Application Methodology for BWROG Stability Limit Analysis

Presentation to NRC

February 20, 2003

1 Introduction

Purpose

Scope

Objective

2 Requirements

Licensing requirements

Other requirements

3 Documentation

TRACG Model Description

TRACG Qualification

PRIME Thermal/Mechanical Model Description and Qualification

Vendor Specific Single-Channel Thermal/Hydraulic and Fuel Rod Thermal/Mechanical Model Description and Qualification

Application Licensing Topical Report

1. Purpose, Scope and Objectives
Application for BWROG Stability Limit Analysis

- **Define generic stability limit to meet GDC 12**
- **Generic limit for all D&S plants and all currently used fuel types**
- **Events: Reactor Instability**
  - Reactor instability during operation at low flow and high power/flow ratio
- **Documentation**
  - Licensing Application Framework for BWROG Stability Limit Analysis
  - TRACG Model Description LTR, NEDE-32176, Revision 3
  - TRACG Qualification LTR, NEDE-32177, Revision 3
  - Vendor Specific Model Description and Qualification LTRs for Single Channel Thermal/Hydraulic and Fuel Rod Thermal/Mechanical Codes.
  - Application LTR for BWROG Stability Limit Analysis
- **Review Scope**
  - One SER for Application Methodology for BWROG Stability Limit Analysis
    - Applicability of TRACG for BWR Stability
    - Applicability of Vendor Proprietary Models for Single Channel Thermal/Hydraulic and Fuel Rod Thermal/Mechanical Analysis
    - Application Methodology for BWROG Stability Limit Analysis

BWRs, Operating Domains, and Fuel Types

- **Bound US BWR types**
  - BWR/2
    - Non Jet Pump Plants with External Recirculation Loops
  - BWR/3/4
    - Jet Pump Plants with Motor/Generator Recirculation Flow Control
  - BWR/5/6
    - Jet Pump Plants with Valve Recirculation Flow Control or Adjustable Speed Drives
- **Bound current operating domains**
- **Bound current fuel types**
  - GNF: GE11, GE12, GE13, GE14
  - Framatome: ATRIUM-9, ATRIUM-10
  - Westinghouse: SVEA-96, SVEA-96+, SVEA-96 Optima2
- **Develop a process for evaluating new operating domains and fuel types**

T/H Instability Events

Same Events as Currently Analyzed for Stability

- **Instability during operation at low flow and high power/flow ratio**
  - Natural circulation
  - Minimum pump speed
- **Most limiting event**
  - Oscillation at or just below the OPRM setpoint
  - No mitigating actions for 30 minutes
  - Fuel may experience repeated boiling transition and rewet
- **Other instabilities that exceed the OPRM set point**
  - Large growth rate
  - OPRM based scram will terminate event
  - Fuel may experience boiling transition
  - Expected to be less severe due to short duration

NRC Review Scope (*see Table 1-1*)

Review Element	Comment	Review Status
Licensing Application Framework	Outlines process and defines regulatory requirements and guidelines	For Information Only
TRACG Model Description NEDE-32176P Rev 3	Proprietary model addition and modifications for application to LOCA and stability	Rev 2 reviewed and approved by NRC. Rev 3 needs NRC review and approval
TRACG Qualification NEDE-32177P Rev 3	Additional proprietary qualification for LOCA and stability	Rev 2 reviewed and approved by NRC. Rev 3 needs NRC review and approval
Vendor Specific Single-Channel T/H and Fuel Rod T/M Model Description and Qualification	Proprietary basis for single-channel T/H and fuel rod T/M models	Need NRC review and approval
Application Methodology	Application methodology consistent with CSAU methodology	Needs NRC review and approval

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2.0 Requirements

• Licensing Requirement

– 10 CFR50 Appendix A

GDC 12 requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits (SAFDLs) are not possible or can be reliably and readily detected and suppressed.

– SAFDLs are not exceeded
  - No Fuel Failures
  - Negligible impact on fuel and cladding properties
    Basis for design and licensing analysis not affected

"You could not tell from an examination of the fuel, if an instability had occurred"

– 10 CFR50 Appendix B
Q/A Requirements

2.0 Requirements

- **Regulatory Guides**

Code Scaling, Applicability and Uncertainty (CSAU)

CSAU Step	Description
1	Scenario Specification
2	Nuclear Power Plant Selection
3	Phenomena Identification and Ranking
4	Frozen Code Version Selection
5	Code Documentation
6	Determination of Code Applicability
7	Establishment of Assessment Matrix
8	Nuclear Power Plant Nodalization Definition
9	Definition of Code and Experimental Accuracy
10	Determination of Effect of Scale
11	Determination of the Effect of Reactor Input Parameters and State
12	Performance of Nuclear Power Plant Sensitivity Calculations
13	Determination of Combined Bias and Uncertainty
14	Determination of Total Uncertainty

Specified Acceptable Fuel Design Limits (SAFDL)

Table 2-1

Phenomena where SAFDLs Required	Critical Parameters	Comments
Cladding and channel stress and strain, including: –creep deformations –annealing of irradiation hardening –pellet cladding mechanical interaction	1. Change in clad stress strain curve 7. Clad temperature	Clad stress/strain curve depends mainly on clad temperature as does annealing.
Cladding fatigue	3. Incremental clad fatigue	Fatigue depends primarily on the number of cycles.
Cladding oxidation	4. Incremental clad oxidation	Oxidation is a strong function of clad temperature.
Dimensional changes (fuel rod growth, cladding collapse)	2. Incremental clad strain	Clad strain depends primarily on the gap size and the fission gas pressure both which change with temperature.
Increased fission gas release and fuel rod internal pressure	5. Incremental fission gas pressure (release from pellet)	Fission gas release mechanisms are strong functions of fuel pellet temperature changes.
Fuel centerline melting	6. Fuel centerline temperature	Fuel temperatures depend on power and heat transfer.

