
- Assembly flow
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SAFLIM2 Computer Code
Initialization *(continued)*

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SAFLIM2 Computer Code
Core Calculations – Outer Loop

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SAFLIM2 Computer Code
Assembly Calculations – Inner Loop

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SAFLIM2 Computer Code

Fuel Rod Calculations - Inner Loop

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SAFLIM2 Computer Code

Number of Rods in BT

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Safety Limit Methodology

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SAFLIM2 Computer Code

Reactor System Uncertainties

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SAFLIM2 Computer Code

Core Monitoring Uncertainties

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SAFLIM2 Computer Code
Fuel Design Uncertainties

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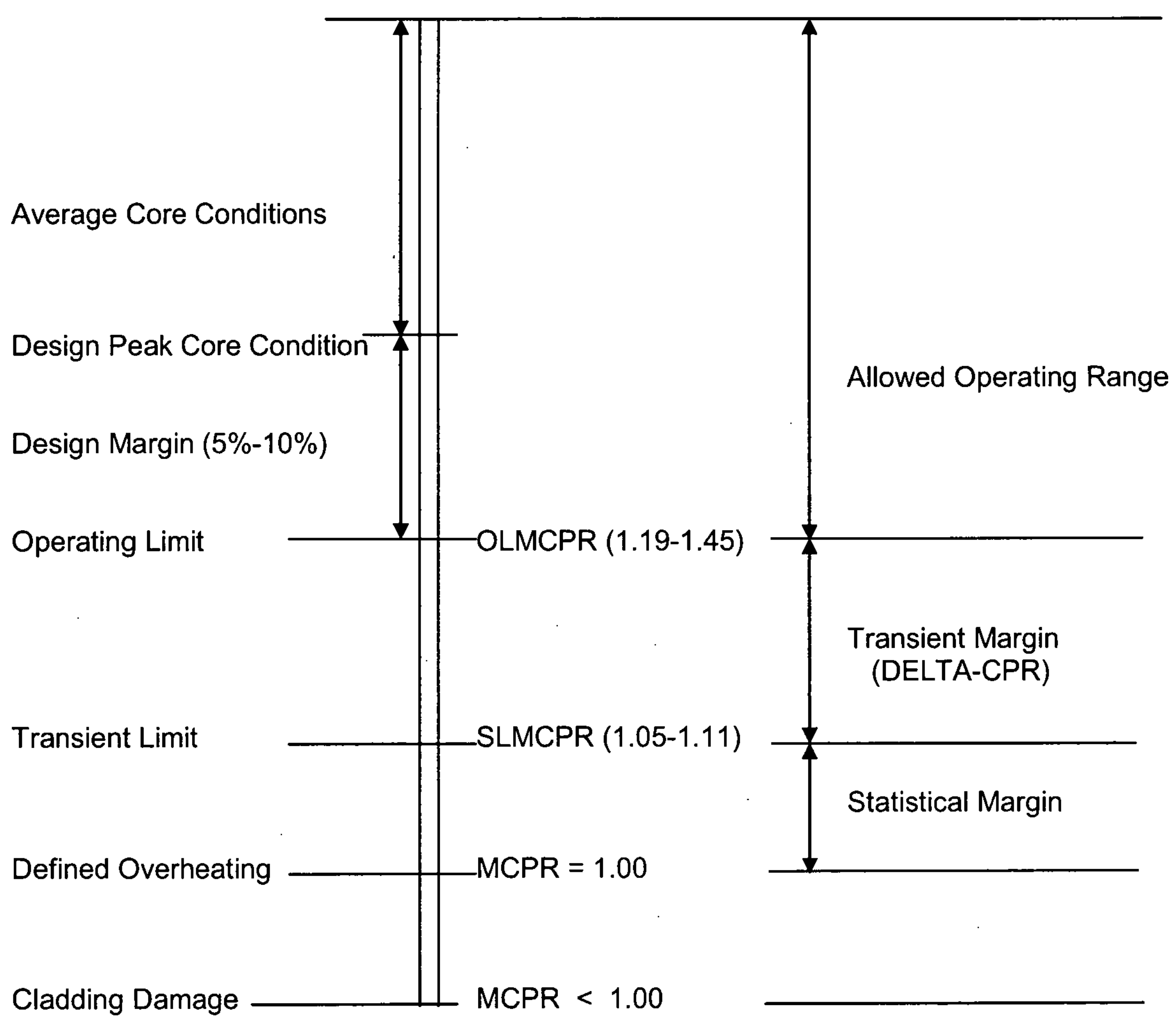
Thermal Limits Methodology



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Safety Limit Analysis

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Safety Limit Analysis

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Safety Limit MCPR Results
Brunswick Unit 1 Cycle 17

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> Analysis results support a 1.11 SLMCPR for two-loop operation

> Analysis results support a 1.12 SLMCPR for single-loop operation

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Neutronic Analysis Methodology

Ken Hartley
Team Leader, BWR Neutronics

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Neutronic Analysis Methodology
Major Computer Codes

Code	Use
CASMO-4	Performs fuel assembly burnup calculations and calculates nuclear data for MICROBURN-B2
MICROBURN-B2	Performs 3-dimensional steady-state reactor core neutronic analyses for assessing impact on thermal limits during localized and quasi-steady-state events

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Neutronic Analysis Methodology
CASMO-4 Computer Code

Description	Multi-group, 2-dimensional transport theory code
Use	Performs fuel lattice burnup calculations and generates nuclear data for use in MICROBURN-B2
Documentation	EMF-2158(P)(A) Rev 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 /MICROBURN-B2*, October 1999
Acceptability	The safety evaluation by the NRC for the topical report EMF-2158(P)(A) approves the CASMO-4/ MICROBURN-B2 methodology for licensing applications

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Neutronic Analysis Methodology
MICROBURN-B2 Computer Code

Description	A 3-dimensional, two group, diffusion theory code incorporating microscopic depletion and pin power reconstruction
Use	Performs 3-dimensional steady-state reactor core neutronic analyses for assessing impact on thermal limits during localized and quasi-steady-state events
Documentation	EMF-2158(P)(A) Rev 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, October 1999
Acceptability	The safety evaluation by the NRC for the topical report EMF-2158(P)(A) approves the CASMO-4/ MICROBURN-B2 methodology for licensing applications

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Neutronic Topical Report SER Restrictions

> EMF-2158(P)(A) (CASMO-4/MICROBURN-B2)

1. The CASMO-4/MICROBURN-B2 code system 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

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Neutronic Topical Report SER Restrictions

> EMF-2158(P)(A) (CASMO-4/MICROBURN-B2)

5. The CASMO-4/MICROBURN-B2 code system is approved as a replacement for the CASMO-3/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 1-4 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

Conformance to No. 1 is addressed through benchmarking the code system against actual operation of previous cycles.

Neutronic Code Input Flow
CASMO-4/MICROBURN-B2

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Neutronic Fuel Cycle Design

> Nuclear fuel assembly design attributes
  - Lattice and assembly reactivity
  - Enrichment distribution
  - Local peaking distribution
  - CPR correlation factors
  - Axial enrichment/gadolinia zoning, natural blankets

> Lattice calculations performed with CASMO-4

> In-core simulations performed with MICROBURN-B2

Neutronic Fuel Cycle Design

- > Constraints on Nuclear Fuel Design due to Mechanical Criteria
  - Limitations on LHGR for PCI, fission gas release, cladding strain
  - Limitations on discharge burnup
- > Licensing and safety constraints
  - MAPLHGR and MCPR limits

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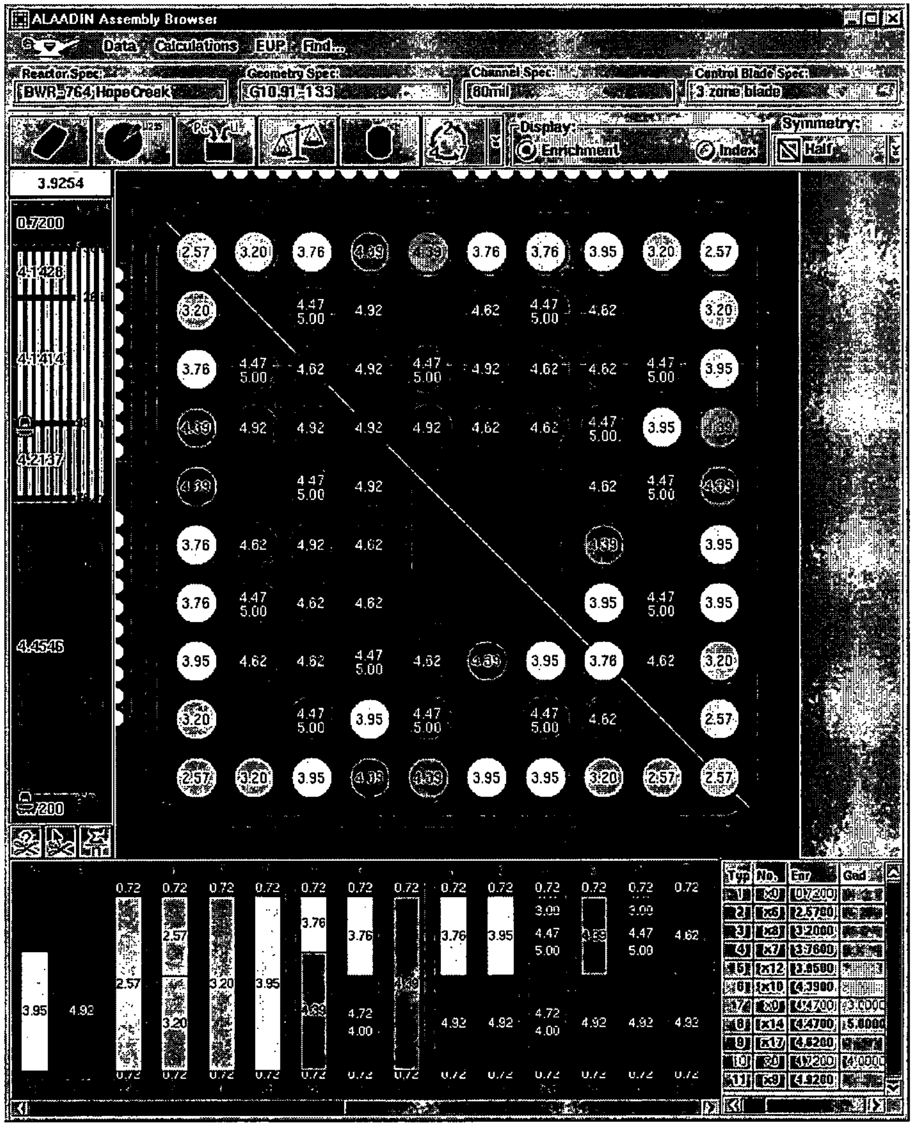
Nuclear Design
ALAADIN Graphical User Interface

Assembly Browser Interface



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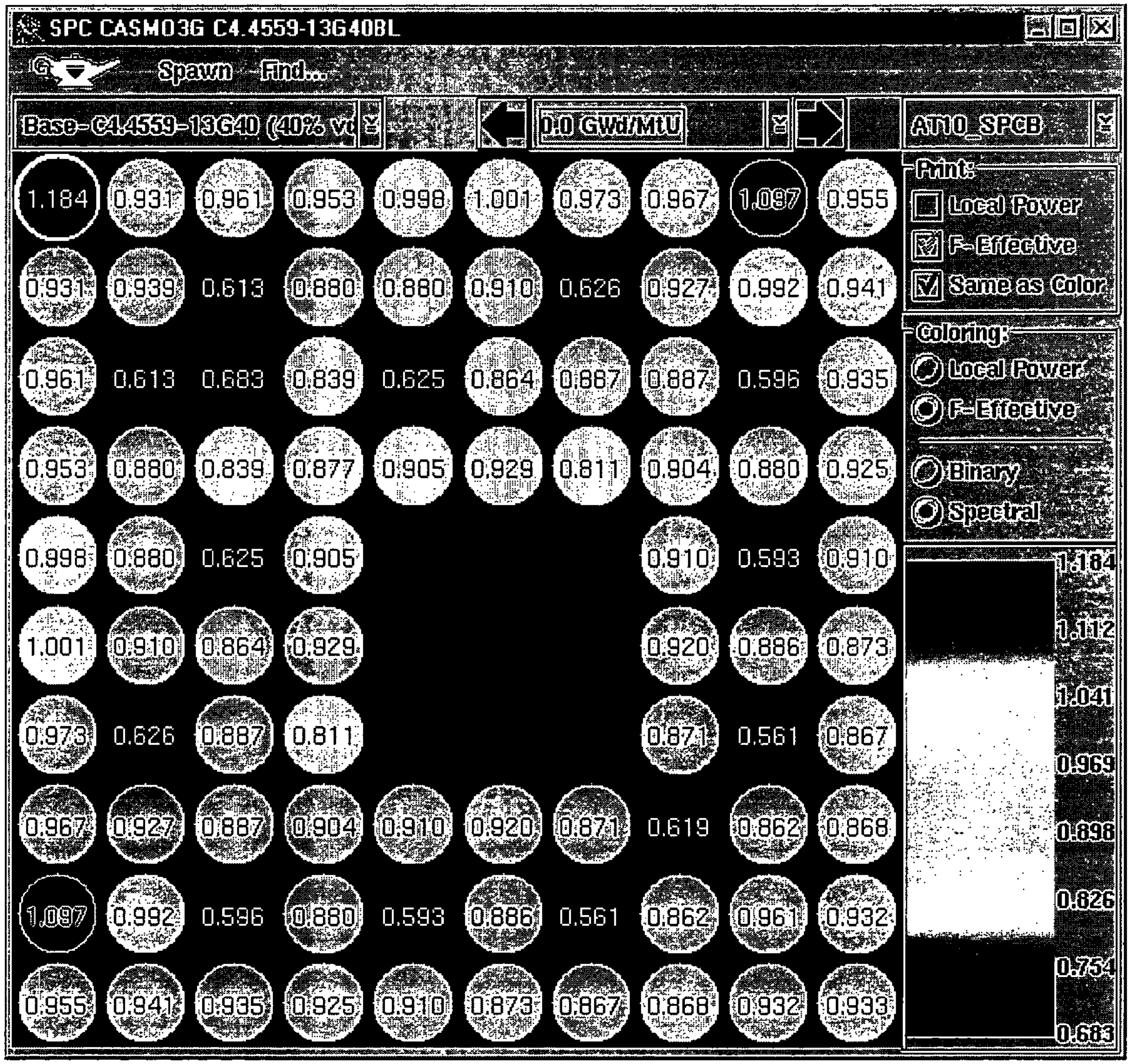
ALAADIN Graphical User Interface *(continued)*

