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Meeting Agenda August 4, 2005

Topic	Time	Presenter
> **Introduction**	15	Holm
> **CASMO4/MB2 Methodology**	150	Grummer
◆ CASMO4 characterization of BWR lattice		
◆ MB2 nodal XSEC representation		
◆ Experience with high voids		
◆ Axial void distribution uncertainty		
◆ Recent gamma scan data		
> **Safety Analysis Methodology Uncertainties**		
◆ Treatment of Uncertainties in Safety Analyses	30	Garrett
◆ SLMCPR Overview	60	Garrett
◆ SLMCPR Sensitivity to Power Distribution Uncertainty		
> **Bypass Modeling**	30	Grummer
> **Summary**	15	Holm

CASMO-4/MICROBURN-B2 Methodology

Ralph Grummer
Manager, Core Physics Methods

Richland, WA August 4, 2005

BWR Methodology Applicability

> **Objective**

- **Describe the cross section re-construction process used by Framatome-ANP**
- **Demonstrate that the Framatome-ANP Methodology is accurate for high void conditions**

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**
- **Two-dimensional transport solution is based upon the Method of Characteristics**

CASMO-4 (cont.)

- Provides pin-by-pin power and exposure distributions
- Produces homogeneous multi-group (2) micro-scopic cross sections as well as macro-scopic 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

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

MICROBURN-B2 (cont.)

- A model for nodal burnup gradient
- A model for spectral history gradient
- Full three-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

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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

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

MICROBURN-B2 Cross Section Representation

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.

MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

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

MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

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 four dimensional representation can be reduced to three dimensions by looking at a single exposure

MICROBURN-B2 Cross Section Representation

This is a smooth well behaved surface

MICROBURN-B2 Cross Section Representation

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

MICROBURN-B2 Cross Section Representation

MICROBURN-B2 Cross Section Representation

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.

MICROBURN-B2 Cross Section Representation

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

MICROBURN-B2 Cross Section Representation

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

MICROBURN-B2 Cross Section Representation

> **Why not use higher void CASMO-4 depletions?**
  - **For example 0,45,90**

> **Introduces more error for intermediate void fractions.**

> **The following example shows the difference between a 0,40,80 and a 0,45,90 interpolation method**

MICROBURN-B2 Cross Section Representation

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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.)

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

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**

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

CASMO-4 / MICROBURN-B2 Methodology

> **Conclusion**

- **The methods used in CASMO-4 are state of the art**
- **The methods used in MICROBURN-B2 are state of the art**
- **The methodology accurately models a wide range of thermal hydraulic conditions**

BWR Methodology Experience

BWR Methodology Experience

> **Objective**
  - **Describe the experience base for Framatome-ANP methodologies**
  - **Demonstrate that the Framatome-ANP Methodology is Applicable to EPU conditions at Browns Ferry**

BWR Methodology Experience

- During the last meeting the range of assembly power and void fraction were presented
- Recent experience shows similar ranges of operation

Topical Report Thermal Hydraulic Conditions

Maximum assembly powers approaching 8 MW are in the benchmark database

Evaluation of Power Uprate for Browns Ferry

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

BWR Methodology Experience

Current Experience is consistent with the topical report

Topical Report Thermal Hydraulic Conditions

Maximum exit voids of 90% are in the benchmark database

Evaluation of Power Uprate for Browns Ferry

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

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**

Browns Ferry Current Design

Browns Ferry with Power Uprate

BWR Methodology Experience

Browns Ferry Current Design

Browns Ferry with Power Uprate

Nodal void fractions between 70 and 80 percent are most prevalent

BWR Methodology Experience

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

Experience with High Void Fractions

> **Conclusions**

- **Reactor conditions for Browns Ferry with power uprate is 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**

BWR Methodology
Power Distribution Uncertainties

Power Distribution Uncertainties

> **Objective**

- **Describe the process used by Framatome-ANP to define the power distribution uncertainties**
- **Demonstrate that the Framatome-ANP Methodology is Applicable to EPU conditions at Browns Ferry**

Power Distribution Uncertainties

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

Power Distribution Uncertainties

Power Distribution Uncertainties

Power Distribution Uncertainties

Power Distribution Uncertainties

Power Distribution Uncertainties

- Axial power distribution uncertainties were determined by the simple relationship
  - Nodal = radial * axial
  - $\delta Nodal^2 = \delta radial^2 + \delta axial^2$
- 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 re-evaluated by looking at the deviations between measured and calculated TIP response for each axial level

Power Distribution Uncertainties

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

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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 [ ]%

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

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

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

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

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

Indication that the higher voids are accurately represented

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

Indication that the higher voids are accurately represented

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

Indication that the higher voids are accurately represented

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

Indication that the higher voids are accurately represented

Axial Void 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 Browns Ferry with power uprate conditions**

Gamma Scan Description

Ralph Grummer
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Richland, WA August 4, 2005