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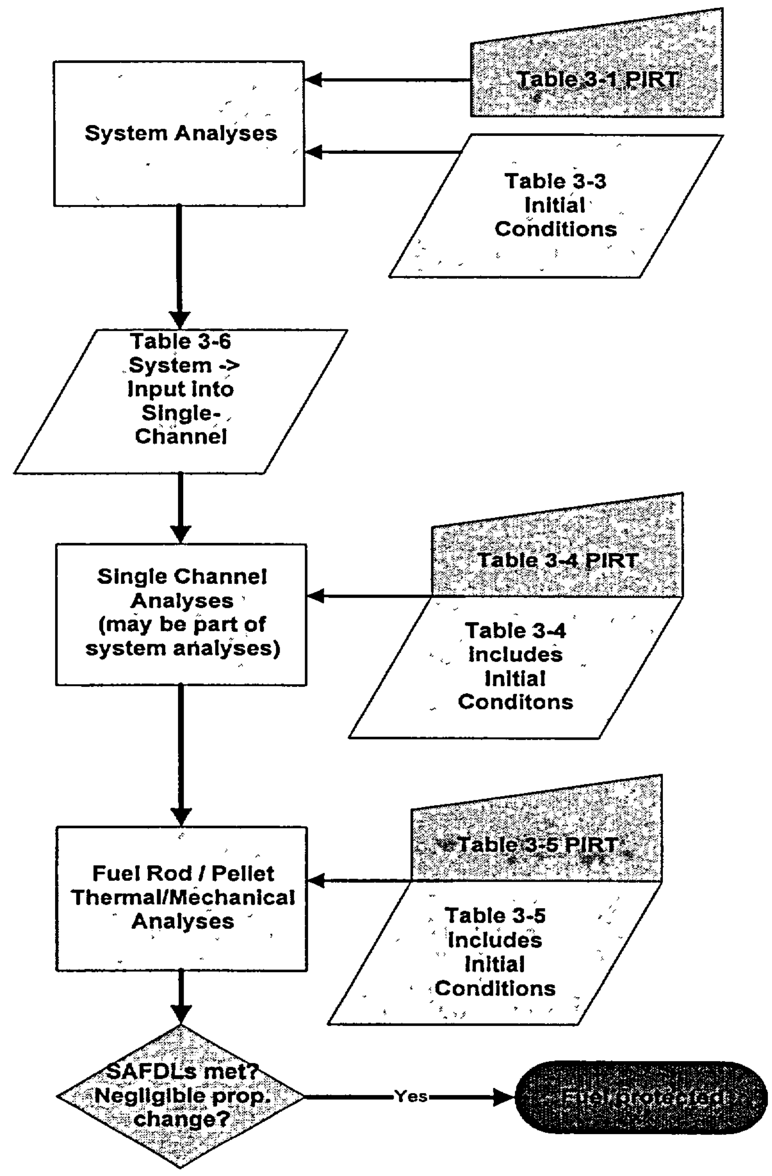
Relationship Between Process Steps and PIRTs
CSAU Step 3

Figure 3-1 (simplified)



Thermal / Mechanical PIRT (High ranked only) - CSAU Step 3

ID	TABLE 3-5: *PIRT and Initial Conditions for Fuel Rod / Pellet Thermal/Mechanical Analyses Subject to Core Stability Phenomena* PHENOMENA DESCRIPTION	Nucleate Boiling	Post-BT Heat Transfer	Highest Ranking	Critical Parameter	**Critical Parameters** 1. Change in clad stress strain curve 2. Incremental clad strain 3. Incremental clad fatigue 4. Incremental clad oxidation 5. Incremental fission gas pressure (release from pellet) 6. Fuel centerline temperature 7. Clad temperature Other core components, i.e., channel box, control blades, spacers, spacer tabs, etc. are not considered here.
	Fuel Rod / Pellet Phenomena					
C3AX	PELLET HEAT DISTRIBUTION	M	H	H	5,6	Affects fuel rod power, temperature and surface heat flux. Determines effective fuel time constant.
C3BX	PELLET HEAT TRANSFER PARAMETERS	M	H	H	6,7	Affects fuel rod power, temperature and surface heat flux. Determines effective fuel time constant.
C3CX	GAP CONDUCTANCE	M	H	H	6,7	Affects fuel rod power, temperature and surface heat flux. Determines effective fuel time constant.
C18	CLAD STRAIN	H	H	H	2,3	Cladding strain is not expected to be a significant parameter for stability; however, it is important to calculate it.
TM2	FISSION GAS RELEASE	H	H	H	2,5,6	Incremental fission gas is not expected to be a significant; however, it is important to calculate it.
TM3	ANNEALING OF IRRADIATION HARDENING	H	H	H	1,2	Expectation is "L" if nucleate boiling is re-established every oscillation so temperatures do not continue to increase.
TM4	PELLET THERMAL EXPANSION / CONTRACTION	H	H	H	2,3	
TM5	PEAK PELLET POWER DISTRIBUTION CHANGE	H	H	H	2,5,6,7	Related to local peaking (IC11) used for thermal/hydraulic analyses.