Assembly Map Inspector -- F-eff



Gadolinia Design

- Gadolinia design is determined based both on CASMO-4 results and in-core performance via MICROBURN-B2
- Position of gadolinia rods determined by acceptable local peaking and CPR factor results
- BOC hot excess reactivity and cold shutdown margin calculations to determine number of gadolinia rods
- Peak reactivity calculations in the cycle to determine concentration
- [

  ]

AREVA

Bundle Enrichment Design

> Enrichment level set based on energy requirement

> [

]

> Multiple fresh sub-batches used when needed to smooth out reactivity distribution in the core

NONPROPRIETARY

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Fuel Loading Plan Design

> Iterative process to determine optimal placement of fresh and irradiated assemblies

> Fresh batch size and enrichment determined to meet cycle energy requirement

> Constraints based on thermal and reactivity margins, channel management, fuel conditioning from previous cycle operation, and licensed fuel burnup limits

> For mixed cores, coresident other vendor fuel limits must also be protected

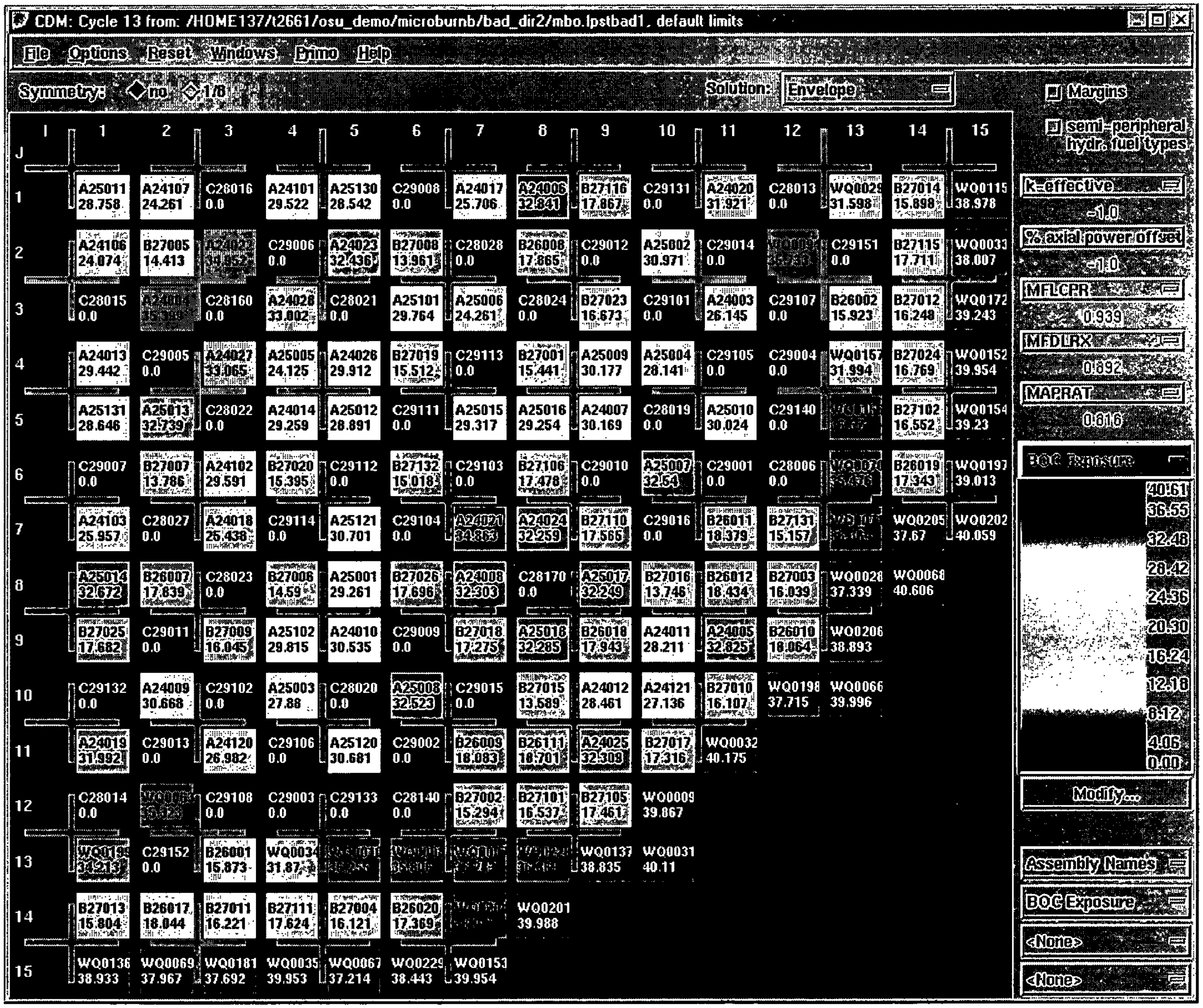
> MICROBURN-B2 core simulator analyses used to set the loading plan

CDM (Core Design Module) Graphical Interface

Loading Pattern

Colored by BOC Exposure

(Invalid Loading)



AREVA

Target Rod Pattern Design

> The step-through depletion is used for detailed analysis of the cycle operational capability

> It provides the best estimate of realistic exposure, void, and control history which is to be expected for the cycle

> It provides confirmation that the core as designed can be operated with margin to assumed technical specification limits

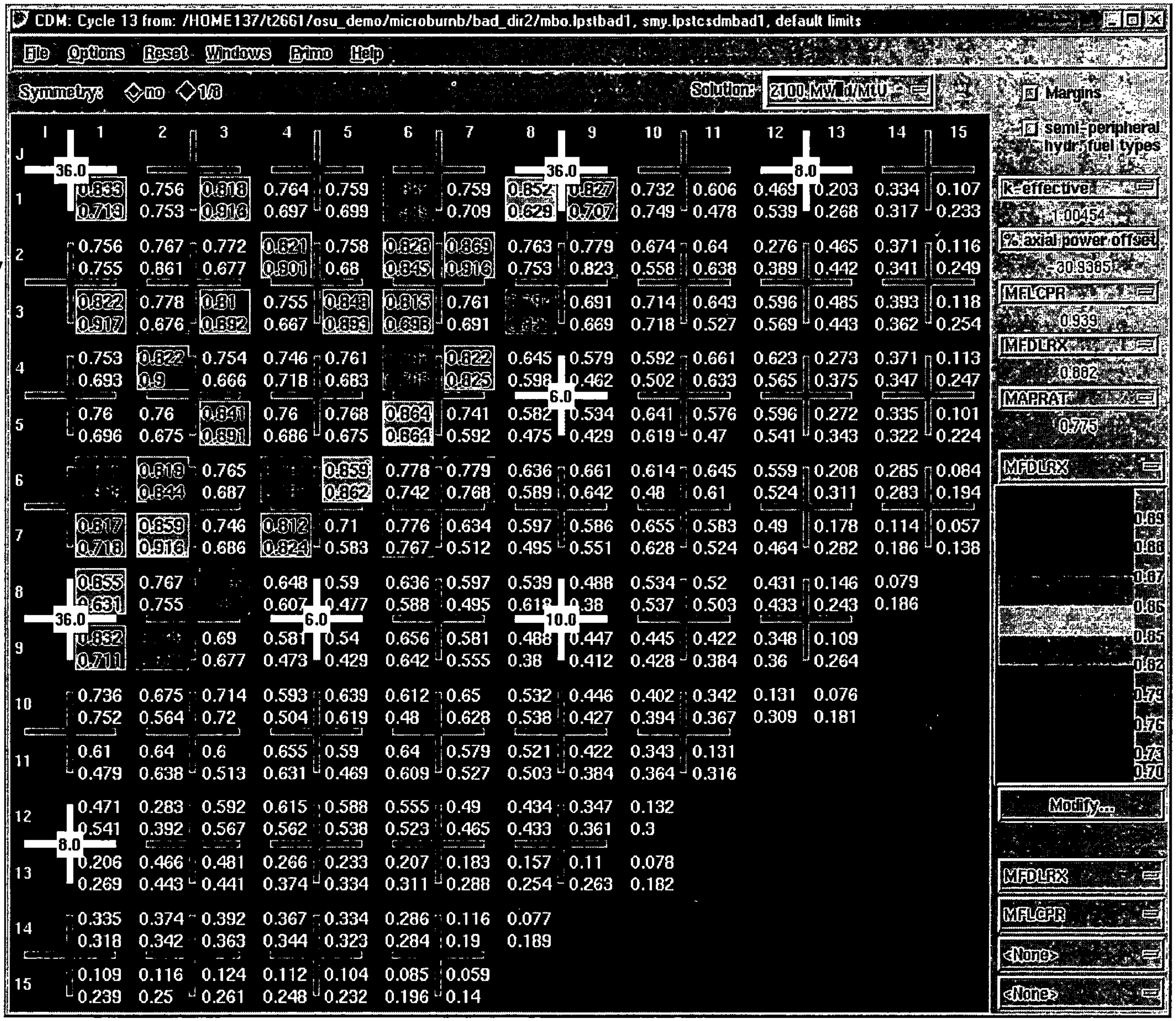
AREVA

CDM (Core Design Module) Graphical Interface - - Step Through Design

Thermal Margin Results

Colored by Limiting LHGR Margin

(Invalid Step Out Pattern)



NONPROPRIETARY

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Neutronic Safety Analysis

> Once the fuel cycle design has been accepted by the customer, licensing of the core can commence

> Approved methods generally defined in XN-NF-80-19(P)(A) or subsequently submitted and approved Topical Reports

- XN-NF-80-19(P)(A) Vol 1 Supplements 1 and 2, and Vol 4

(No specific SER restrictions listed)

Neutronic Safety Analysis Methodology
Cycle-Specific Analyses

> Items Analyzed With CASMO-4/MICROBURN-B2
  - Cold shutdown margin
  - Standby boron liquid control shutdown margin
  - Control rod withdrawal error
  - Loss of feedwater heating
  - Control rod drop accident
  - [

]

NONPROPRIETARY

Neutronic Safety Analysis Methodology
Cycle-Specific Analyses

> Items Analyzed With CASMO-4/MICROBURN-B2 *(continued)*

- Core flow increase event ($LHGR_f$)
- Neutronic input provided to BWR-SA for
  - SLMCPR
  - $MCPR_f$
  - Fast transient analyses
  - LOCA

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Neutronic Safety Analysis Methodology
Cycle-Specific Analyses

> Other Neutronic analyses
- Fuel storage criticality (*) (CASMO, KENO V.a)
- Fuel handling accident (*) (ORIGEN-S)
- Reactor core stability (STAIF)

* Cycle-specific confirmation that analysis remains bounding

Cold Shutdown Margin and Standby Liquid Control Shutdown Margin

> Cold shutdown margin (CSDM) - Evaluation of core reactivity at cold conditions with the analytically determined strongest control rod fully withdrawn, all other rods fully inserted. Must meet 0.38% $\Delta k/k$ technical specification requirement for subcriticality. *(Evaluated on a cycle-specific basis with MICROBURN-B2.)* Conditions analyzed are:

- Isothermal 68 °F, no Xenon
- Exposures throughout the cycle

> Standby boron liquid control (SLC) shutdown margin - Reactivity control by injection of boron in the moderator (720 ppm B for Brunswick). Must be able to render the core subcritical in event control rods become inoperable. *(Evaluated on a cycle-specific basis with MICROBURN-B2.)* Conditions analyzed are:

- [

]

- Analyzed at exposures throughout the cycle

Control Rod Withdrawal Error

> Control rod withdrawal error (CRWE) - Inadvertent withdrawal of a control rod at power until it is stopped by the Rod Block Monitor (RBM) on BWR/3–5 plants or the Rod Withdrawal Limiter (RWL) on BWR/6 plants (ΔCPR protected by the MCPROL selected for the cycle). *Conditions of the analysis:*

- For BWR/3–5 plants - analyzed on a cycle-specific basis at BOC and peak reactivity exposure for the cycle with MICROBURN-B2
- Analyzed at quasi-steady-state conditions, e.g. as a series of steady solutions as the error rod is withdrawn
- RBM setpoints corresponding to a range of MCPROLs determined
- Generic Topical approved for BWR/6 plants

AREVA

Stability

> For Brunswick, Long Term Stability Solution (LTS) Option III is used
  - MCPR operating limits versus OPRM setpoints are determined

> Backup Stability Protection analyses are provided - Avoidance of core power oscillations by assessment of core loading (decay ratios) with the NRC-approved STAIF code. For Brunswick, scram regions and exclusion regions are determined consistent with BWROG guideline OG02-0199-260.