Gamma Scans

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

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

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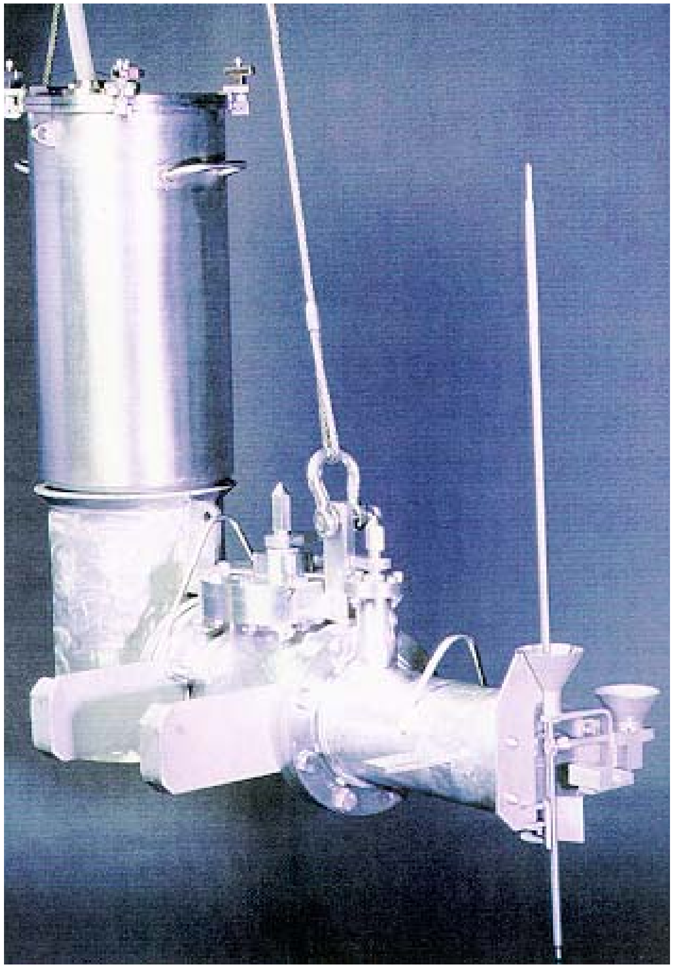
Gamma Scan Equipment

> **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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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

No more ATRIUM-10 gamma scanning is necessary

Safety Analysis Methodology Uncertainties

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Presentation Topics

> Treatment of Uncertainties in Safety Analysis
  - Deterministic safety analysis approach

> Safety Limit MCPR (SLMCPR) Methodology Overview

> SLMCPR Sensitivity to Power Distribution Uncertainty
  - Local power peaking
  - Radial power peaking
  - Axial power peaking

Treatment of Uncertainties in Safety Analysis

Safety Analysis Methodology
Treatment of Uncertainties

> The MCPR safety limit methodology explicitly considers important uncertainties in the Mont Carlo calculations performed to determine the number of rods in boiling transition

> Other safety analysis methodologies do not explicitly account for uncertainties; deterministic, bounding approaches are used to ensure that all licensing criteria are satisfied

Safety Analysis Methodology Deterministic Approach

- Current deterministic methods are not best estimate
  - Individual phenomena are not treated statistically
- Current methods provide conservative, bounding results
- Current methods have adequate conservatism to offset methodology uncertainties
- Conservatism incorporated in two ways
  - Computer code models produce conservative results on an integral basis
  - Important input parameters are conservatively bounding
- All conservatisms are additive and not statistically combined
  - Assuming all parameters are bounding at the same time produces very conservative results

Safety Analysis Methodology

Examples of Analysis Conservatism for Limiting Events

Pressurization Events

- COTRANSA2 conservative prediction of Peach Bottom turbine trip tests
- Steady-state CPR correlation demonstrated to be conservative for transients (predicted dryout time occurs earlier than test data)

Safety Analysis Methodology

Examples of Analysis Conservatism for Limiting Events

Pressurization Events *(continued)*

- The four steam lines are represented as a single, average steam line
  - Accounting for differences causes the pressurization rate to be reduced
- Bounding scram insertion times (delay and insertion rate)
- All control blades assumed to insert at the same time and rate
  - Control blades actually insert at a distribution of speeds
  - Control blades faster than average provide more negative reactivity than lost by control blades slower than average
- All control rods assumed to be initially fully withdrawn (conservative for off-rated conditions and pre-EOC exposures)

Safety Analysis Methodology
Examples of Analysis Conservatism for Limiting Events