Thermal/Mechanical Critical Parameters are Controlled by Power and Temperature Oscillations

System Thermal / Hydraulic PIRT - CSAU Step 3
(partial listing of High ranked phenomena)

ID	TABLE 3-1: *PIRT for System Thermal/Hydraulic Analyses of Stability Phenomena* REGION or PHENOMENA DESCRIPTION	Core-wide Stability	Regional Stability	Highest Ranking	Critical Parameter	1. Thermal Margin and/or PCT Controlled by heat flux, flow, pressure, and inlet subcooling. - Power oscillations - Flow oscillations 2. Power Oscillation Amplitude/Frequency – Controls stability margin/ growth rate of perturbations which for this application is assumed to be a limit cycle oscillation (i.e , DR=1 0)	?	←*Can Phenomenon be affected by core design, fuel type, or plant type?* Comments on dependency of phenomena to core design, fuel type, plant design
B	BYPASS							
B6	CHANNEL-BYPASS LEAKAGE FLOW	H	H	H	1,2	Affects channel flow, voids, and thermal margin	Yes	
C	CORE / BUNDLE							
C1AX	VOID COEFFICIENT	H	H	H	2	Determines reactivity and power due to void fraction change. Determines forward loop "gain" for void perturbations	Yes	
C1DX	3-D KINETICS (CORE POWER DISTRIBUTION DURING TRANSIENT - INCLUDES AXIAL AND RADIAL POWER DISTRIBUTION)	M	H	H	1,2	Power distribution from 3D reactivity distribution affects total power, hot region power, axial power shape and CPR 3 D effects primarily important for regional evaluations.	Yes	
C1FX	SUBCRITICALITY OF FIRST HARMONIC MODE	N/A	H	H	2	Neutronics "damping" offsets thermal hydraulic gain for regional mode	Yes	Dependent on core design.
C12	NATURAL CIRCULATION FLOWS	H	H	H	1,2	Affects channel flow and thermal margin	Yes	
C13	DRYOUT / BOILING TRANSITION	H	H	H	1	Important for determining thermal margin	Yes	Uncertainty in predicting BT depends on fuel type
C15	FILM BOILING / HIGH VOID / FILM DRYOUT	H	H	H	1	Post-BT heat transfer impacts clad temperature.	No	
C19X	TMIN (MINIMUM STABLE FILM BOILING TEMPERATURE)	H	H	H	1	TMIN impacts the ability to rewet.	No	
C20	REWETTING BASED ON QUALITY	H	H	H	1	Impacts heat transfer mode.	No	
I	SEPARATOR							
I3	SEPARATOR PRESSURE DROP	H	H	H	1,2	Affects total core flow and level. Separator pressure drop affects core stability evaluation	Yes	Depends on plant type

Model Capability - CSAU Step 6

- **Structure**
  - Capability to model plant geometry
- **Basic Equations**
  - Capability to address global processes
- **Models and Correlations**
  - Capability to model and scale individual processes
- **Numerical Methods**
  - Capability to perform efficient and reliable calculations

- **Cross Correlation Against PIRT**
  - Demonstrate models are capable of simulating all high ranked phenomena for the identified event

Model Qualification - CSAU Steps 7-10

- **Qualification Strategy**
  - Separate Effects Tests
  - Component Performance Data
  - Integral System Effects Tests
  - Full Scale Plant Data

- **Determination of Adequacy of Models**

- **Address Nodalization and Scaling Issues**

- **Determination of Model and Experimental Uncertainty**

- **Cross Correlation Against PIRT**
  - Qualified for all high ranked phenomena for the identified event

Reactor Input Parameters and State - CSAU Step 11

ID	*TABLE 3-3: Initial Conditions for System Thermal/Hydraulic Analyses of Stability Phenomena* INITIAL CONDITION	Core-wide Stability	Regional Stability	Highest Ranking	Critical Parameter	**1. Thermal Margin and/or PCT** Controlled by heat flux, flow, pressure, and inlet subcooling - Power oscillations - Flow oscillations **2. Power Oscillation Amplitude/Frequency -** controls stability margin/ growth rate of perturbations which for this application is assumed to be a limit cycle oscillation (i e , DR=1 0)	?	*←Can quantity be affected by core design, fuel type, or plant type?* Comments on dependency of phenomena to core design, fuel type, plant design
IC1	TOTAL CORE POWER	H	H	H	1,2	Power/Flow ratio is more important than individual quantities.	Yes	
IC2	TOTAL CORE FLOW	H	H	H	1,2	Power/Flow ratio is more important than individual quantities	Yes	
IC3	FEEDWATER TEMPERATURE	H	H	H	1,2	Controls core inlet sub cooling	Yes	
IC4	STEAM DOME PRESSURE	L	L	L		Not significant in operating range	Yes	
IC5	INITIAL DOWNCOMER WATER LEVEL	M	M	M	1,2	Not expected to be sensitive to initial water level in normal range	Yes	
IC6	CORE SIZE	M	H	H	1,2	Affects subcriticality of harmonic mode		
IC7	CORE LOADING PATTERN AND CORE EXPOSURE	H	H	H	1,2	Affects core nuclear parameters	Yes	
IC8	CORE AXIAL POWER DISTRIBUTION	H	H	H	1,2	Affects oscillation characteristics	Yes	
IC9	HOT CHANNEL AXIAL POWER DISTRIBUTION	H	H	H	1,2	Affects oscillation characteristics	Yes	
IC10	RADIAL POWER DISTRIBUTION	H	H	H	1,2	Affects bundle power, oscillation characteristics Includes control rod patterns	Yes	
IC11	LOCAL PEAKING IN HOT BUNDLE	H	H	H	1,2	Affects thermal margin/rod heatup in hot bundle	Yes	
IC12	CONTROL ROD PATTERN	H	H	H	1,2	Effects are through power distribution Affects axial and radial power distribution	Yes	
IC13	HOT BUNDLE EXPOSURE	M	M	M	1,2	Affects core nuclear parameters in limiting region	Yes	
IC14	PEAK PELLET EXPOSURE	L	L	L		Bundle exposure effects more important		
IC15	INITIAL MCPR	H	H	H	1,2	Governs margin to BT during oscillations	Yes	
IC16	XENON CONDITION	M	M	M	1,2	Affects core nuclear parameters	Yes	
IC17	SURFACE SCALE (CRUD AND OXIDE)	M	M	M	1,2	Affects fuel time constant.	Yes	