> BWR core stability is sensitive to fuel rod thermal time constant, void coefficient, bundle 2- to 1-phase pressure drop, core power distribution, and operating point on the P/F map. (*Methodology capable of supporting interim corrective actions (ICAs), exclusion Z-region, and long-term solutions, e.g., 1D, E1A, and Option III.)*

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Loss of Feedwater Heating

- Loss of feedwater heating (LFWH) - A loss of feedwater heating capability due to the closing of a steam extraction line or the bypassing of feedwater flow around a heater, causing insertion of reduced temperature water into the core at power, i.e., reactivity insertion
- ΔCPR protected by the MCPROL selected for the core. *(Generic Topical approved,* ANF-1358(P)(A) Rev 3*)*

LFWH Topical Report SER Restrictions

> ANF-1358(P)(A) Rev 3 (Loss of Feedwater Heating)

1. The methodology applies to BWR/3–6 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) Rev 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) Rev 0 and comply with the guidelines and conditions identified in the associated NRC SER.

LFWH Topical Report SER Restrictions

> ANF-1358(P)(A) Rev 3 (Loss of Feedwater Heating)

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

If core conditions are outside any of the SER restrictions for cycle exposure, ratio of rated power and rated steam generation rate, rated feedwater temperature, or change in feedwater temperature, cycle specific analyses are performed.

Control Rod Drop Accident

> The Control Rod Drop Accident (CRDA) is a postulated reactivity insertion accident (RIA). A control rod is assumed to become decoupled from the rod drive mechanism during rod withdrawal and is assumed to remain fully inserted as the drive mechanism moves to a new axial location. At some point later in the withdrawal sequence, when the remaining control rods are in a configuration that maximizes the worth of the fully inserted decoupled blade, the decoupled blade slips free and falls to the drive mechanism location. The event results in a sudden reactivity addition causing a localized rapid increase in power.

> CRDA acceptance criteria are:
- a deposited enthalpy less than 280 cal/gm at any axial location in any fuel rod
- maximum reactor pressure during any portion of the accident should be less than the value that will cause stresses to exceed "Service Limit C" as defined in the ASME code
- the number of failed fuel rods shall be sufficiently small to remain within the accepted radiological consequences for the site

> The 280 cal/gm limit is the threshold at which rapid expulsion of the fuel can occur. The fuel rod failure threshold is 170 cal/gm.

> Parametric analysis as discussed in XN-NF-80-19(P)(A) Vol 1 Supplements 1 and 2

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Control Rod Drop Accident (continued)

> The severity of the event is determined based on the following parameters as determined from CASMO-4 and MICROBURN-B2 analysis:
  - Doppler coefficient (αD)
  - Delayed Neutron Fraction (βeff)
  - Control Rod Worth
  - Four-Bundle-Local-Peaking factor (P4BL)

> The CRDA is conservatively analyzed at hot-zero-power conditions (isothermal temperature, non-voided, and xenon free) at high reactivity cycle exposures.

> Additional conservatism is incorporated by assuming that adiabatic conditions remain during the power excursion (i.e., no direct moderator heating is credited during the analysis), and that the reactor remains at hot zero power conditions for the entire withdrawal sequence

> Deposited enthalpy for each candidate rod drop is determined by applying the XN-NF-80-19(P)(A) parameterization using the four parameters listed above

AREVA

Fuel Assembly Loading Error

- For BWRs, there are two types of fuel assembly loading errors, the mislocation of an assembly into an unintended core location, and the misorientation of the assembly with respect to the control blade corner. It is assumed that the error goes unnoticed and the cycle operates with the misloaded assembly.
- Operating the cycle with an undiscovered fuel assembly in an improper core location results in changes in the core power distribution and potentially an increase in the local power density.
- An increase in the local power density will lower the minimum critical power ratio (MCPR) and increase the linear heat generation rate (LHGR) of the mislocated fuel assembly.
- The fuel assemblies in the cells surrounding the mislocated fuel assembly are also impacted.

AREVA

Fuel Assembly Loading Error (continued)

> Per Section 15.4.7 of NUREG-0800, plant operating procedures and design features minimize the likelihood of fuel assembly loading errors. Nevertheless, analyses are performed to demonstrate that in the event a fuel loading error is not detected and a fuel assembly is operated for the entire cycle in an improper position, the offsite dose consequences would be no more than a small fraction (≤10%) of the offsite dose (10 CFR Part 100 or 10 CFR 50.67 as applicable).

> The fuel rod failure mechanisms that are important in the fuel assembly mislocation accident are overheating of the clad and overheating of the fuel pellets.

> [

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NONPROPRIETARY

AREVA

Fuel Assembly Loading Error (continued)

> Operating the cycle with an undiscovered fuel assembly in an improper orientation with respect to the control blade corner results in changes in the local peaking distribution in the misrotated assembly. Misrotations of 180 degrees with respect ot the control blade corner are the most limiting.

> The misorientation results in interference of the channel fastener at the top of the channel and the upper core support grid. The interference causes the assembly to lean toward the control blade corner. The lean of the assembly skews the out-channel water gaps leading to a skewed local peaking distribution within the assembly.

> The increase in the out-channel water gap on two sides of the assembly leads to increased local peaking density in the rods near the increased water gaps, which lowers the minimum critical power ratio (MCPR) and increases the linear heat generation rate (LHGR), of the misoriented fuel assembly. The fuel assemblies in the cells surrounding the misoriented fuel assembly are also impacted (effective change in water gap between the assemblies).

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Fuel Assembly Loading Error (continued)

- Again, analyses are performed to demonstrate that in the event a fuel loading error is not detected and a fuel assembly is operated in an improper orientation throughout the cycle, the offsite dose consequences would be no more than a small fraction (≤10%) of the offsite dose (10 CFR Part 100 or 10 CFR 50.67 as applicable).
- The fuel rod failure mechanisms that are important in the fuel assembly misorientation accident are overheating of the clad and overheating of the fuel pellets.
- [

]

NONPROPRIETARY

Core Flow Excursion Event

> Boiling Water Reactors (BWRs) which operate within an extended operating domain of the power flow map, e.g., MEOD and MELLA plants, or those which have implemented ARTS use a flow-dependent linear heat generation rate limit multiplier, $LHGRFAC_f$ or $MAPFAC_f$.

> This flow-dependent limit multiplier protects the fuel from exceeding mechanical criteria in the event of an Anticipated Operational Occurrence (AOO) which results in a flow excursion. The limit set-down (enforced by multipliers < 1.0) prevents the plant from operating at an initial LHGR or MAPLHGR large enough such that at the end of the excursion, criteria would be violated.

> It has been established that a slow recirculation flow runout produces the most limiting LHGR increase among all envisioned flow excursion scenarios.

> [

]

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Core Flow Excursion Event *(continued)*

> [

]

> Based on the LHGR or MAPLHGR increase, flow-dependent multipliers are determined to protect
>
> - Protection Against Power Transient (PAPT) criteria (AREVA fuel)
> - MOP/TOP criteria (GE fuel)

> The most limiting slope of power increase versus core flow throughout the cycle is provided to BWR-SA

AREVA

New and Spent Fuel Storage
Criticality Safety Analyses

> Maximum k-eff is established for each array assuming worst case conditions and including applicable tolerances and uncertainties

> Acceptance criteria are as defined in plant technical specifications or in NUREG-0800 (SRP), Chapters 9.1.1 and 9.1.2 (typically array k-eff $\leq$ 0.95)

> Based on the analyses, criticality safety limits are defined for each storage array which must be met in order to store fuel

NONPROPRIETARY

Transition Core Approach
Steps in Transition

> Data collection (neutronic data includes coresident fuel geometry and neutronic design, core geometry, as-loaded loading patterns, results from measured TIP traces, startup conditions, cold criticals, and actual hot operating state points)

> Model setup - CASMO-4/MICROBURN-B2 (mixed core effects accounted for by explicit modeling of all fuel in the core)

> [ ]

> Define core hot and cold target k-effs

> Establish current licensing basis (TS, FSAR, other documents)

> Disposition of events

> Provide neutronic input to transient, LOCA models

> Prepare core monitoring system input for Cycle N-1 parallel operation

NONPROPRIETARY

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Transition Core Approach
Steps in Transition (continued)

> Perform criticality safety (new and spent fuel) and fuel handling accident analyses, per contract scope

> Develop neutronic fuel/core designs for Cycle N

> Obtain customer concurrence with fuel cycle design

> Perform Cycle N neutronic safety reload licensing analyses

> Issue fuel cycle design report

> Provide startup/core monitoring data for Cycle N

> Licensing support/training as needed

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AREVA

Stability Analysis Methodology

Dan Tinkler
Engineer, BWR Safety Analysis

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OPRM Setpoint Methodology
Major Computer Codes

Code	Purpose
STAIF	Stability analysis code used to characterize the core and channel stability at the highest rod line at natural circulation
RAMONA5-FA	System analysis code used to generate the Delta over Initial CPR Versus Oscillation Magnitude (DIVOM) relationship
MICROBURN-B2	Neutronics code that provides initial 3-D nodal cross-sections, core loading/geometry, and statepoint information to both STAIF and RAMONA5-FA. Used to calculate initial core MCPR used to determine the supportable OPRM setpoint

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OPRM Setpoint Methodology
Calculation

OPRM Setpoint Analysis Calculations

> Hot Channel Oscillation Magnitude (HCOM) for a given S_p

- Obtained from statistical analysis by OPRM vendor
- Plant specific / Cycle and fuel independent
- Statistical 95/95 value calculated for a range of OPRM setpoints

NONPROPRIETARY

OPRM Setpoint Methodology
Calculation

OPRM Setpoint Analysis Calculations

> DIVOM Relationship

- Simulate regional oscillations to determine the relationship between power oscillations and CPR response
- DIVOM slope is the relative CPR response divided by hot bundle oscillation magnitude
- Uses a piece-wise linear interpolation in between DIVOM points to determine the slope

OPRM Setpoint Methodology
Calculation

OPRM Setpoint Analysis Calculations

> DIVOM Relationship *(continued)*

- DIVOM Point Definition
  - [

]

OPRM Setpoint Methodology
Calculation

OPRM Setpoint Analysis Calculations

> DIVOM Relationship *(continued)*

- Cycle-specific DIVOM analysis is best estimate
  - Best estimate RAMONA5-FA runs at prescribed exposures at the highest rod line at natural circulation
  - Sensitivity cases are performed
    - Limiting DIVOM exposure performed at +5% flow
    - [ ]
  - Limiting DIVOM is selected
- HCOM multiplied by DIVOM slope gives the relative ΔCPR response for a given S_p

Example DIVOM Curve

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OPRM Setpoint Methodology
Calculation

OPRM Setpoint Analysis Calculations

> Initial MCPR

- Determine the MCPR margin that exists prior to the onset of oscillations
  - Two scenarios
    - Two-pump trip to natural circulation at the highest rod line
    - Steady-state OLMCPR at 45% flow
  - For the two-pump trip scenario, include the MCPR margin gained by moving down the rod line
  - Calculations are cycle-specific

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OPRM Setpoint Methodology
Calculation

OPRM Setpoint Analysis Calculations

> Two approaches for SLMCPR protection:

- Stability operating limit can then be calculated versus S_P
- Compare SLMCPR with final MCPR (initial MCPR - ΔCPR)
  - Iterate to get the maximum S_P that protects SLMCPR

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OPRM Setpoint Methodology
RAMONA5-FA Computer Code

Description	RAMONA5-FA is a BWR system transient analysis code with models representing the reactor core, reactor vessel, and recirculation loops
Use	Perform core-wide and regional instability calculations to determine the relationship between the core power oscillations and the core CPR response
Documentation	BAW-10255(P) Rev 2, *Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code,* January 2006
Acceptability	The NRC performed an audit of the RAMONA5-FA DIVOM methodology spring 2005. The methodology has been submitted to the NRC as part of the Enhanced Option III methodology.