Pressurization Events *(continued)*

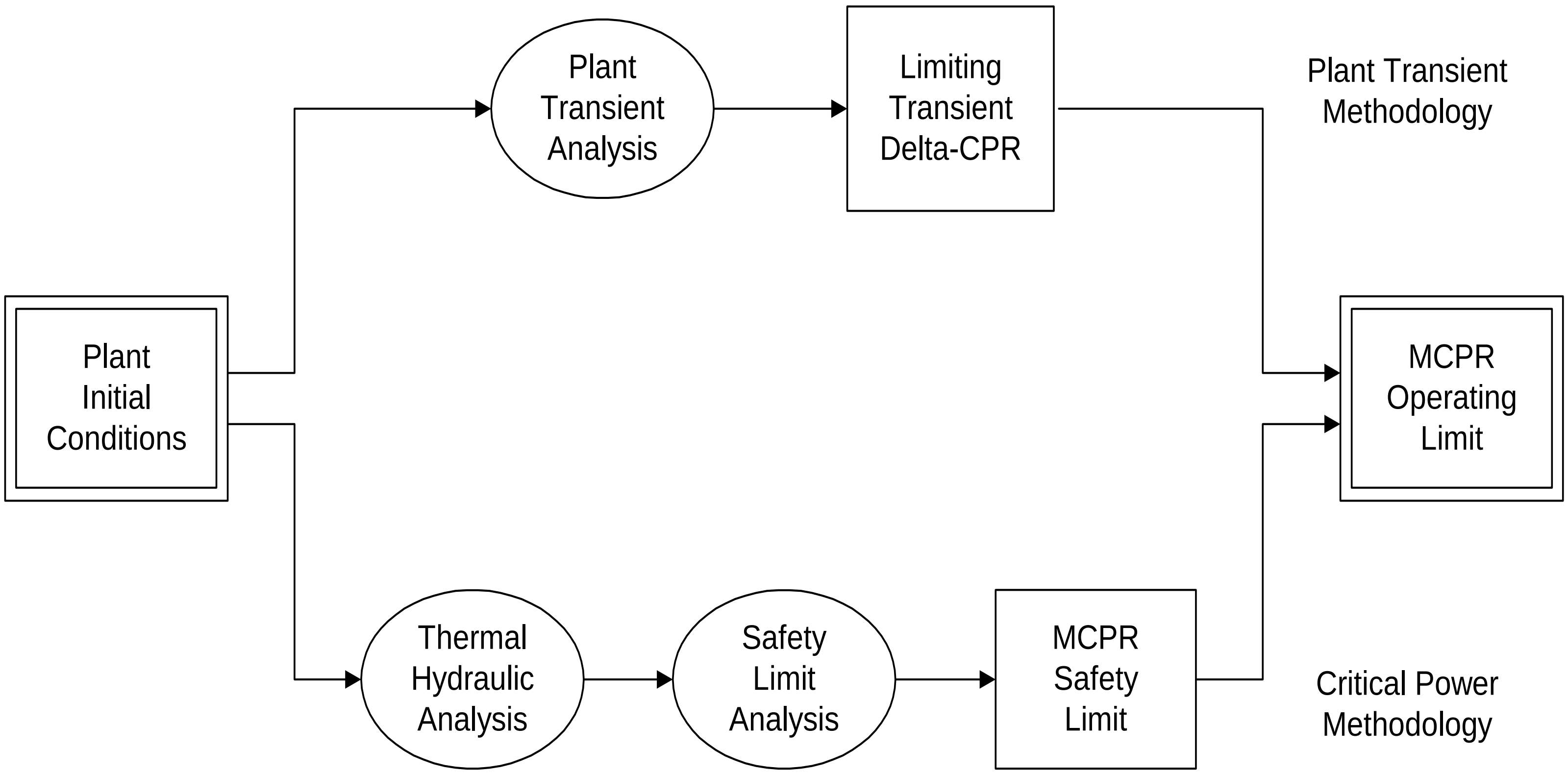
- Conservative licensing basis step-through used for neutronics input
  - More top-peaked axial power shape than design basis
  - Longer cycle exposure than design basis
- Bounding setpoints (analytical limits) and delays used
  - Reactor protection system
  - Turbine protection system
- Bounding equipment performance assumed
  - Turbine control and stop valve closure times
  - RPT delay time
  - Turbine bypass
  - Safety and relief valves