Combination of Uncertainties - CSAU Step 13

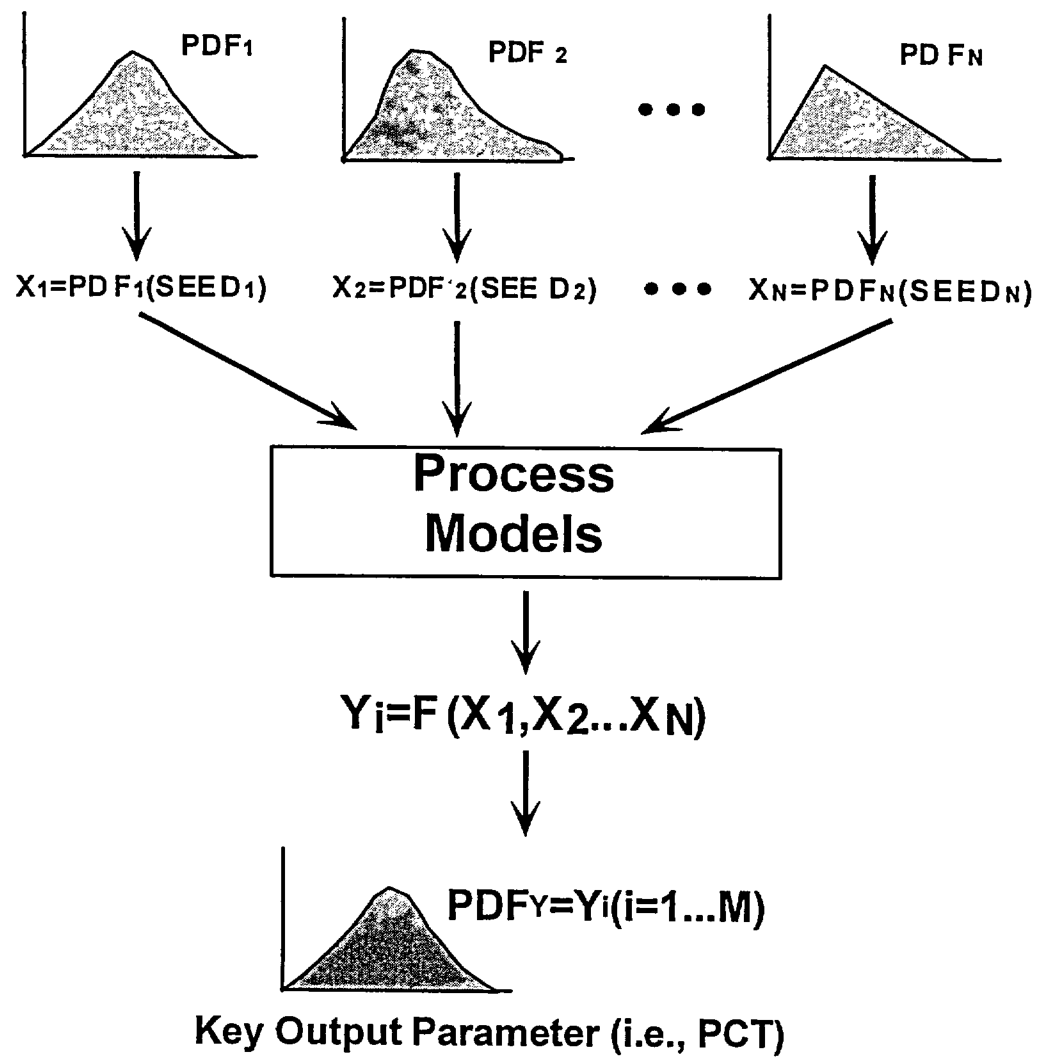
• Statistical treatment of Overall Calculational Uncertainty

<table>
<tr><th>Method</th><th>Description</th><th colspan="2">Comments</th></tr>
<tr><td>Order Statistics Method - Single Bounding Value
(GRS Method)</td><td>Monte Carlo Method using random perturbations of all important parameters.
Statistical upper bound determined from n'th limiting perturbation (for first order statistics).</td><td rowspan="2">The number of random trials is independent of the number of input parameters considered.
The method requires no assumption about the PDF of the output parameter.
It is not necessary to perform separate calculations to determine the sensitivity of the response to individual input parameters.
It is not necessary to make assumptions about the effect on the output of interactions of input parameters.</td><td>Results can be very conservative for small number of trials, and consistent results may require a larger set of trials.</td></tr>
<tr><td>Analysis of Variance
(OSUSL)</td><td>Monte Carlo Method using random perturbations of all important parameters.
Statistical upper bound determined from sample variance from all perturbations.</td><td>The normality of the PDF for the output variable must be demonstrated.</td></tr>
</table>