NONPROPRIETARY

RAMONA5-FA Computer Code
Major Features

> [

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Reactor Core Stability

AREVA

Transient Analysis Methodology

Darrell Carr
Team Leader, BWR Safety Analysis

NONPROPRIETARY

AREVA

Transient Analysis Methodology

> Transient analysis method overview
  - COTRANSA2
  - XCOBRA-T
> Event descriptions
> Equipment out-of-service scenarios
> Thermal limits

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Transient Analysis Methodology

> Most transient events are classified as moderate frequency events (i.e. may occur during a calendar year to once in 20 years, AOO)

> Anticipated operational occurrence
- Fuel cladding integrity limits are not violated:
  - MCPR safety limit
  - LHGR transient limit for 1% strain
- Reactor vessel pressure limit is not violated

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Transient Analyses
Major Computer Codes

Code	Use
MICROBURN-B2	3D cross-sections at state point of interest
PRECOT2	1D cross-sections at state point of interest
RODEX2	Gap conductance for core and hot channel
XCOBRA	Hot channel active flow
COTRANSA2	System and core average transient response
XCOBRA-T	Hot channel response and ΔCPR calculation

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Transient Analysis Methodology

Supporting Analyses

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ΔCPR Calculation Flow Chart

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ΔCPR Calculation Flow Chart
(continued)

NONPROPRIETARY

ΔCPR Calculation Flow Chart
(continued)

NONPROPRIETARY

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COTRANSA2 Computer Code

Description	COTRANSA2 is a BWR system transient analysis code with models representing the reactor core, reactor vessel, steam lines, recirculation loops, and control systems
Use	Evaluate key reactor system parameters such as power, flow, pressure, and temperature during core-wide BWR transient events Provide boundary conditions for hot channel analyses performed to calculate ΔCPR
Documentation	ANF-913(P)(A) Volume 1, Revision 1 and Supplements, *COTRANSA2:A Computer Program for Boiling Water Reactor Transient Analysis,* August 1990
Acceptability	The safety evaluation by the USNRC for the topical report ANF-913 approves COTRANSA2 for licensing applications

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COTRANSA2 Computer Code
Major Features

> Nodal (volume-junction) code with 1-dimensional homogeneous flow for the reactor system

> 1D neutron kinetics model for the reactor core (neutronics data obtained from MICROBURN-B2 and PRECOT2) that captures the effects of axial power shifts during the transient

> Core thermal-hydraulic model consistent with XCOBRA and XCOBRA-T

> Dynamic steam line model

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COTRANSA2 Computer Code
Additional Features

- Fuel rod transient heat conduction model
- Dual recirculation loops
- Dynamic pump model with homologous curves
- Steam separator model with level tracking
- Jet pump model
- Safety and relief valve models
- Turbine control, stop, and bypass valve models
- Control system models
- Reactor protection system trip logic

NONPROPRIETARY

COTRANSA2 Computer Code

Typical Nodalization

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COTRANSA2
Input Data

> System geometry
(volumes, areas, lengths, elevations)

> Component characteristics
(pumps, jet pumps, SRVs, TCVs, TSVs, TBVs)

> Reactor protection system characteristics
(setpoints, delays, scram performance, RPT performance)

> Control system characteristics
(feedwater controller, pressure regulator)

> Fuel characteristics
(gap conductance, neutronics data)

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COTRANSA2
Code Modules

- Core (COTRAN)
- Vessel and recirculation loops
- Steam line
- Control system
- Trip system

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Core Module - COTRAN

> Neutronics model characteristics

- [

]

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Core Module - COTRAN
(continued)

> Thermal-hydraulic model
- [

]

> Fuel rod model
- [

]

COTRANSA2

System and Recirculation Loop Module

> General hydraulic models

- 1D flow equations
- Homogeneous flow
- Conservation of mass and energy applied to control volumes; volume average density, pressure, and enthalpy defined at volume center
- Conservation of momentum applied at junctions; mass flow rate defined at junctions

COTRANSA2

System and Recirculation Loop Module

- Phase separated volume
  - [ ]
- Centrifugal pump model
  - Homologous pump curves used in dynamic model
- Jet pump model
  - Applicable to 1 and 5 jet drive nozzle designs
- Implicit iteration scheme with core and steam line modules

NONPROPRIETARY

COTRANSA2
Steam Line Module

> Hydraulic models

- 1-dimensional, 3-equation (mass, energy, momentum) gas dynamics formulation
- [

  ]
- Provisions are made to model turbine control, stop, and bypass valves as well as main steam isolation and safety/relief valves

COTRANSA2
Steam Line Module *(continued)*

COTRANSA2

Steam Line Module *(continued)*

> Turbine flow (control and stop valve)

- Valve positions can be changed by control system or trip system actions
- If valve has not been tripped, control valve position is based on control system demand (limited by maximum valve ramp rate)
- Turbine flow is based on control valve flow versus position characteristics

> Turbine bypass valve flow

- Valve position can be changed by control system or trip system actions
- Bypass valve position based on control system demand.
- Bypass flow is calculated based on position and valve flow versus position curve

COTRANSA2

Steam Line Module (continued)

> Safety/relief valve flow

- For relief mode, valve position is calculated based on delay time and valve ramp rate
- For safety mode, valve position is pressure dependent

> MSIV junction flow

- MSIV junction flow is calculated in the same way as other steam line junction flows
- Junction flow area is based valve ramp rate
- Once MSIV is fully closed, downstream nodes are dropped from the calculation

> Implicit iteration scheme with core and steam line modules

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COTRANSA2
Control System Module

> Provides flexibility to model most plant control systems – typically feedwater and pressure control systems are modeled

> Typically used to model feedwater flow, TCV position and bypass valve position

> Multiple control functions such as three-element, single-element and failed feedwater controllers can be entered

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COTRANSA2

Trip System Module

- Allows modeling of plant protection functions (RPS, ECCS, turbine, pumps, etc.)
- System valves, pumps, and control rods can be connected to trip system
- Trip system can be used for event marking and time step control
- Comparative trips (GT, LT, EQ, GE, LE)
- Logical trips (AND, OR)

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COTRANSA2
Benchmark Analyses

> Startup tests
  - Level setpoint change
  - Pressure regulator setpoint change
  - Load rejection
  - Recirc pump trip

> Peach Bottom turbine trip tests
  - TT2
  - TT2
  - TT3

> NRC Licensing Basis Transient (LBT)
  - BNL results
  - GE results

NONPROPRIETARY

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Transient Analysis Methodology
XCOBRA-T Computer Code

Description	XCOBRA-T predicts the transient thermal-hydraulic performance of BWR cores during postulated system transients
Use	Evaluate the transient thermal-hydraulic response of individual fuel assemblies in the core during transient events Evaluate the ΔCPR for the limiting fuel assemblies in the core during system transients
Documentation	XN-NF-84-105(P)(A) Volume 1 and Supplements, *XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis,* February 1987
Acceptability	The safety evaluation by the USNRC for the topical report XN-NF-84-105(P)(A) approves XCOBRA-T for licensing applications

NONPROPRIETARY

XCOBRA-T Computer Code
Major Features

- Hydraulic models are consistent with XCOBRA and COTRANSA2
- Transient fuel rod model with CHF prediction capability. The code iterates on hot channel RPF until CHF occurs at the limiting node at the limiting time during the transient. ΔCPR is equal to the initial CPR minus 1.0 (CPR when CHF occurs)
- A flow channel is used to represent the limiting assembly for each fuel type
- Nonlimiting fuel assemblies are grouped into average flow channels
- Boundary conditions (core, power, axial power shape, inlet enthalpy, upper- and lower-plenum pressure) are applied to the core

XCOBRA-T
Core Representation

> Parallel channels representing one or more fuel assemblies

> [

]

> Time-dependent total core power

> Time-dependent axial power shape

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XCOBRA-T
Thermal-Hydraulic Models

> 1-dimensional, 2-phase flow

> Unequal phase velocity (void-quality relationship)

> Thermal nonequilibrium (subcooled boiling)

> Fully implicit hydraulic solution includes effects of friction, local losses, elevation, area changes and density changes due to heating

> Mass and energy equations solved together using Newton-Raphson solution technique

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XCOBRA-T
Fuel Rod Models

> Steady-state and transient 1-dimensional (radial) heat conduction equation

> Fuel rod model is consistent with the RODEX2 fuel rod model

> Fuel-clad gap conductance is an input parameter (obtained from RODEX2)

> Internal heat generation based on:
- Core average power (time-dependent)
- Axial peaking factor (time-dependent)
- Radial peaking factor (constant with time)

System Transient Analyses

> Overpressurization Analysis (COTRANSA2 only)
  - ASME
  - ATWS
> Load rejection no bypass (LRNB)
> Turbine trip no bypass (TTNB)
> Feedwater controller failure (FWCF)
> Pressure regulator failure - downscale (PRFDS)

NONPROPRIETARY

ASME Overpressurization Analyses

> Description

- The ASME overpressurization events analyzed include the closure of all main steam isolation valves (MSIVs), closure of the TSVs, and closure of the TCVs with failure of direct scram (on valve position)

> Event classification

- The ASME event is a special analysis performed to demonstrate compliance with ASME pressure vessel design criteria

> Purpose of analysis

- Demonstrate that peak vessel pressure does not exceed 110% of the design vessel pressure
- Criteria: maximum vessel pressure (at bottom of RPV) ≤ 1375 psig and maximum steam dome pressure ≤ 1325 psig

NONPROPRIETARY

ATWS Event

> [

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NONPROPRIETARY

Transient Analysis
LRNB

Description	A significant grid disturbance occurs resulting in a loss of load on the generator and fast closure of the turbine control valve (TCV) The turbine valves are required to close rapidly to prevent overspeed of the turbine generator rotor
Event classification	Anticipated Operation Occurrence
Purpose of analysis	Calculate the ΔCPR for use in confirming the current OLMCPR or in establishing a new OLMCPR

NONPROPRIETARY

AREVA

Transient Analysis
LRNB

- Analytical methods
  - COTRANSA2 and XCOBRA-T are the primary analysis codes
- Event is mitigated by control rod insertion, revoiding due to increased heat flux and SRV actuation
- Special considerations
  - Below ~50% power, the power load unbalance will not cause a fast TCV closure and the valves close in servo mode. A generator protection system produced turbine trip occurs 1 second after event initiation
  - No direct scram on TCV motion or TSV position occurs below Pbypass (26% power)
  - Partial arc operation for TCVs at high power

Transient Analysis
TTNB

Description	A variety of turbine or nuclear system malfunctions will initiate the closure of the turbine stop valves (TSVs) and turbine control valves (TCVs) The turbine valves are required to close in order to protect the turbine
Event classification	Anticipated Operation Occurrence
Purpose of analysis	Calculate the ΔCPR for use in confirming the current OLMCPR or in establishing a new OLMCPR

Transient Analysis
TTNB

> Analytical methods
  - COTRANSA2 and XCOBRA-T are the primary analysis codes

> Event is mitigated by control rod insertion, revoiding due to increased heat flux and SRV actuation

> Special considerations
  - No direct scram on TCV motion or TSV position occurs below Pbypass (26% power)

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Transient Analysis
FWCF

Description	A failure in the feedwater control system results in a maximum demand signal to the feedwater pumps. The feedwater flow increases to the maximum capability of the feedwater pumps and results in an increase in reactor water level. A turbine trip occurs on high water level. The high water level trip is initiated to protect the turbine from damage due to liquid water entering the steam lines and turbine
Event classification	Anticipated Operation Occurrence
Purpose of analysis	Calculate the ΔCPR for use in confirming the current OLMCPR or in establishing a new OLMCPR

AREVA

Transient Analysis
FWCF

> Analytical methods
  - COTRANSA2 and XCOBRA-T are the primary analysis codes

> Event is mitigated by control rod insertion, revoiding due to increased heat flux and SRV actuation

> Special considerations
  - No direct scram on TCV motion or TSV position occurs below Pbypass (26% power)
  - Mismatch in feedwater flow and steam flow affects the overcooling portion of the event prior to the turbine trip
  - Lower feedwater temperature can make the event more severe
  - Lower initial water level results in longer overcooling phase

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Transient Analysis
PRFDS

Description	A pressure regulator failure downscale event results in the closure of all turbine control valves in servo mode. There is no scram on valve position or valve motion. The event is terminated by a scram on either high flux or high pressure. When the backup pressure regulator is operational, the event is benign and bound by the consequences of another event
Event classification	Anticipated Operation Occurrence
Purpose of analysis	Calculate the ΔCPR for use in confirming the current OLMCPR or in establishing a new OLMCPR

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Transient Analysis
PRFDS

> Analytical methods
  - COTRANSA2 and XCOBRA-T are the primary analysis codes

> Event is mitigated by control rod insertion, revoiding due to increased heat flux and SRV actuation

> Special considerations
  - High flux scram is attained during the event at high power
  - If high flux scram does not occur, the event continues until the high pressure scram set point is reached
  - Calculations at Brunswick are performed at 90-100% power to show that operation with a pressure regulator out-of-service in that power range is supported by the base case operating limits

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Brunswick Pressurization Transient Analyses

> Analyses performed to support nominal scram speed (NSS) insertion times and technical specification scram speed (TSSS) insertion times

> Analyses support two individual EOOS operating limit sets and one combined EOOS limit set

> Analyses support exposure-dependent operating limits

> [

]

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Brunswick Pressurization Transient Analyses

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Equipment Out-of-Service

- Operation with equipment out-of-service (EOOS) can impact the consequences of an event
  - Changes the system response
  - Can impact events differently
  - EOOS scenarios are plant dependent
- Brunswick EOOS scenarios that may impact pressurization transient analysis results
  - Feedwater heaters out-of-service (FHOOS)
  - Turbine bypass valves out-of-service (TBVOOS)
  - Safety/relief valves out-of-service (SRVOOS)

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Equipment Out-of-Service
FHOOS

> FWCF event become more severe
- Overcooling phase is worse due to larger decrease in inlet enthalpy
- The steam flow may increase slightly making the pressurization phase more severe

> LRNB and TTNB events are typically less severe
- Lower steam flow
- Initial TCV position is further closed which can make the event more severe

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Equipment Out-of-Service
TBVOOS

> Fast opening capability of the turbine bypass valves is assumed inoperable

> For LRNB and TTNB events, the base case situation already assume TBVOOS

> For FWCF events
  - Pressurization portion of the event becomes more severe
  - Below Pbypass, a high-pressure scram will occur

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Equipment Out-of-Service
SRVOOS

> SRVOOS

- The lowest setpoint safety/relief valve is assumed inoperable
- All base case pressurization events (overpressurization and transient) are performed with at least 1 SRVOOS

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Thermal Limits

> MCPR limits
> LHGR limits
> Coresident fuel considerations

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MCPR Limits

> Limits assembly power such that if an AOO were to occur, less than 0.1% of the fuel rods would experience boiling transition (no violation of the SLMCPR)
> Power-dependent MCPR (MCPRp) limits are established to protect the sum of the limiting ΔCPR and the SLMCPR
> A step change in the MCPRp limit is applied at Pbypass because of the loss of direct scram
> Exposure-dependent MCPRp limits are applied to provide MCPR margin early in the cycle
> Limits that support EOOS conditions are established
> Limiting MCPR limit from either the power-dependent or flow-dependent MCPR limit is applied in monitoring
> Limits are established for AREVA and coresident fuel