Safety Limit MCPR (SLMCPR) Methodology Overview

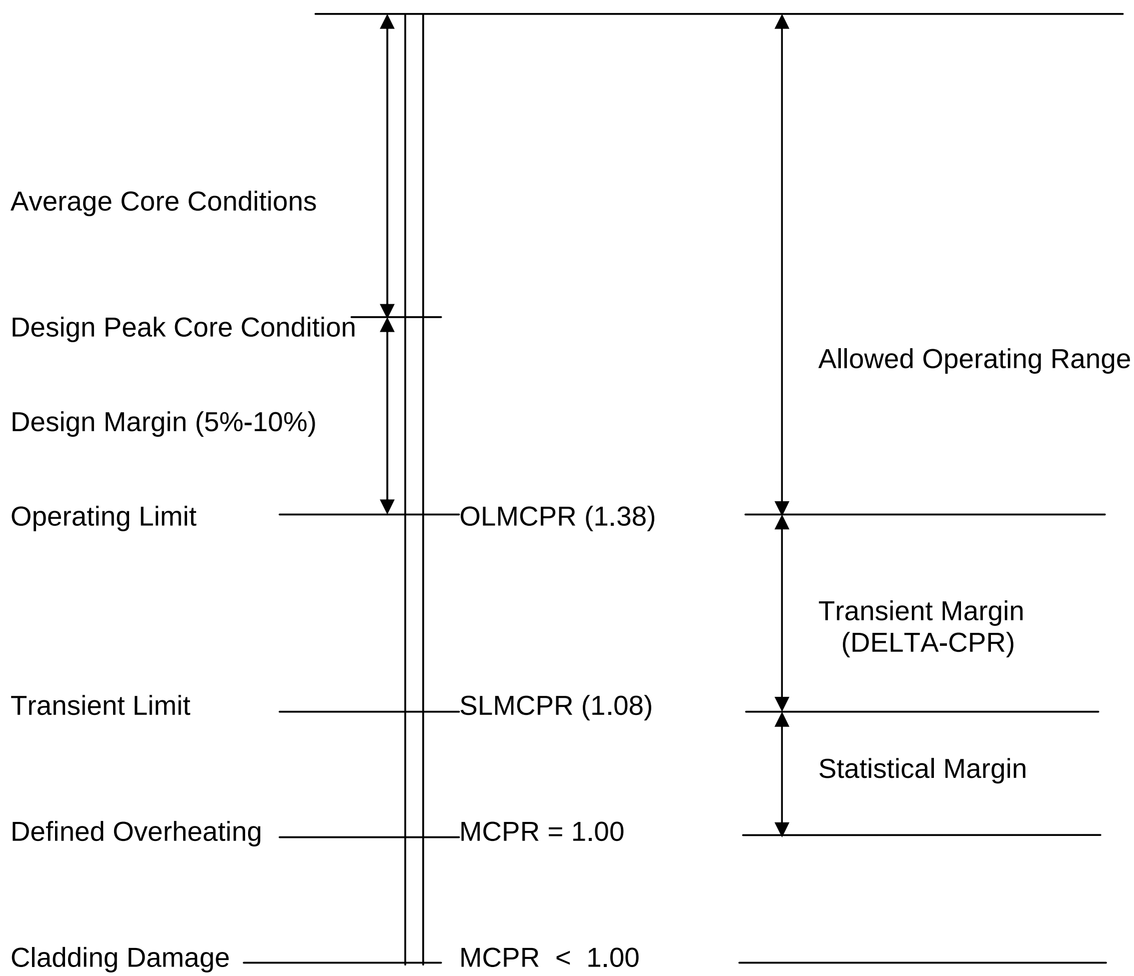
SLMCPR Analysis Methodology

- 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 (transients)
- At least 99.9% of the 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

SLMCPR Analysis Methodology



Thermal Limits Methodology



SLMCPR Analysis Methodology

Major Computer Codes

Code	Use
MICROBURN-B2	Provides radial peaking factor and exposure for each bundle in the core and the core average axial power shape
CASMO-4	Provides local peaking factor distribution for each fuel type
XCOBRA	Provides hydraulic demand curves for each fuel type
SLPREP	Automation code which obtains neutronic data from MICROBURN-B2 and CASMO-4 and prepares SAFLIM2 input
SAFLIM2	Calculates the fraction of rods in boiling transition (BT) for a specified SLMCPR

SLMCPR Analysis Methodology

SLMCPR Analysis Methodology

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

SLMCPR Analysis Methodology
SAFLIM2 Computer Code

Description	SAFLIM2 is a computer code used to determine the number of fuel rods in the core expected to experience boiling transition for a specified core MCPR
Use	Evaluate the safety limit MCPR (SLMCPR) which ensures that at least 99.9% of the fuel rods in the core are expected to have a MCPR value greater than 1.0
Documentation	EMF-2392(P), *SAFLIM2 Theory, Programmer's, and User's Manual*
Acceptability	ANF-524(P)(A) Rev 2 and Supplements, *ANF Critical Power Methodology for Boiling Water Reactors,* November 1990 The safety evaluation by the NRC for the topical report approves the SAFLIM2 methodology for licensing applications

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SAFLIM2 Computer Code

Major Features

- Convolution of uncertainties via a Monte Carlo technique
- Consistent with POWERPLEX® CMSS calculation of MCPR
- Appropriate critical power correlation used directly to determine if a rod is in boiling transition
- 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

SLMCPR Statistical Parameters

SAFLIM2 Computer Code

Reactor System Uncertainties

SFW **Feedwater flow rate uncertainty.** Obtained from NSSS vendor documentation or customer. A typical value is 1.8%

SFWT **Feedwater temperature uncertainty.** Obtained from NSSS vendor documentation or customer. A typical value is 0.8%

SP **Core pressure uncertainty.** Obtained from NSSS vendor documentation or customer. A typical value is 0.7%

SCG **Total core flow rate uncertainty.** Obtained from NSSS vendor documentation or customer. A typical value is 2.5%

SAFLIM2 Computer Code

Core Monitoring Uncertainties

SAFLIM2 Computer Code

Fuel Design Uncertainties

SPCB Critical Power Correlation

F-eff

SAFLIM2 Computer Code
Calculation Procedure

- > Initialization
- > Monte Carlo Trials
  - Core Calculations (Outer Loop)
  - Fuel Assembly Calculations (Inner Loop)
- > Rods in BT Calculation

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
  -

SAFLIM2 Computer Code

Initialization (continued)

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SAFLIM2 Computer Code

Core Calculations - Outer Loop

SAFLIM2 Computer Code

Assembly Calculations - Inner Loop

SAFLIM2 Computer Code

Fuel Rod Calculations - Inner Loop

SAFLIM2 Computer Code

Number of Rods in BT

SAFLIM2 Computer Code
Monte Carlo Trial

SLMCPR Sensitivity to Power Distribution Uncertainty

SLMCPR Sensitivity

Power Distribution Uncertainty

Topics

> Sensitivity to radial peaking factor (RPF) and local peaking factor (LPF) uncertainty
  - Conservative range for potential changes in RPF and LPF uncertainties from additional gamma scan data were estimated
  - LPF uncertainty: less than 1.5x current estimate
  - RPF uncertainty: -0.3% to +0.4% change in current estimate

> Basis for not explicitly modeling axial power shape uncertainty

SLMCPR Sensitivity

Local Peaking Factor (LPF) Uncertainty

> Sensitivity analyses performed using Browns Ferry equilibrium ATRIUM™-10 EPU core design