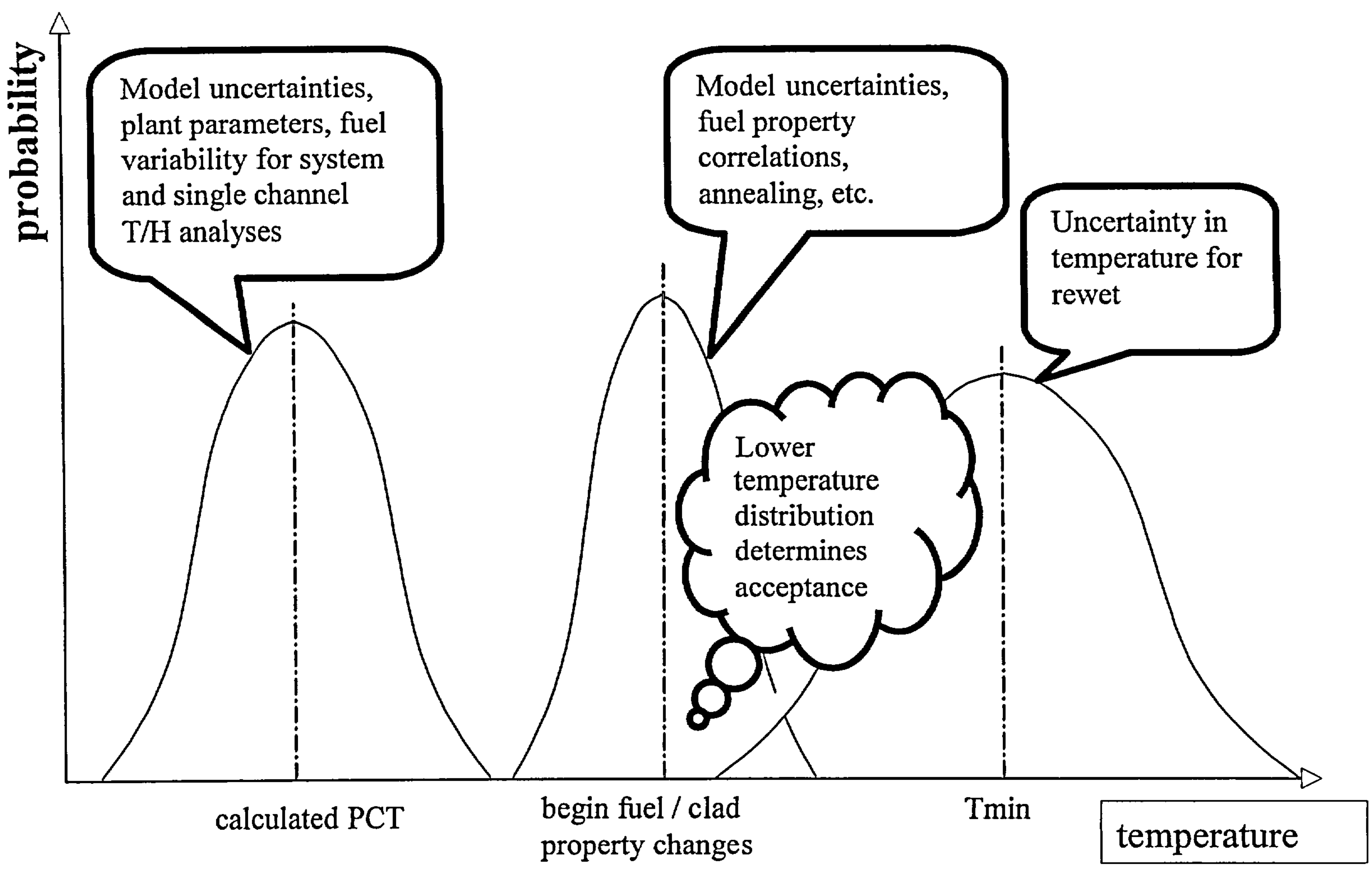
Approach: Analysis of Variance if Normal Distribution, otherwise Order Statistics for the Limiting Value.

Order Statistics or Analysis of Variance - CSAU Step 13

Figure 3-4



OSUSL Determination - CSAU Steps 13 and 14 *Figure 3-3*



Application Methodology Demonstration Calculations

- **Objective**
  - Demonstrate application methodology
  - Confirm that OSUSL bounds data

- **Plant Calculations**
  - Reactor instability at or just below the OPRM set point
    Oscillation continues for 30 minutes without mitigating actions
  - Fast growing reactor instability with scram at OPRM set point

Application Methodology for BWROG Stability Limit Analysis

Summary

- **Scope:**
  - Reactor Instability
  - BWR/2-6
  - Current operating domains and fuel types
- **Meets All Regulatory Requirements**
- **Demonstration of Model Capability and Applicability**
- **Rigorous and Sound Statistical Methodology**
  - Model Uncertainty
  - Initial Conditions and Plant Parameter Uncertainties
  - One Sided Upper Statistical Limit for Critical Safety Parameters
- **Application Methodology Demonstrated for all Instability Events**

Obtain SER for Application to Stability Limit Analysis

SAFDL Selection

Improved Stability Evaluation Basis

- **Approach**
  - Permit oscillations to take fuel into and out of boiling transition.
  - Avoid condition of sustained boiling transition (rewetting necessary)
  - Evaluate acceptance relative to applicable T-M SAFDLS including
  - any accumulative margin loss.