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LHGR Limits

- LHGR limits are applied to ensure that the mechanical design criteria are satisfied. This includes the 1% plastic strain and fuel centerline temperature criteria
- To ensure that the 1% strain and centerline melt criteria are met during an AOO, a power- or flow- dependent set-down or multiplier (LHGRFACp or LHGRFACf) may be applied to the steady-state LHGR limit
- LHGR multipliers may be established on an exposure and/or EOOS-dependent basis
- The limiting power-dependent or flow-dependent LHGR multiplier is applied in monitoring

Thermal-Mechanical Limits
Coresident Fuel

> For GE14 fuel, power- and flow-dependent multipliers are applied to the MAPLHGR limits to ensure that the thermal-mechanical design criteria continue to be met consistent with fuel vendor's methodology

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LOCA Analysis Methodology

Mike Garrett
Manager, BWR Safety Analysis

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Loss-of-Coolant Accident (LOCA) Analyses

> Two types of LOCA analyses are performed
  - LOCA break spectrum analysis
  - LOCA MAPLHGR limit analysis (or heatup analysis)

LOCA Break Spectrum Analysis

> A LOCA may occur over a wide range of break locations and sizes

> Largest possible break is double-ended rupture of a recirculation pipe; however, largest break may not be the most severe challenge to the core

> Breaks in ECCS piping are a special concern because the break results in a loss of an ECCS system as well as the LOCA

> Due to these complexities, a LOCA analysis must address a full range of break sizes and locations

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LOCA Break Spectrum Analysis

(continued)

> The purpose of the break spectrum is to identify the characteristics of the pipe break that result in the highest calculated PCT

> Characteristics considered

- Break location
- Break type
- Break size
- Limiting ECCS single failure

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LOCA Break Spectrum Analysis
(continued)

> Break locations
- Recirculation pump suction piping
- Recirculation pump discharge piping
- ECCS piping
- Other non-recirculation piping — generally dispositioned as being nonlimiting relative to fuel design requirements

LOCA Break Spectrum Analysis
(continued)

> Break type (recirculation line breaks)
- Double-ended guillotine (DEG)
  - Piping is assumed to be completely severed resulting in two independent flow paths to the containment
  - Total break area is equal to twice full pipe cross-section area
- Split break
  - Longitudinal opening or hole in the pipe that results in a single flow path to containment
  - Maximum flow area considered is equal to the cross-section area of pipe

LOCA Break Spectrum Analysis
(continued)

> Break sizes

- For recirculation DEG breaks, a range of discharge coefficients (1.0–0.4) is analyzed to cover uncertainty in break geometry
- For recirculation split breaks, a range of break sizes is analyzed (from 0.05 ft^2 to pipe cross-section area)
- For non-recirculation lines, a break size equal to the full cross-section area is assumed unless the event is potentially limiting and requires further assessments

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LOCA Break Spectrum Analysis
(continued)

> Limiting ECCS single failure

- Regulatory requirements specify that the most limiting single failure of ECCS equipment be assumed in LOCA analysis
- "Most limiting" refers to the ECCS equipment failure which produces the greatest challenge to acceptance criteria (generally PCT)
- Potentially limiting single failures are identified by the NSSS vendor and/or utility

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LOCA MAPLHGR Limit Analysis

> Purpose of analysis is to demonstrate that the desired MAPLHGR limit versus exposure ensures that the LOCA-ECCS acceptance criteria are met for the limiting break identified in the break spectrum analysis

> The break spectrum analysis is performed using BOL fuel parameters (e.g. stored energy, local peaking)

> Fuel parameters dependent on exposure have an insignificant effect on reactor system response during a LOCA; therefore, the limiting break characteristics from the break spectrum analysis are not exposure-dependent

> The thermal response of the fuel rods in the limiting plane of the hot assembly during a LOCA is dependent on parameters that vary with exposure

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LOCA MAPLHGR Limit Analysis

(continued)

> The reactor system response during a LOCA for the limiting break size and location, axial power shape, single failure, etc. is obtained from the break spectrum analysis

> Boundary conditions from the limiting LOCA are applied to each reload fuel and lattice designs in the MAPLHGR limit analysis

> Exposure-dependent PCT and MWR are calculated to confirm that the MAPLHGR limit protects the LOCA acceptance criteria

> AREVA generally establishes a MAPLHGR limit that is less restrictive than the fuel design LHGR limit

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Loss-of-Coolant Accident

> Characteristic phases
  - Blowdown phase
  - Refill phase
  - Reflood phase

Loss-of-Coolant Accident
Blowdown Phase

- Initial phase following pipe break
- Net loss-of-coolant inventory in the reactor vessel
- Rapid decrease in system pressure for large breaks
- Clad temperature increase due to degraded core flow
- Depending on break size, core becomes fully or partially uncovered
- Core cooling is provided by exiting coolant and by core spray late in the blowdown phase
- Blowdown defined to end when LPCS reaches rated flow (or time when LPCS rated flow would have occurred in analyses with degraded LPCS

Loss-of-Coolant Accident

Refill / Reflood Phase

> Net increase in coolant inventory due to ECCS operation

> Core spray provides some cooling

> ECCS systems (LPCS, LPCI, and HPCS) supplies liquid to refill lower portions of the reactor vessel

> Fuel heat transfer to coolant is less than decay heat and fuel cladding temperature increases

> At the start of reflood, lower plenum mixture level reaches bottom of core

> During reflood, cooling is provided below the mixture level by pool cooling and above the mixture level by entrained liquid flow

> Eventually the entrained liquid reaching the hot node provides sufficient cooling and the cladding temperature begins to decrease (time of hot node reflood)

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LOCA Analyses
Acceptance Criteria

> The calculated maximum fuel element cladding temperature shall not exceed 2200°F

> The calculated local oxidation of the cladding shall not exceed 0.17 times the cladding thickness

> The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water shall not exceed 0.01 times the amount that would be generated if all of the cladding surrounding the fuel were to react

> Calculated changes in core geometry shall be such that the core remains coolable

> The calculated core temperature shall be maintained at an acceptable value for the extended period of time required by the long-lived radioactivity remaining in the core

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LOCA Analyses
Acceptance Criteria

> LOCA-ECCS criteria are commonly referred to as:
  - PCT criterion (<2200°F)
  - Local oxidation criterion (<17%)
  - Hydrogen generation criterion (<1%)
  - Coolable geometry criterion
  - Long-term coolability criterion

LOCA MAPLHGR Limit

> A limit on the maximum average planar LHGR (MAPLHGR) is established to ensure the LOCA-ECCS criteria are met

> The EXEM BWR-2000 LOCA analysis results demonstrate that the PCT, local oxidation, and hydrogen generation criteria are met

> Compliance with these three criteria ensures that a coolable geometry is maintained

> Long-term coolability is confirmed for initial reloads – not fuel design dependent

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LOCA MAPLHGR Limit
Basis for Selected Limit

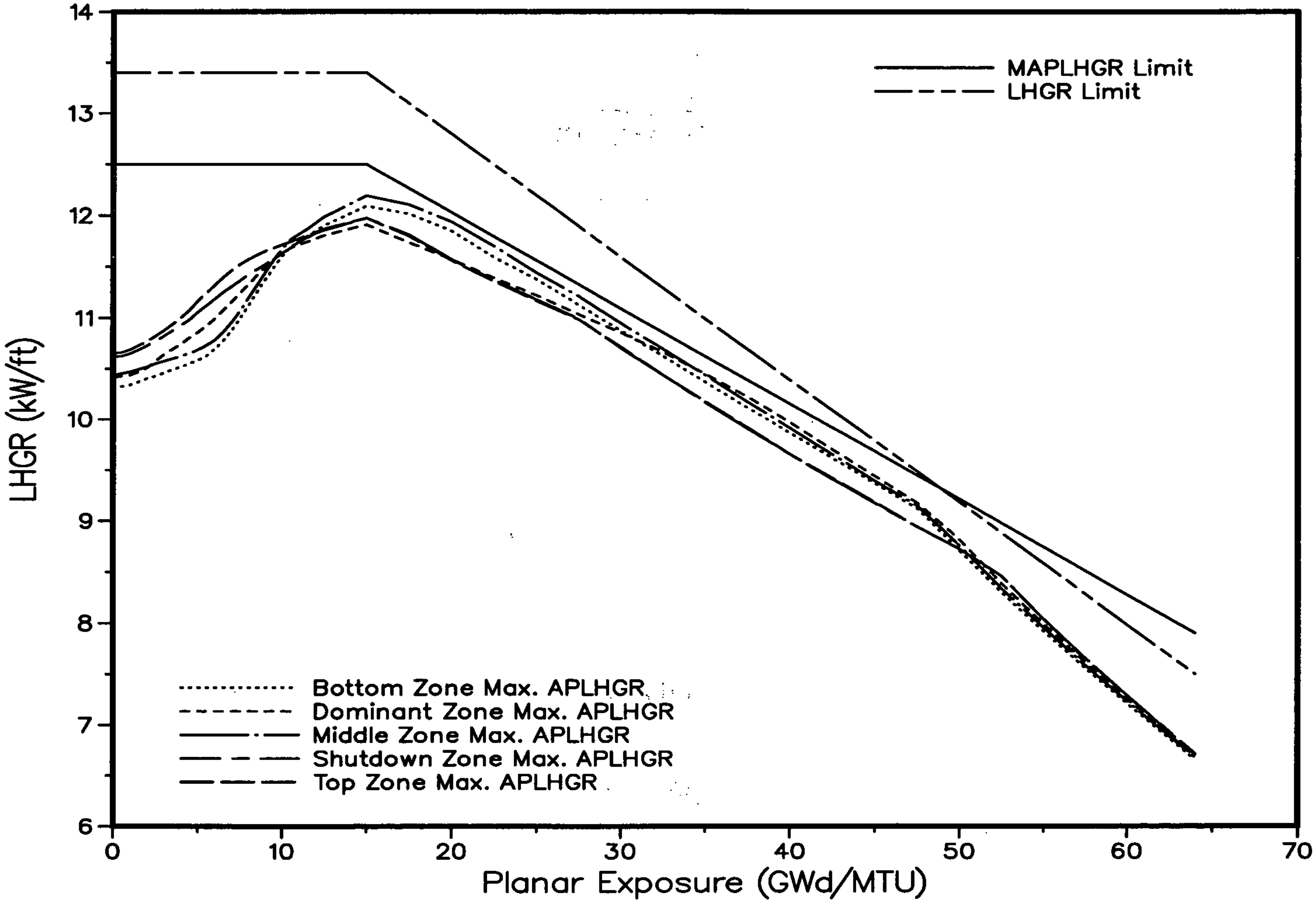
> Goal is to establish MAPLHGR limit that is less restrictive than the fuel design LHGR limit

> The maximum local peaking factor as a function of exposure for each void fraction is obtained from the neutronics cross-section generation calculation

> Divide the fuel design LHGR limit by the lowest maximum local peaking factor at each exposure

> Repeat for each lattice design in the reload (ignoring blankets)

Basis for MAPLHGR Limit
(continued)



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LOCA Analysis Methodology

- The NRC-approved LOCA-ECCS Evaluation Model used by AREVA is referred to as EXEM BWR-2000
- EXEM BWR-2000 is an Evaluation Model that meets the requirements of 10 CFR 50 Appendix K
- The EXEM BWR-2000 methodology consists of three major computer codes and several auxiliary computer codes which are used to evaluate the reactor system and fuel response during a LOCA

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LOCA Analysis Methodology
Major Computer Codes

Code	Purpose
RODEX2	Fuel rod performance code used to predict the thermal-mechanical behavior of BWR fuel rods as a function of exposure
RELAX	BWR system analysis code used to calculate the reactor system and hot channel response during the blowdown, refill, and reflood phases of a LOCA
HUXY	Heat transfer code used to calculate the heatup of a BWR fuel assembly during all phases of a LOCA

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LOCA Analysis Methodology
Auxiliary Computer Codes

Code	Purpose
LPF	Provide local peaking factors for rod groups from CASMO tape71 file
READW3	Incorporates power spikes to the MAPLHGR limit at defined exposures (typically every 5000 MWd/MTU) into the LPH
RDXHXY	Generates HUXY inputs from RODEX2 results
PREHUXY	Reads restart file from RELAX hot channel calculation and writes HUXY inputs for normalized power, fluid temperature, fluid quality and blowdown HTC
SECHECK	Compares HUXY initial stored energy against RODEX2 calculated stored energy

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LOCA Analysis Methodology

RODEX2 Computer Code

Description	Fuel rod performance code used to predict the thermal-mechanical behavior of BWR fuel rods as a function of exposure and power history
Use	Fuel rod stored energy Initial fuel rod thermal and mechanical properties
Documentation	XN-NF-81-58(P)(A) Rev 2 and Supplements, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model,* March 1984
Acceptability	The safety evaluation by the NRC for XN-NF-81-58(P)(A) Rev 2 and Supplements approves RODEX2 for licensing applications

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Major RODEX2 Models

- Fission gas release
- Fuel swelling, densification and cracking
- Fuel to clad gap conductance
- Radial thermal conduction
- Free volume and internal gas pressure
- Fuel and cladding deformation
- Cladding corrosion

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LOCA Analysis Methodology
RELAX Computer Code

Description	RELAX is a BWR systems analysis code used to calculate the reactor system and core hot channel response during a LOCA
Use	Evaluate the time required to reach the end of the blowdown phase and to reach core reflood during the refill/reflood phase of the LOCA analysis Evaluate hot channel fluid conditions during the blowdown phase of LOCA and time to reach hot channel reflood during the refill/reflood phase of the LOCA analysis
Documentation	EMF-2361(P)(A), *EXEM BWR-2000 ECCS Evaluation Model*, May 2001
Acceptability	The safety evaluation by the NRC for the topical report EMF-2361(P)(A) approves RELAX for licensing applications

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RELAX Computer Code
Major Models