> LPF uncertainty increased by 1.5 multiplier

> SLMCPR — σlpf — Rods in BT

SLMCPR	σlpf	Rods in BT
1.08	1.48%	60
1.08	2.22%	62
1.0810	2.22%	60

> SLMCPR insensitive (+0.001) to 1.5x increase in LPF uncertainty

Additional gamma scan data not expected to result in significant impact to SLMCPR

SLMCPR Sensitivity

Radial Peaking Factor (RPF) Uncertainty

- Sensitivity analyses performed using Browns Ferry equilibrium ATRIUM™-10 EPU core design
- RPF uncertainty increased 0.4%
-

SLMCPR	σ_{rpf}	Rods in BT
1.08	4.6%	60
1.08	5.0%	71
1.0855	5.0%	60

- SLMCPR not very sensitive (+0.0055) to 0.4% increase in RPF uncertainty

Additional gamma scan data not expected to result in significant impact to SLMCPR

SLMCPR Methodology

Axial Power Shape

SLMCPR Sensitivity

Axial Power Shape Assessment for ATRIUM™-10

- Original methodology assessment performed for fuel designs without part-length fuel rods and with ANFB critical power correlation
- CHF tests indicated ATRIUM™-10 fuel more sensitive to axial power shape
- SLMCPR sensitivity to axial power was reassessed (1998)
- Three types of assessments performed
  - Variations in core average axial power shape
  - Use of assembly specific axial power shape for each assembly
  - Perturbing power shape during Monte Carlo trials

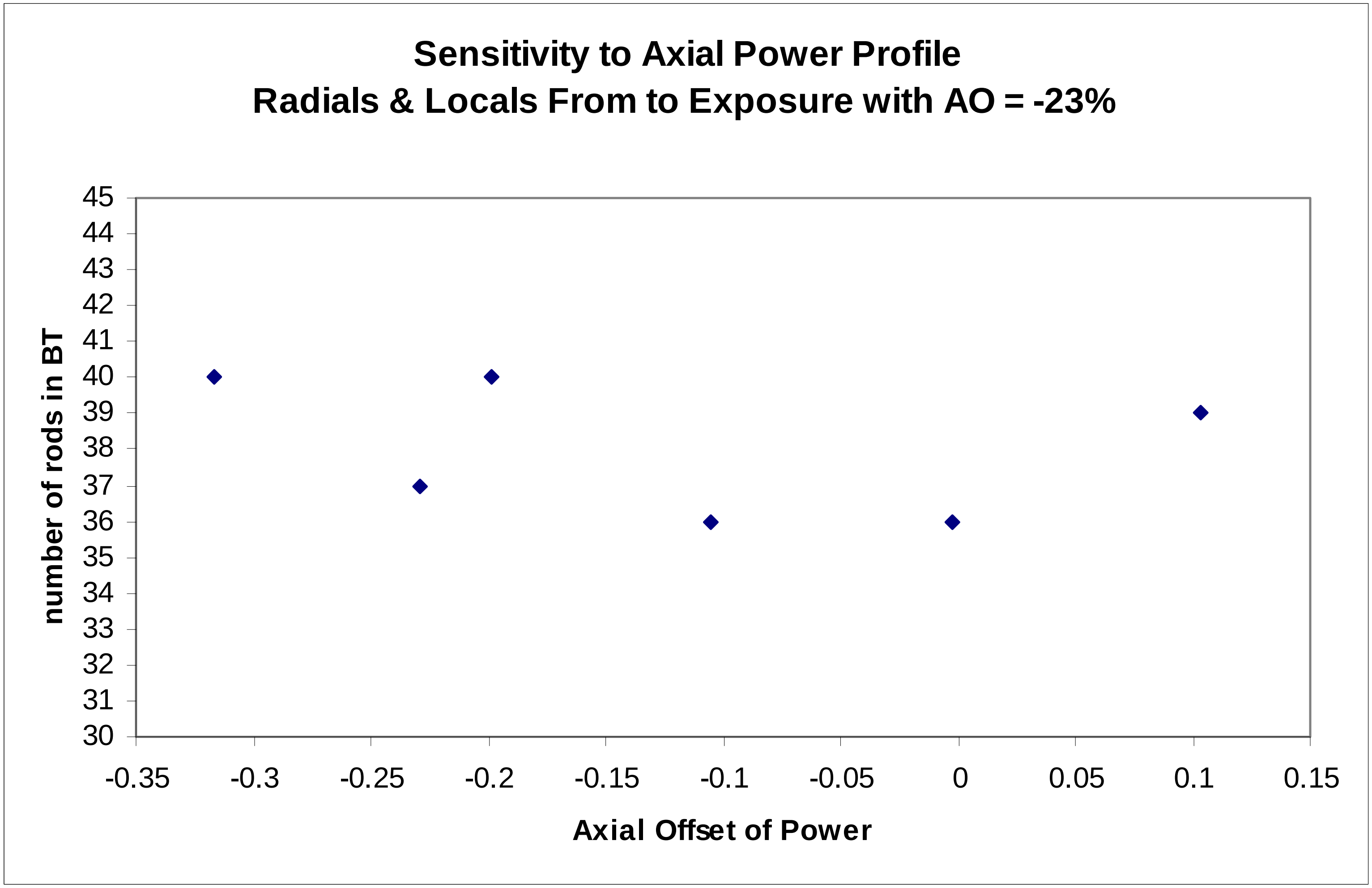
SLMCPR Sensitivity

Axial Power Shape Assessment for ATRIUM™-10 (continued)