- **Evaluation considerations**
  - Cladding and channel stress and strain, including creep deformations
  - Cladding oxidation
  - Fuel center melting
  - Increased fission gas release and fuel rod internal pressure
  - Cladding fatigue

Event Scenario

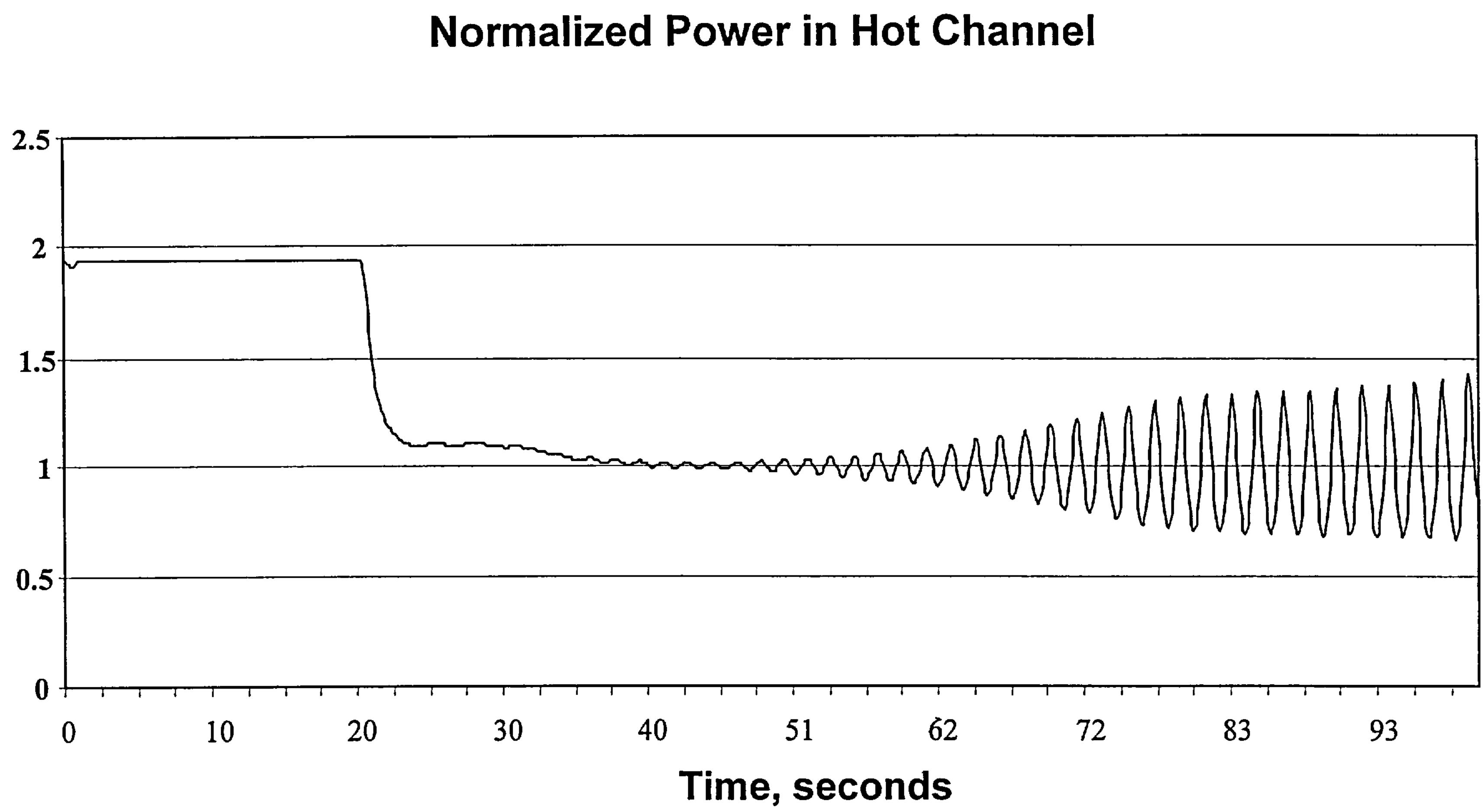
- Oscillating neutron flux (e.g. $\pm 50\%\frac{P-P}{A}$ with 2 sec period for 30 minutes)

- Two cases considered
  Low temperature - no boiling transition
  Elevated temperature - boiling transition (with rewetting)

- Low Temperature Considerations
  Power increase -> Fuel temperature increase -> increased fission gas release
  -> increased fuel rod internal pressure
  -> increased pellet-clad interaction
  -> increased cladding stress/strain
  Multiple power cycles -> multiple strain cycles -> cladding fatigue