> Reactor system is nodalized into control volumes and junctions

> Mass and energy conservation equations are solved for control volumes

> Fluid momentum equation is solved at junctions to determine flow rates

> 1-dimensional, homogeneous equilibrium

> Three-equation model with drift flux model

> Complies with Appendix K requirements for ECCS analysis

> Separate models for average core and hot assembly

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RELAX System Model

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RELAX Hot Channel Model

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LOCA Analysis Methodology
HUXY Computer Code

Description	Heat transfer code used to calculate the heatup of the peak power plane in a BWR fuel assembly during the blowdown, refill, and reflood phases of a LOCA
Use	Evaluate the peak clad temperature and metal-water reaction in the fuel assembly resulting from a LOCA
Documentation	XN-CC-33(A) Rev 1, *HUXY: A Generalized Multirod Heatup Code With 10CFR50 Appendix K Heatup Option – User's Manual,* December 1975
Acceptability	The safety evaluation by the NRC for the topical report XN-CC-33(A) Rev 1 approves HUXY for licensing applications

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HUXY Computer Code
Major Features

> Models an axial plane in a fuel assembly

> Models individual rods in plane of interest

> Models assembly local power distribution and rod-to-rod radiant heat transfer

> Uses RELAX hot channel boundary conditions during blowdown

> Uses Appendix K spray heat transfer coefficients during refill

> Uses Appendix K reflood heat transfer coefficient after hot node reflood

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HUXY Computer Code
Major Features

- > Fuel rod conduction heat transfer model
- > Convection heat transfer
- > Radiation heat transfer (rod-to-rod, rod-to-channel)
- > Fuel channel and rod quenching
- > Metal-water reaction (additional heat source and oxide buildup)
- > Clad swelling and rupture

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EXEM BWR-2000 LOCA Analysis Methodology

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EXEM BWR-2000 Methodology
Calculation Methodology

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LOCA Analysis Methodology
Cycle-Specific Analyses

> For each transition cycle, a complete plant-specific LOCA break spectrum analysis is performed
- Break location
- Break geometry (split, guillotine)
- Break size
- ECCS failure
- Axial power shape
- Initial core flow

> For each cycle, MAPLHGR limit analysis is performed
- Limiting break characteristics from break spectrum analysis
- Each lattice design in core
- Full exposure range

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LOCA Analysis Methodology
Break Spectrum Analyses

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LOCA Analysis Methodology
MAPLHGR Analyses

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EPU and Non-EPU Analysis Conditions

Doug Pruitt

Manager, Codes and Methods

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Reload Licensing Methodology

> Reload licensing analysis are performed to ensure that all fuel design and operating limits are satisfied for the limiting assembly in the core

> Applicability of design methodology was determined by reviewing the explicit SER restrictions on the BWR methodology

- No SER restrictions on power level for the AREVA topical reports
- No SER restrictions on the parameters most impacted by the increased power level
  - Core average void fraction
  - Steam/Feedwater flow
  - Jet Pump M-ratio

> The impact of EPU on core and reactor conditions was evaluated to determine any challenges to the theoretical validity of the models

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Power Uprate Considerations

- > Thermal operating limits (MCPR, MAPLHGR, LHGR) are fairly insensitive to power uprate
- > The ranges of key physical phenomena (e.g., heat flux, fluid quality, assembly flow) in limiting assemblies during normal operation or transient events are not significantly different for uprated and non-uprated conditions
- > Fuel specific determination of critical power is the most limiting methodology for non-uprated and uprated BWR operation
- > AREVA analysis methodologies impose critical power correlation limits so the fundamental range of assembly conditions must remain within the same parameter space under uprate conditions

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Power Uprate Observations

> Maintaining the same critical power limits with increased core power requires flattening of the normalized radial power distributions
  - Leads to a more uniform core flow distribution and slightly higher flow rates in the hottest assemblies
> More assemblies and fuel rods are near thermal limits and may result in a higher safety limit MCPR
> Higher steam flow rate and associated feedwater flow rate
> Core average void fraction will increase
> Higher core average power will lead to an increased core pressure drop and a slight decrease in jet pump performance

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Representative Assembly Power Distribution

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Power Uprate Considerations

> Changes to the hot assemblies
  - Power will be approximately the same
  - Flow will slightly increase

> Changes to the average assemblies
  - Power will increase
  - Flow will slightly decrease

Conclusion:

> The current parametric envelope will continue to encompass the conditions for all assemblies in an uprated reactor

> Therefore, the methods used to assess assembly thermal-hydraulics are applicable to power uprate

Thermal-Hydraulic Core Analyses Testing Based

- AREVA tests to confirm or establish the applicability of methods
  - PHTF test measurements provide assembly flow and pressure drop characteristics (e.g., pressure loss coefficients)
  - Karlstein test facility provides both the assembly two-phase pressure drop and CHF performance characteristics
  - FCTF tests confirm the conservatism of the Appendix K spray heat transfer coefficients
- Supplemental testing at Karlstein extends the validation and applicability of our methods
  - Hydraulic stability
  - Oscillatory dryout and rewet
  - Void fractions

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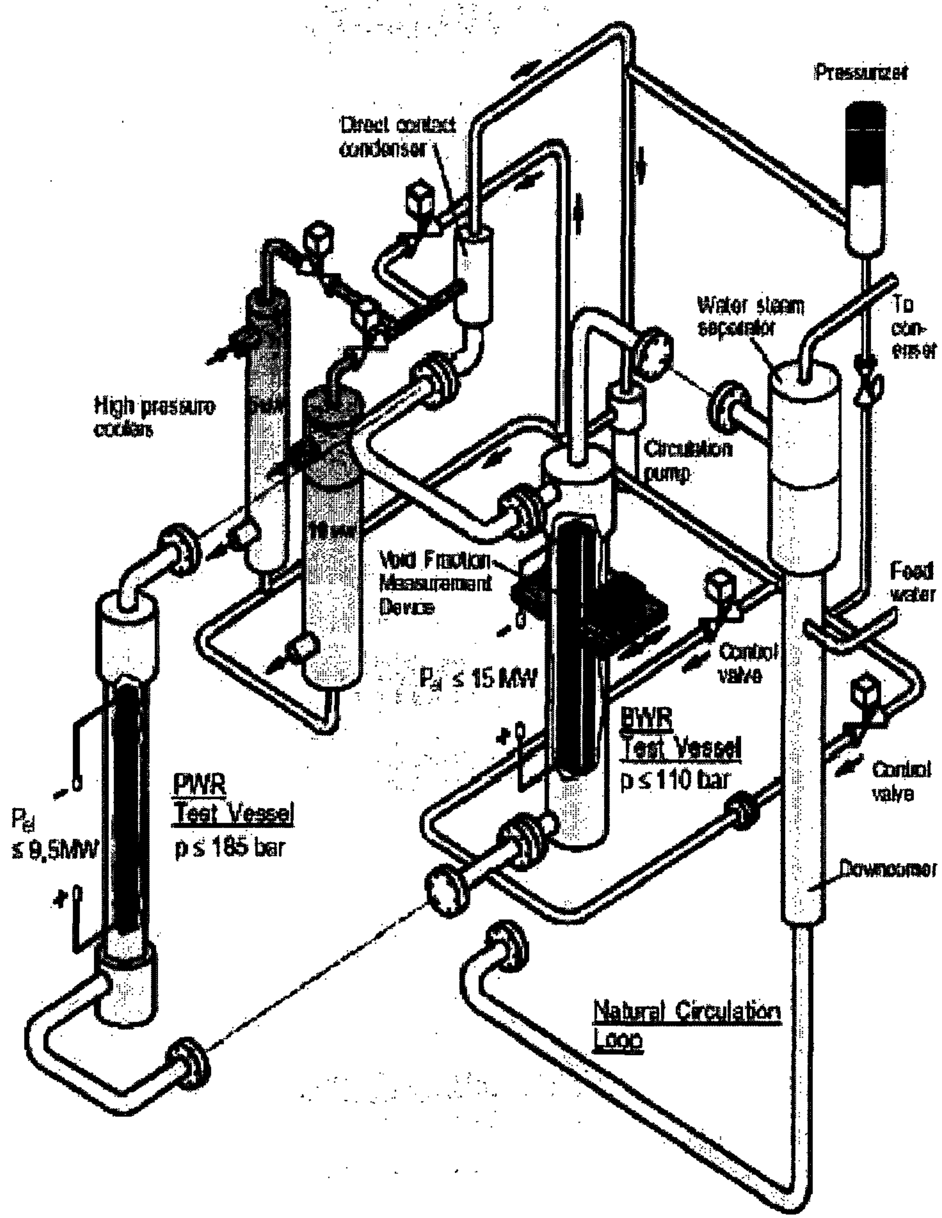
Karlstein Thermal Hydraulic (KATHY) Test Loop



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Critical Power Constraints

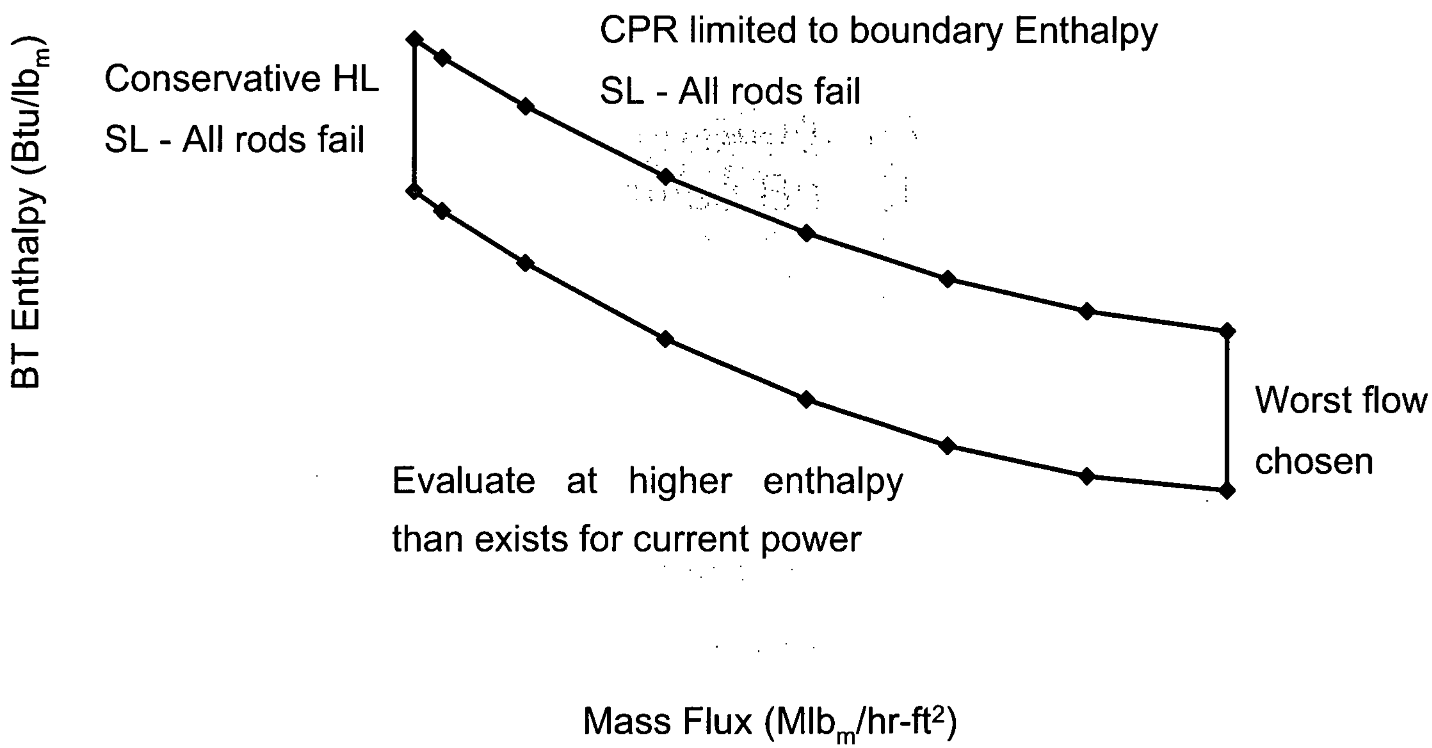
> SPCB fuel-specific CHF correlation based on KATHY test data

> Approved range of applicability for the SPCB correlation is enforced in codes (inlet subcooling, flow, pressure, boiling transition enthalpy) - uprate does not change this

- In some calculations, state conditions outside the limits are handled by NRC-approved conservative assumptions

> LOCA calculations fall outside the SPCB parametric envelope during the accident simulation. In this case, the local conditions formulation of the modified Barnett correlation is used.