> Variation in core average axial power shape
  - Range of core average axial power shapes obtained from a core design analysis
  - SLMCPR analysis performed for each axial shape with all other input parameters held constant
  - Variation observed in BT rods typical of Monte Carlo process; no trend with changes in axial power shape

Number of rods in BT is not sensitive to changes in core average axial power shape

Sensitivity to Axial Power Profile



SLMCPR Sensitivity

Axial Power Shape Assessment for ATRIUM-10 (continued)

> Use of assembly-specific axial power shape

- Special code version developed with capability to model a different axial power shape for each assembly
- Axial power distribution obtained from cycle design step-through for each assembly in the core
- Rods in BT calculated for each bundle based on bundle-specific axial power shape

Number of rods in BT is not sensitive to the use of core average or bundle-specific power distribution

Sensitivity of Modeling Assembly-Specific Axial Power Profile

Core Flow Mlb/hr	Number of Rods in BT (maximum from all exposures)	
	Core Average Axial Power	Assembly-Specific Axial Power
108	35	34
70	43	46

Conclusion: ***Results are within normal variation for Monte Carlo results***

SLMCPR Sensitivity

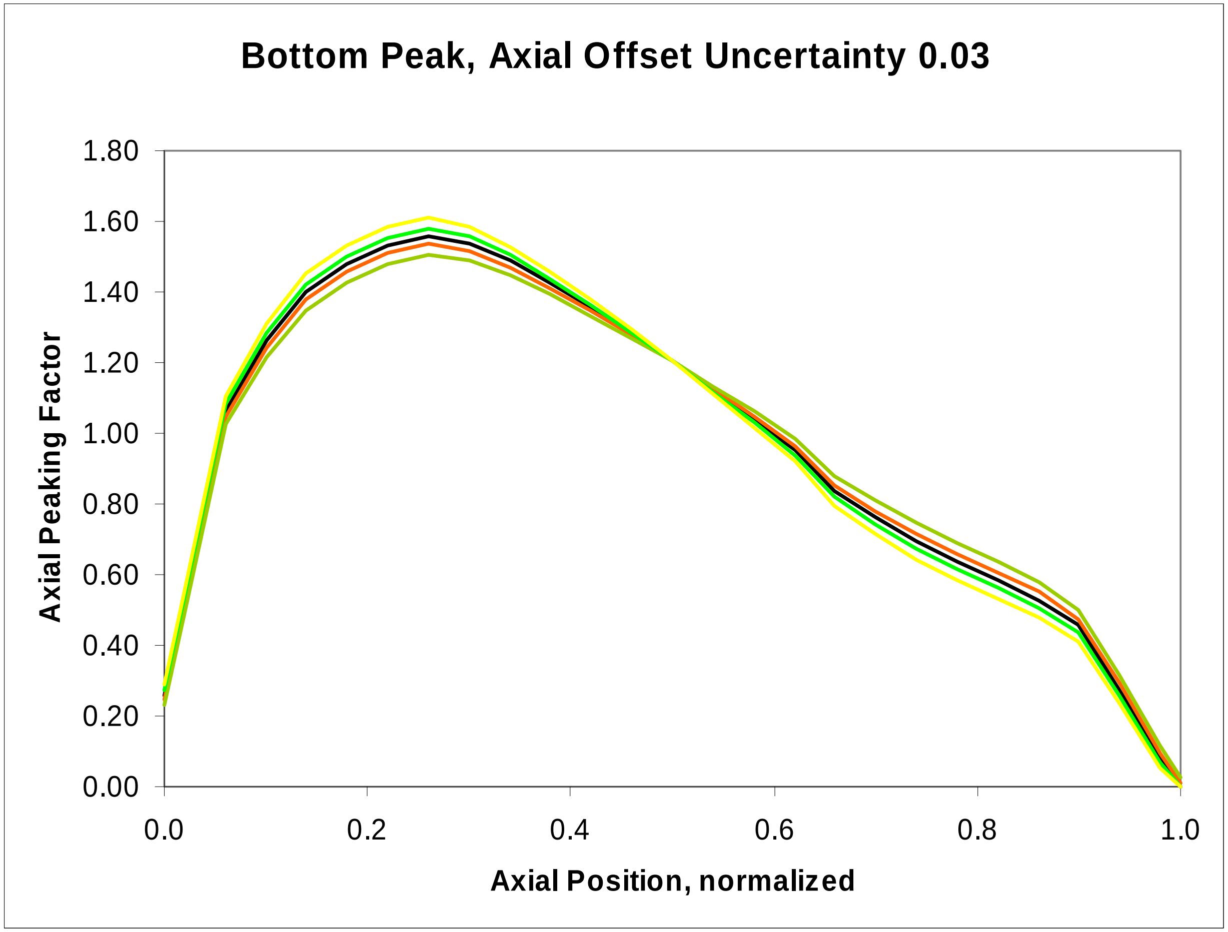
Axial Power Shape Assessment for ATRIUM™-10 (continued)