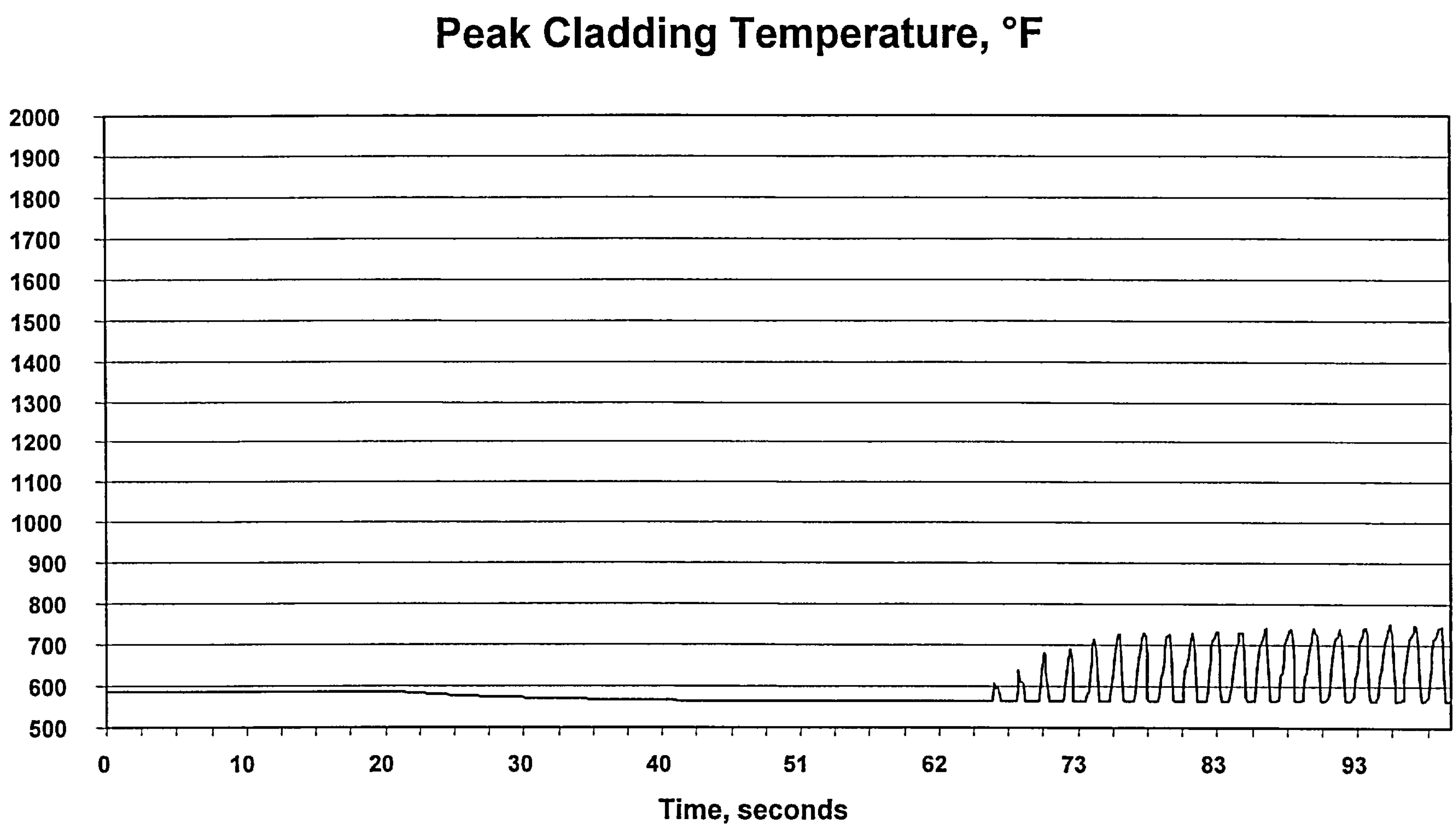
- High Temperature Considerations
  Cladding oxidation -> embrittlement
  Cladding creep collapse/ballooning
  Even higher fuel temperatures -> fuel melting
  -> increased fission gas release
  -> increased fuel rod internal pressure
  -> increased pellet-clad interaction
  -> increased cladding stress/strain

Perspective



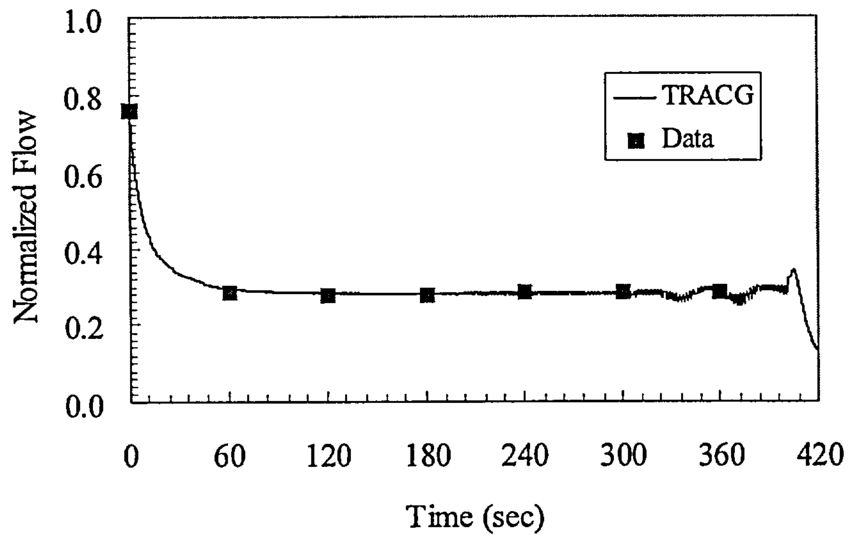
Power level during event remains below prior steady-state condition
Significant fuel temperature decrease