SPCB Out-of-Bounds Conditions



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Critical Power Constraints

- > Since the CHF performance is characterized and imposed on a fuel design specific basis the assembly operating conditions must remain within the approved application range
- > This fundamental restriction results in minimal differences between the benchmarked core conditions and those calculated for power uprate conditions
- > This similarity is confirmed by comparing the assembly exit conditions
  - KATHY pressure drop measurements
  - CASMO4/MICROBURN-B2 approved benchmark conditions (EMF-2158(P)(A))
  - Cycle depletion conditions for a Brunswick 120% power uprate / MELLLA+ core design

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Pressure Drop Tests Vs. Reactor Benchmark and Design Conditions

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CASMO-4/MICROBURN-B2 Operating Experience

Reactor	Reactor Size, #FA	Power, MWt (% Uprated)*	Ave. Bundle Power, MWt/FA	Approximate Peak Bundle Power, MWt/FA	Fuel/Cycle Licensing**	Uprate Comments
A	592	2575 (0.0)	4.4	7.2	X	
B	592	2575 (0.0)	4.4	7.4	X	
C	532	2292 (0.0)	4.3	7.3	(X)	
D	840	3690 (0.0)	4.4	7.5	X	Licensing only through Cycle 20
E	500	2500 (15.7)	5.0	8.0	X	For 3 cycles oper.
F	444	1800 (5.9)	4.1	7.3	X	
G	676	2928 (8.0)	4.3	7.6	(X)	
H	700	3300 (9.3)	4.7	8.0	(X)	
I	784	3840 (0.0)	4.9	8.1	(X)	
J	624	3237 (11.9)	5.2	7.8	(X)	
K	648	3600 (14.7)	5.6	8.6	(X)	With ATRIUM-10XM
L	648	2500 (10.1)	3.9	6.9	(X)	
M	624	3091 (6.7)	5.0	7.7	X	
N	800	3898 (1.7)	4.9	7.7	X	
O	764	3489 (5.0)	4.6	7.2	X	
Brunswick†	560	2923 (20.0)	5.2	7.5	X	Previously licensed by GNF

* Latest power uprates.
** (x)=currently fuel licensing only (Europe).

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Conclusions
Thermal-Hydraulic Core Analysis

> Power uprate introduces changes in core design and steam flow rate

> Assemblies are subject to the same LHGR, MAPLHGR, MCPR, and cold shutdown margin limits

> These LCOs restrict the assembly powers, flows and void fractions typically within the ranges observed in current plant operation, the neutronics benchmarking database and the AREVA testing experience

> Therefore,

- Hydraulic models and constitutive relationships used to compute the core flow distribution and local void content remain applicable
- Neutronic methods used to compute the nodal reactivity and power distributions remain applicable

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Power Uprate Impact on Transient Analysis

> Phenomena of interest for BWR AOO transient analysis
  - Void fraction/quality relationships
  - Determination of CHF
  - Pressure drop
  - Reactivity feedbacks
  - Heat transfer characteristics

> The dominant phenomena of interest are related to the local assembly conditions, not the total core power

> AREVA transient CHF measurements in KATHY are used to qualify the transient hydraulic solution
  - Benchmarks capture the transient integration of the conservation equations and constitutive relations (including the void-quality closure relation) and determination of CHF with SPCB

> AREVA benchmarks illustrate conservative predictions of time of dryout

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Transient Qualification

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Power Uprate Impact on Transient Analysis

> Outside the core, the system simulation relies on solutions of the basic conservation equations and equations of state
  - Feedwater flow and Jet Pump M-ratio changes
  - Steam flow rate and steam line dynamics for pressurization events
    - Impact of steam-flow rate dependent on valve characteristics for pressurization events
  - Solution of conservation equations have no limitations within the range of variation associated with power uprate

> Reactivity feedbacks are validated in a variety of ways
  - Fuel lattice benchmarks to Monte Carlo results (SER restriction)
  - Steady-state monitoring of reactor operation (power distributions and eigenvalue)
  - Benchmark of coupled system to the Peach Bottom 2 turbine trip transients that exhibit a minimum of 5% conservatism

> Transient analysis remain valid for power uprate

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Power Uprate Impact on LOCA

> Local hot assembly parameters (PCT & % M/W reaction) are determined primarily from the hot assembly initial stored energy, hot assembly transient decay heating and primary system liquid inventories
  - Hot assembly initial stored energy, decay heating, and fluid inventory are not expected to change significantly (same LHGR and MCPR limits)
  - System inventory differences due to the increased core power have a transient feedback on the hot channel flow and fluid conditions
    - Transient inventory differences due to power uprate are encompassed by the variation required to assess the entire break spectrum
    - Code capabilities are not challenged by the differences

> Local hot assembly PCT and % M/W reaction exhibit only small changes due to power uprate

> Core-wide parameters (core-wide M/W reaction and demands on long-term cooling) increase due to power uprate

> Current LOCA methodology covers all phenomena for uprated conditions

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Power Uprate Impact on Stability

> The flatter radial power profile induced by the power uprate will have a small impact on stability for same operating state point
  - The flatter radial power profile may increase the core decay ratios
    - Potential reduction in the eigenvalue separation
    - More assemblies operating at higher P/F ratios

> The STAIF code computes the stability characteristics of the core
  - Frequency domain solution of the applicable conservation and closure relationships
  - Computes the regional mode directly using the actual state-point eigenvalue separation
  - Benchmarked against full assembly tests, as well as global and regional reactor data as late as 1998

> The impact of the "flatter" core design on stability limits will be directly computed based on the projected operating conditions

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Power Uprate Impact on Special Events

> AREVA performs ASME overpressurization analysis to demonstrate compliance with the peak pressure criteria
  - System response and sensitivities are essentially the same as AOO pressurization events

> AREVA performs ATWS analysis to demonstrate compliance with the peak pressurization criteria which occurs early in the event
  - Early system response and sensitivities are essentially the same as the transient simulations presented earlier

> Appendix R analysis is performed using the approved LOCA analysis codes
  - Like LOCA, the impact of power uprate is primarily through the increase in decay heat in the core
  - Decay heat is conservatively modeled using industry standards applied as specified by regulatory requirements
  - Use of Appendix K heat transfer correlations and logic is conservative for Appendix R calculations

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EPU Impact

> EPU operation does not challenge the applicability of the methods used to compute and monitor against licensing limits

> EPU operation is expected to impact the following areas:
- Safety Limit
- Transient response due to different balance between core voids, feedwater/steam flow rates and steamline valve characteristics
- LOCA core-wide metal-water reaction
- LOCA long term cooling
- Backup stability protection – exclusion regions

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Power Uprate Applicability Summary

> Maintaining margin to fuel design safety limits imposes restrictions on the range of operating conditions an assembly may experience during steady-state and transient conditions
> Increasing the core thermal power is accommodated by radial power flattening so that limiting assembly conditions deviate only slightly from current operating experience values
> The AREVA-approved licensing methods directly assess the impacts of power uprate on operating limits without modification
> The AREVA-approved licensing methods remain valid for power uprate conditions

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CASMO-4/MICROBURN-B2 Methodology

Ralph Grummer
Manager, Nuclear Technology

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BWR Methodology Applicability

> Objective

- Describe the cross section reconstruction process used by AREVA NP
- Demonstrate that the AREVA Methodology is accurate for high void conditions

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CASMO-4

> CASMO-4 performs a multi-group (70) spectrum calculation using a detailed heterogeneous description of the fuel lattice components

- Explicit modeling of fuel rods, absorber rods, water rods/channels and structural components
- The library has cross sections for 108 materials including 18 heavy metals
- Depletion performed with a predictor-corrector approach in each fuel or absorber rod
- 2-dimensional transport solution is based upon the Method of Characteristics

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CASMO-4 (continued)

- Provides pin-by-pin power and exposure distributions
- Produces homogeneous multi-group (2) microscopic cross sections as well as macroscopic cross sections
- Determines discontinuity factors
- Performs 18-group gamma transport calculation
- Ability to perform colorset (2X2) calculation with different mesh spacings
- Reflector calculations are easily performed

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MICROBURN-B2

> Microscopic fuel depletion
> Full two energy group neutron diffusion equation solution
> Modern nodal method solution is used
> Uses a higher order spatial method
> Water gap dependent flux discontinuity factors
> Multilevel iteration technique for efficiency
> MICROBURN-B2 treats a total of 11 heavy metal nuclides to account for the primary reactivity components

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MICROBURN-B2 (continued)

> A model for nodal burnup gradient
> A model for spectral history gradient
> Full 3-dimensional pin power reconstruction method
> TIP (neutron and gamma) and LPRM response models
> Steady-state thermal-hydraulics model
> Direct moderator heat deposition based upon CASMO-4 calculations
> Calculation of CPR, LHGR, and MAPLHGR

BWR Methodology

> Let us look at the cross section representation used in MICROBURN-B2

> MICROBURN-B2 determines the nodal macroscopic cross sections by summing the contribution of the various nuclides

MICROBURN-B2 Cross Section Representation

$$\Sigma_x(\rho,\Pi,E,R) = \sum_{i=1}^{I} N_i \sigma_x^i(\rho,\Pi,E,R) + \Delta\Sigma_x^b(\rho,\Pi,E,R)$$

where

Σ_x = nodal macroscopic cross section

$\Delta\Sigma_x^b$ = background nodal macroscopic cross section $(D, \Sigma_f, \Sigma_a, \Sigma_r)$

N_i = nodal number density of nuclide "i"

σ_x^i = microscopic cross section of nuclide "i"

I = total number of explicitly modeled nuclides

ρ = nodal instantaneous coolant density

Π = nodal spectral history

E = nodal exposure

R = control fraction

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MICROBURN-B2 Cross Section Representation

- Functional representation of σ_x^i and $\Delta\Sigma_x^b$ comes from 3 void depletion calculations with CASMO-4
- Instantaneous branch calculations at alternate conditions of void and control state are also performed
- The result is a multi-dimensional table of microscopic and macroscopic cross sections

MICROBURN-B2 Cross Section Representation

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MICROBURN-B2 Cross Section Representation

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MICROBURN-B2 Cross Section Representation

> At BOL the relationship is fairly simple
  - The cross section is only a function of void fraction (water density)
  - The reason for the variation is the change in the spectrum due to the water density variations

> At any exposure point, a quadratic fit of the three CASMO-4 data points is used to represent the continuous cross section over instantaneous variation of void or water density

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MICROBURN-B2 Cross Section Representation

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MICROBURN-B2 Cross Section Representation

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MICROBURN-B2 Cross Section Representation

> Detailed CASMO-4 calculations confirm that a quadratic fit accurately represents the cross sections

MICROBURN-B2 Cross Section Representation

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MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

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MICROBURN-B2 Cross Section Representation

> With depletion the isotopic changes cause other spectral changes

> Cross sections change due to the spectrum changes

> Cross sections also change due to self-shielding as the concentrations change

> These are accounted for by the void (spectral) history and exposure parameters

> Exposure variations utilize a piecewise linear interpolation over tabulated values at 100 exposure points

> The 4-dimensional representation can be reduced to 3 dimensions by looking at a single exposure

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MICROBURN-B2 Cross Section Representation

This is a smooth well behaved surface

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MICROBURN-B2 Cross Section Representation

> Quadratic interpolation is performed in each direction independently for the most accurate representation

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MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

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MICROBURN-B2 Cross Section Representation

> The results of this process for all isotopes and all cross sections in MICROBURN-B2 were compared for an independent CASMO-4 calculation with continuous operation at 40% void (40% void history) and branch calculations at 90% void for multiple exposure

> The results show very good agreement for the whole exposure range

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MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

> At the peak reactivity point multiple comparisons were made to show the results for various instantaneous void fractions

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MICROBURN-B2 Cross Section Representation

Quadratic fit using 0-40-80 provides excellent representation of data

MICROBURN-B2 Cross Section Representation

> MICROBURN-B2 uses water density rather than void fraction in order to account for pressure changes as well as sub-cooled density changes

> MICROBURN-B2 uses spectral history rather than void history in order to account for other spectral influences due to actual core conditions (fuel loading, control rod inventory, leakage, etc.)

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MICROBURN-B2 Cross Section Representation

> The Doppler feedback due to the fuel temperature is modeled by accumulating the Doppler broadening of microscopic cross sections of each nuclide

$$\Delta\Sigma_{\mathrm{x}} = (\sqrt{T_{eff}} - \sqrt{T_{ref}})\sum_{i}^{I} \frac{\partial \sigma_{\mathrm{x}}^{\mathrm{i}}}{\partial \sqrt{T_f}} N_i$$

where:

T_{eff} = Effective Doppler Fuel Temperature

T_{ref} = Reference Doppler Fuel Temperature

$\sigma_{\mathrm{x}}^{\mathrm{i}}$ = microscopic cross section (fast and thermal absorption) of nuclide i

N_i = density of nuclide i

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MICROBURN-B2 Cross Section Representation

> The partial derivatives are determined from branch calculations performed with CASMO-4 at various exposures and void fractions for each void history depletion

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MICROBURN-B2 Cross Section Representation

> The tables of cross sections include data for controlled and uncontrolled states.