> Perturbing power shape during Monte Carlo trials

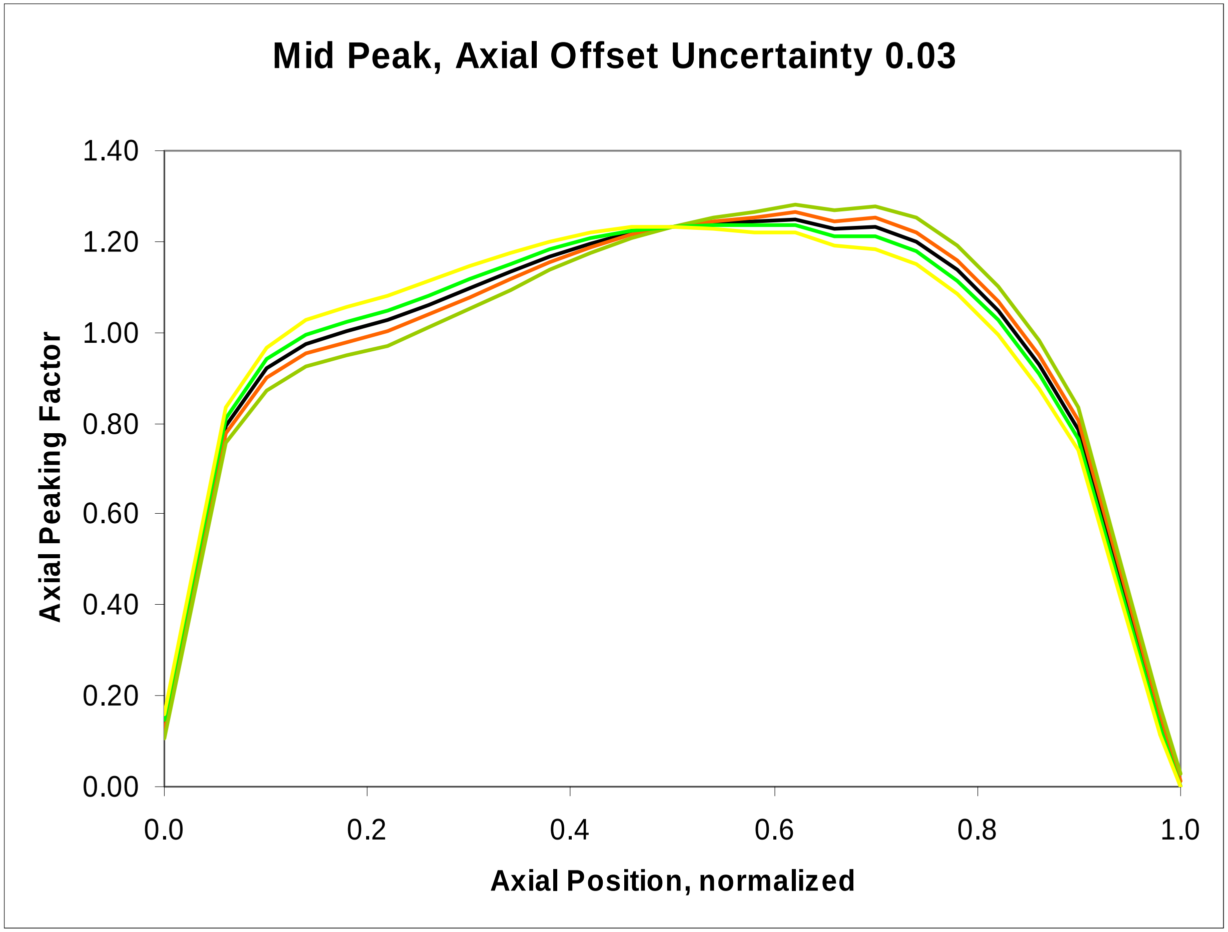
- Special code version developed with capability to perturb the core average axial power shape during each Monte Carlo trial
- The code used a process to adjust the initial axial power shape to produce a power shape with a different axial offset
- Axial power uncertainty reported in the MICROBURN-B2 topical report is 1.8% for C-lattice and 2.9% for D-lattice
- Analyses performed assuming an axial power offset uncertainty of 3%
- Results showed little variation in the number of rods in BT

Number of rods in BT is not sensitive to perturbing the axial power shape in Monte Carlo trials