> Otherwise the process is the same for controlled states

> Other important feedbacks to nodal cross sections are lattice burnup/spectral history gradient and instantaneous spectral interaction between lattices of different spectra

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Validation of MICROBURN-B2 for EPU Conditions

Ralph Grummer
Manager, Nuclear Technology

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BWR Methodology Applicability

> Objective

- Describe the validation process used by AREVA NP
- Demonstrate that the AREVA Methodology is applicable to EPU conditions at Brunswick
- Demonstrate that data provided in the Neutronic Methodology Topical report bounds the expected conditions of EPU operation at Brunswick

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BWR Methodology Applicability

> Validation of Steady-State Neutronic Methods for EPU conditions

- Tabulate the key parameters being validated (nodal power, pin power etc.), the type of benchmarking/validation that was performed and the bundle conditions corresponding to the validation
- AREVA's neutronic method was validated by gamma scan and core follow benchmarking based upon the current fuel designs operated under the current operating strategies and core conditions

EMF-2158(P)(A) Validation Basis

> EMF-2158(P)(A) defined a set of criteria to demonstrate the acceptability of the Neutronic design code system

> Code system results were compared against critical experiments, higher order methods and actual commercial operating experience

> The SER states that the code system shall be applied in a manner such that results are within the range of the validation criteria (Tables 2.1, 2.2, and 2.3)

Fuel Lattice Criteria Table 2.1

Fuel Lattice Criteria Table 2.1 *(continued)*

Fuel Lattice Criteria Table 2.1 *(continued)*

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Core Simulator Validation Table 2.2

Core Simulator Validation Table 2.2 *(continued)*

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Core Simulator Validation Table 2.2 *(continued)*

> TIP data taken from operating commercial power plants

> Gamma scan data taken from Quad Cities measurements on 8X8 assemblies

- [
-]

> Gamma scan data taken from KWU-S measurements on ATRIUM-10 assemblies

- [
-]

Includes current fuel designs and operating strategies

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KWU-S Gamma Scan Benchmark Results

EMF-2158(P)(A) pp. 8-8

Local power distribution uncertainty is not axial level dependent

Measured Power Distribution Uncertainty
Table 2.3

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Continuous Validation Process

> AREVA Work Practice P104,129 requires evaluation for a significant fuel design change

> CASMO-4 and MCNP calculations are performed

> Fission rate distribution statistics are compared to Table 2.1

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ATRIUM™-10 Lattice Validation

Fission Rate Criteria Met

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Continuous Validation Process

> For a new reactor, benchmark calculations are performed

> Hot operating eigenvalue statistics are compared to Table 2.2

> Cold startup eigenvalue statistics are compared to Table 2.2

> TIP statistics are compared to Table 2.2

> Local peaking comparisons are determined from the lattice calculations

Reactor Validation Results

Eigenvalue criteria are met

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Reactor Validation Results

TIP comparisons include calculation and measurement uncertainties

Reactor Validation Results

Measured power distribution uncertainties are a convolution of calculation and measurement uncertainties

δB is calculated power uncertainty

δD is synthesized TIP uncertainty

δT is calculated TIP uncertainty

NIJ is the number of TIPs

Reactor Validation Results

TIP measurements are taken with Gamma TIPs

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Reactor Validation Results

> Measured and calculated TIP comparisons meet the requirements
> Measured symmetric TIP comparisons meet the requirements
> Together these indicate that the measured power uncertainty requirements are met

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Reactivity Coefficients – Void Coefficient

> Reactivity Coefficients – Void Coefficient

- Evaluate the AREVA methods and establish if the uncertainties and biases used in reactivity coefficients (e.g. void coefficient) are applicable or remain valid for the neutronic and thermal-hydraulic conditions expected for EPU operation.

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Additional Validation

> In order to evaluate the accuracy of the void coefficient, MCNP runs have been made

> These results indicate that CASMO performs an accurate assessment of the void effect

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CASMO-4 Vs. MCNP Results

Casmo-4 void coefficient is nearly identical to MCNP

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Void Coefficient Verification

> A measure of the quality of the simulator calculation is the variation of the critical eigenvalue

> Observations of this behavior relative to core average void fraction indicate that there is no systematic bias

> Cycle exposure trends are accounted for by the use of target eigenvalue curves

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Void Coefficient Verification from Topical Report

[

]

There is no trend in core eigenvalue relative to void fraction

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Void Coefficient Verification

> The void coefficient is calculated accurately for a wide variety of core average void fractions

> The methodology retains the same accuracy for the conditions represented by EPU

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Additional Validation

> Validation of Steady-State Neutronic Methods for EPU conditions

- Demonstrate the current uncertainties and biases established in the benchmarkings and presented in Tables 9.8 and 9.9 of EMF-2158 (P)(A) remain valid for the neutronic and thermal-hydraulic conditions predicted for the EPU operation

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Additional Validation

> TIP measurements taken at reactors that have operated in extended power uprate conditions indicate that the calculation accuracy is not impacted

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Additional Validation

Power uprate experience shows that uncertainties are unchanged

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Conclusion

- The neutronic methodology utilizing CASMO-4 and MICROBURN-B2 accurately models reactor cores with a wide range of operating conditions including those anticipated for EPU at Brunswick
- The uncertainties presented in EMF-2158(P)(A) continue to be applicable for EPU operation at Brunswick

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CASMO-4/MICROBURN-B2 Methodology Experience Relative to Brunswick Application

Ralph Grummer
Manager, Nuclear Technology

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BWR Methodology Experience

> Objective

- Describe the experience base for AREVA NP methodologies
- Demonstrate that the AREVA Methodology is applicable to EPU conditions at Brunswick

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Topical Report Thermal-Hydraulic Conditions

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BWR Methodology Experience

[redacted]

Current Experience is consistent with the topical report

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Topical Report Thermal-Hydraulic Conditions

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BWR Methodology Experience

Current Experience is consistent with the topical report

BWR Methodology Experience

> At the point of the highest exit void fraction, additional detail was evaluated

- Core average void axial profile
- Axial profile of the peak assembly
- Histogram of the nodal void fractions in core

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BWR Methodology Experience

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BWR Methodology Experience

Current Experience has Similar Void Population as Expected for SQH Power Uprate

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Evaluation of Power Uprate for Brunswick

Max assembly powers are less than those presented in the topical report

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Evaluation of Power Uprate for Brunswick

Max exit voids are less than those presented in the topical report

Brunswick with Power Uprate

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Brunswick with Power Uprate

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Experience with High Void Fractions

> Conclusions

- Reactor conditions for Brunswick with power uprate are not significantly different from current experience
- The range of void fractions in the topical report data exceeds that expected for the power uprate conditions
- The distribution of voids is nearly the same as current experience
- Cross section representation is accurate for power uprate conditions

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Power Distribution Uncertainties

> Objective

- Describe the process used by AREVA NP to define the power distribution uncertainties
- Demonstrate that the AREVA Methodology is applicable to EPU conditions at Brunswick

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Power Distribution Uncertainties

- First we will look at how AREVA determined the measured power distribution uncertainties
- One of the major components is the comparison of measured and calculated TIPs
- This includes measurement uncertainty as well as calculation uncertainty

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Power Distribution Uncertainties

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Power Distribution Uncertainties

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Power Distribution Uncertainties

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Power Distribution Uncertainties

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Power Distribution Uncertainties

> Axial power distribution uncertainties were determined by the simple relationship
  - Nodal = radial * axial
  - $\delta \text{Nodal}^2 = \delta \text{radial}^2 + \delta \text{axial}^2$

> Axial uncertainty was determined to be 1.81 % for C-lattice plants and 2.91% for D-lattice plants

> Another component might be the radial uncertainty at an axial level

> The EMF-2158(P)(A) data was reevaluated by looking at the deviations between measured and calculated TIP response for each axial level

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Power Distribution Uncertainties

There does not appear to be any axial dependency on the standard deviation

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Purpose

> Gamma scans have been used to measure the assembly and individual rod power distribution

> These measurements are used to validate core physics methods and determine the associated uncertainties

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Gamma Scan Measurements

- Gamma scans measure the relative gamma flux resulting from isotopic decay
- Certain isotopes can be identified by gamma spectroscopy
- Power measurements target the gamma spectrum associated with La^{140}
- La^{140} is a decay product of Ba^{140} which is direct fission product

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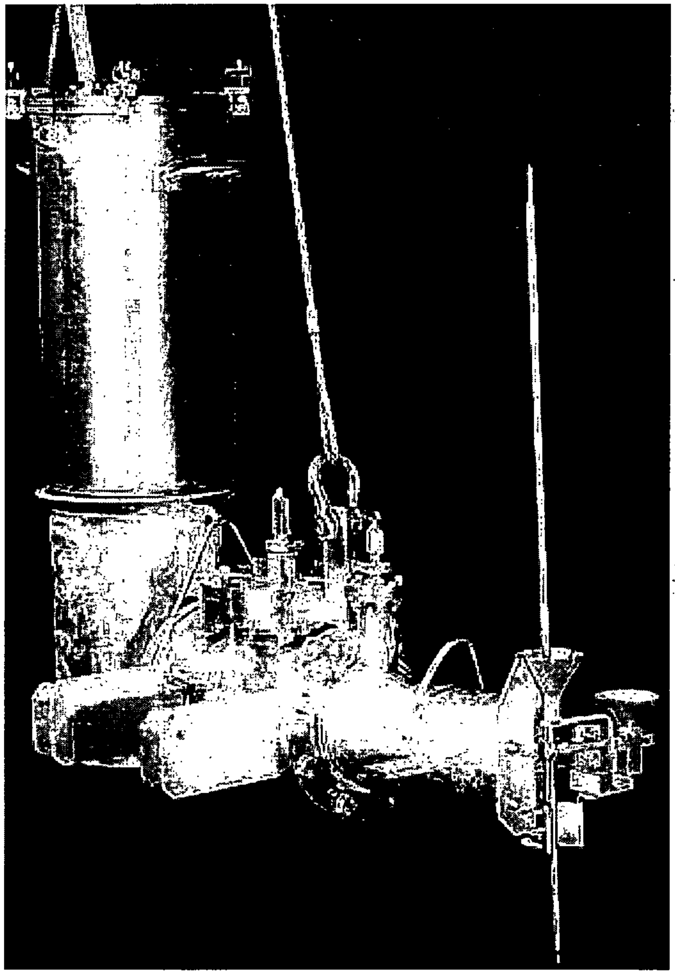
Gamma Scan Measurements

- The half life of Ba^{140} is 12.8 days
- The half life of La^{140} is 40 hours
- La^{140} activity is therefore related to the density of Ba^{140}
- The Ba^{140} density is representative of the integrated fissions over the last 25 days
- Gamma scan measurements need to be taken shortly after shutdown before the Ba^{140} decays to undetectable levels

Gamma Scan Equipment

> Equipment is tailored to the specific application
  - Assembly scans use a broad window to capture gamma particles from all of the rods
  - Individual rod scans use a narrow window to isolate the rod
  - An axial level measurement uses a broader (axial) window to get a higher count rate
  - Axial scans use a narrow (axial) window to get a finer resolution

Gamma Scan Equipment



- Gamma scan measurements are performed on individual fuel rods removed from assemblies using a high-purity germanium (HPGe) detector and an underwater collimator assembly

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Gamma Scan Comparisons

> In order to compare core physics models to the gamma scan results the calculated pin power distribution is converted into a Ba^{140} density distribution

- A mathematical process using CASMO-4 pin nuclide inventory and MICROBURN-B2 nuclide inventory is used
- This is an additional uncertainty in the overall comparison

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BWR Power Distribution Uncertainties

> There is very limited data on measured power distributions

> The measured power is determined by modifying the calculated power distribution using the measured and calculated LPRM values

- Measured LPRM values are calibrated to the TIP measurements

> Assembly gamma scan measurements at Quad Cities were used to define the uncertainty of the correlation coefficients

> These correlation coefficients indicate the accuracy of the "UPDATE" methodology

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BWR Power Distribution Uncertainties

> The Bundle Correlation Coefficient for QC Cycle 2 was [ ]

> The Bundle Correlation Coefficient for QC Cycle 4 was [ ]

> The average value of [ ] was used in the determination of the measured power uncertainty

> Using the minimum correlation coefficient increases the measured uncertainty by [ ]%

> Using the maximum correlation coefficient decreases the measured uncertainty by [ ]%

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Gamma Scan Data

> Pin-by-Pin Gamma scan data is used for verification of the local peaking uncertainty

> Quad Cities Data indicated that this uncertainty was approximately [ ]%

> KWU measurements of 9X9 and ATRIUM-10 assemblies provided additional validation that this uncertainty was accurate

> Comparisons to Monte Carlo calculations indicated an uncertainty of approximately [ ]%

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Quad Cities Gamma Scan Benchmark Results
EMF-2158(P)(A) pp. 8-6,7

This data includes measurement uncertainty.
Local power distribution uncertainty is not axial level dependent

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Local Peaking Uncertainty

- Recent gamma scan measurements including ATRIUM-10 show similar comparisons at various axial levels
- These results do not indicate any trend relative to axial position

KWU-S Gamma Scan Benchmark Results EMF-2158(P)(A) pp. 8-8

Local power distribution uncertainty is not axial level dependent

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KWU-S Gamma Scan Benchmark Results EMF-2158(P)(A)

> Full axial scans were performed on 16 fuel rods

> Comparisons to calculated data show excellent agreement at all axial levels

> The dip in power associated with spacers is not modeled in MICROBURN-B2

> There is no indication of reduced accuracy at higher void fractions

KWU-S Gamma Scan Benchmark Results EMF-2158(P)(A)

Measurements were performed for moderate void fractions

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KWU-S Gamma Scan Benchmark Results EMF-2158(P)(A)

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KWU-S Gamma Scan Benchmark Results EMF-2158(P)(A)

Indication that the higher voids are accurately represented

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KWU-S Gamma Scan Benchmark Results
EMF-2158(P)(A)

Indication that the higher voids are accurately represented

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KWU-S Gamma Scan Benchmark Results EMF-2158(P)(A)

Indication that the higher voids are accurately represented

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KWU-S Gamma Scan Benchmark Results EMF-2158(P)(A)

Indication that the higher voids are accurately represented

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Power Distribution Uncertainties

- Gamma scanning provides data on relative local and radial power during last few weeks of operation
- Uncertainty in gamma scan results has small effect on measured radial power distribution uncertainty
  - 50% decrease in correlation coefficient results in 0.4% increase in measured
    radial power distribution uncertainty
  - Additional ATRIUM-10 gamma scan data would not significantly affect measured power distribution uncertainty
- Local gamma scan data available for various designs
  - 11 assemblies in two reactors
  - 7x7, 8x8, 9x9, ATRIUM-10
  - Exposures include once- and twice-burned assemblies
  - Various gadolinia concentrations
  - Various water rod configurations
- No void dependence observed for local power uncertainties
- More ATRIUM-10 gamma scanning is not expected to change uncertainties

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BWR Power Distribution Uncertainty

> Conclusion

- Recent gamma scan data has confirmed the local power uncertainty
- There is no axial dependency in the uncertainty
- There is no void dependency in the local peaking power uncertainty
- Current uncertainties are applicable to Brunswick with power uprate conditions

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