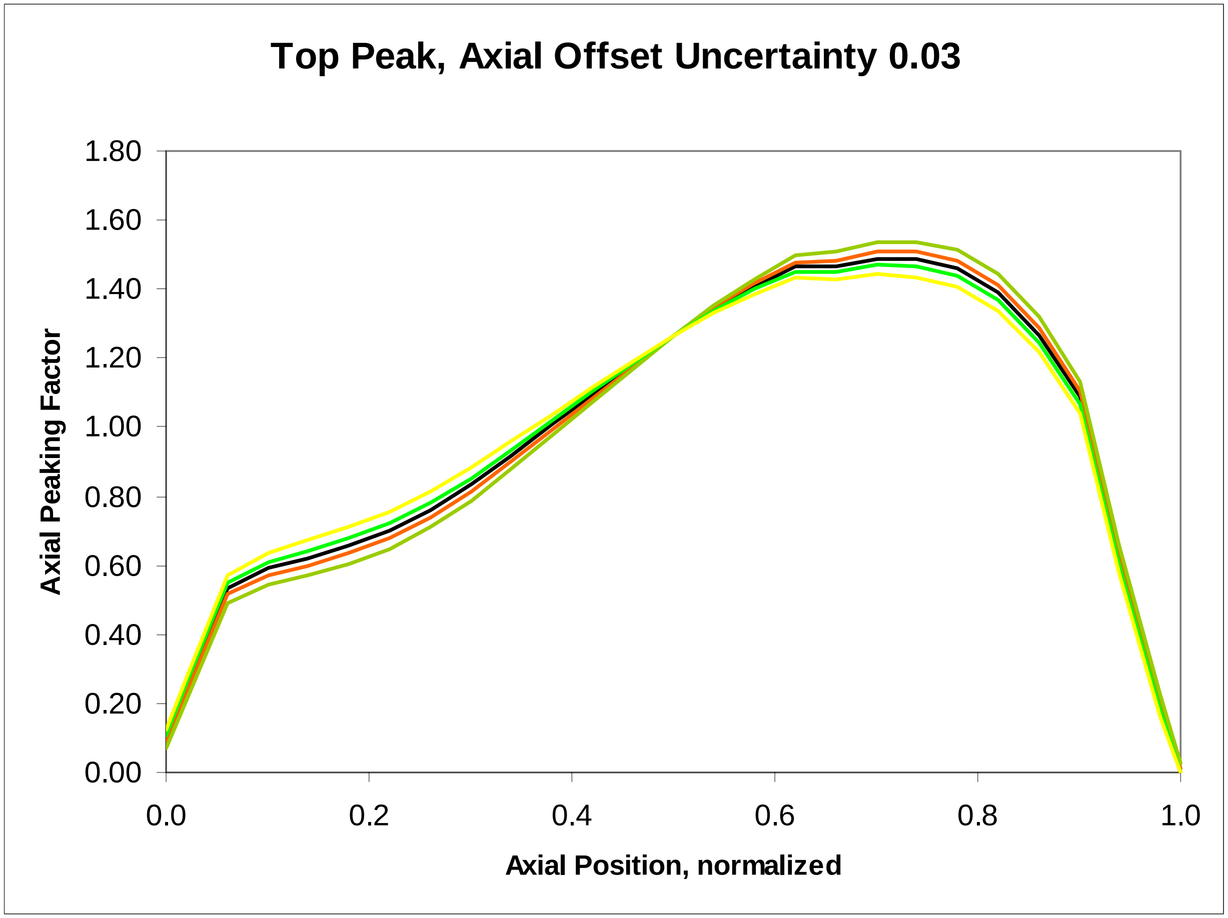
Change Axial During Monte Carlo Trials



Change Axial During Monte Carlo Trials



Change Axial During Monte Carlo Trials



Sensitivity to Changing Axial Power During
Monte Carlo Trials

Axial Power Shape	Rods in BT	
	Constant Core Average Axial	Perturb Core Average Axial
Bottom peak	27	29
Middle peak	22	22
Top peak	19	17

Conclusion: ***Results are within normal variation for Monte Carlo results***

SLMCPR Sensitivity

Axial Power Shape Assessment for ATRIUM™ -10 (continued)

Conclusion from 1998 assessment

> SLMCPR methodology remains insensitive to axial power shape and axial power shape uncertainty

> Approved methodology is applicable for ATRIUM™ -10 fuel

Bypass Modeling

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Modeling Voiding in the bypass

> **Objective**

- **Demonstrate that anticipated boiling in the bypass does not impact the safety margins**
- **Demonstrate that the Framatome-ANP Methodology is Applicable to EPU conditions at Browns Ferry**

AREVA

Modeling Voiding in the bypass

- Calculations for Browns Ferry with Power Uprate do not indicate boiling in the bypass at rated power conditions
  - With single lumped bypass channel
  - With multi-channel bypass and explicit water rod models
- Browns Ferry has 10% more inlet subcooling than similar plants due to lower feedwater temperature

Multi-Channel Bypass Model

- In order to evaluate the effect of voiding in the bypass a theoretical case was developed
- Voiding in the bypass was forced to 5% voids by decreasing the inlet sub-cooling from 27.15 to 15 BTU/lbm
- The multi-channel bypass produces conservative results
  - Multi-channel bypass model is an independent flow path for each assembly
  - The boundary condition is equal pressure drop from inlet to exit
  - No cross flow between bypass channels
  - Heat deposition based upon single assembly
  - No Gamma smearing

AREVA

Bypass Void Distribution

EDIT OF AXIALLY AVERAGED VOID FRACTION IN BYPASS CHANNEL **IN UNITS OF %**

J:	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
I:															
1															
2															
3															
4															
5															
6															
7															
8															
9															
10															
11															
12															
13															
14															
15															

Bypass Void Distribution

EDIT OF VOID FRACTION IN BYPASS CHANNEL IN UNITS OF %

Axial	Level	Multi Channel: Core Average	Multi Channel: Peak Assembly	Single Channel
Top	1			
	2			
	3			
	4			
	5			
	6			
	7			
	8			
	9			
	10			
	11			
	12			
	13			
	14			
	15			
	16			
	17			
	18			
	19			
	20			
	21			
	22			
	23			
	24			
Bottom	25			

Modeling Voiding in the bypass

There is a minimal change in the power distribution of the peak assembly

Voiding in the Bypass

> **Conclusion**

- **Boiling in the bypass is not expected at rated power with power uprate conditions**
- **The effects of boiling in the bypass, should it occur are very small with exit void fraction of 5%**
- **Voiding in the bypass has a negligible impact on the LPRM instrumentation as the void fraction is near 1% at the top most LPRM**