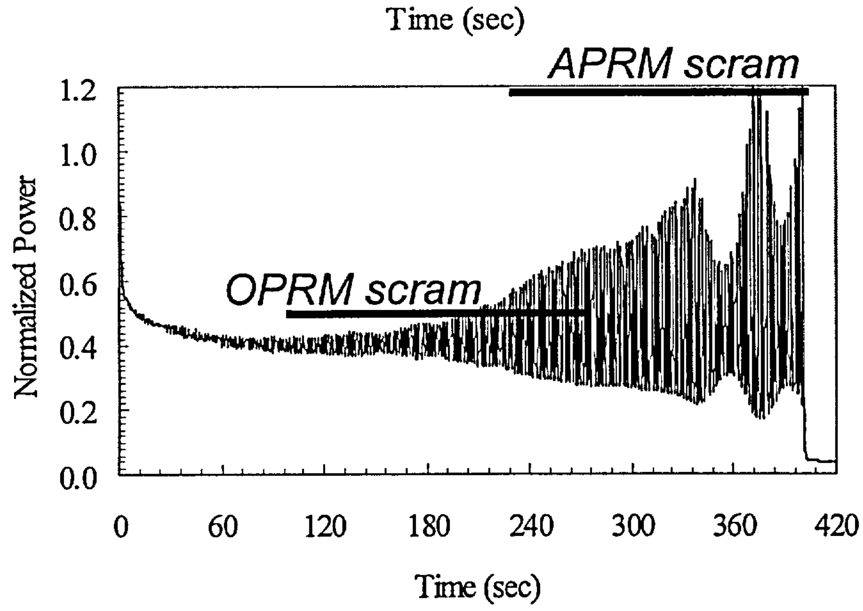
Perspective



Increase in cladding temperature will translate to increase in fuel temperature

LaSalle Instability Event (1988)

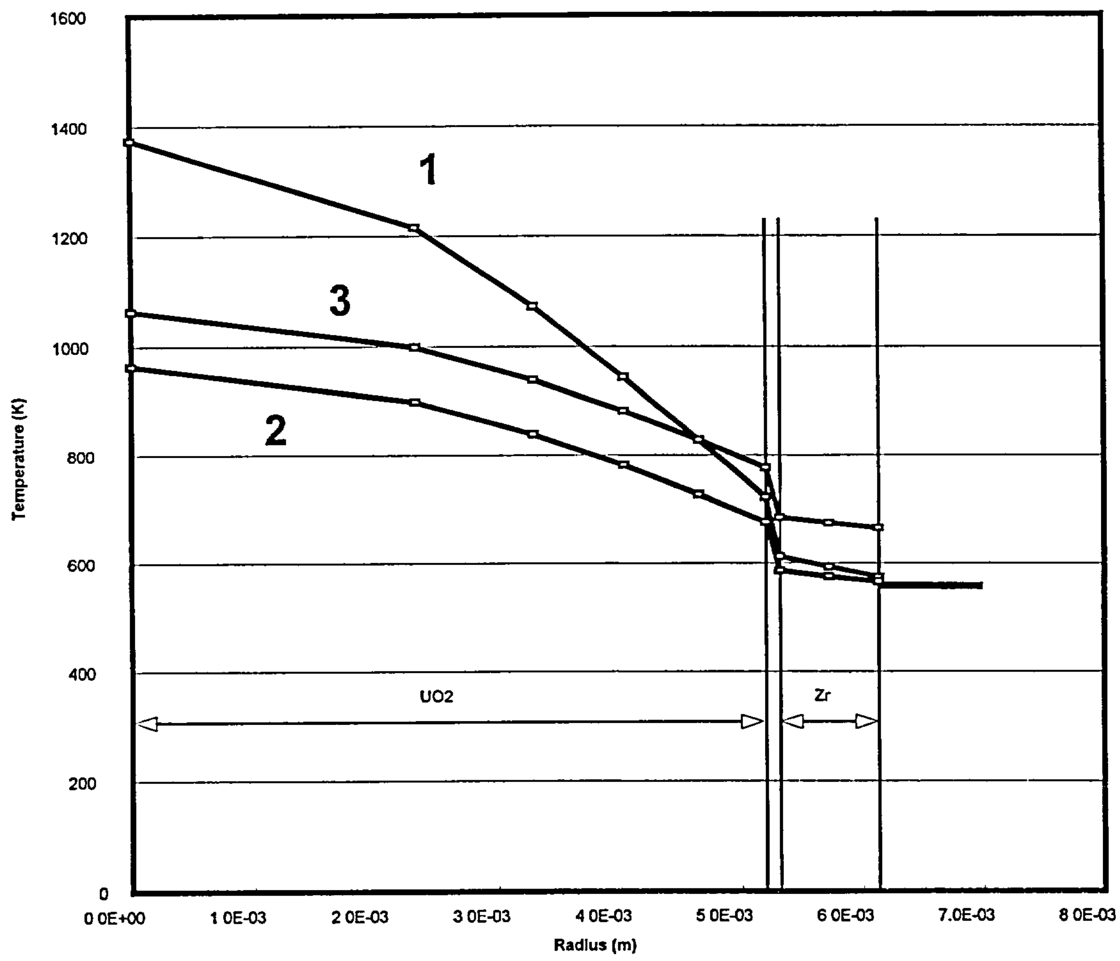




- Two reactor pump trip
  - Initial conditions:
    - 84% of rated power
    - 76% of rated flow
- Core wide instability
  - Terminated by APRM scram

OPRM system will result in a scram at a much lower oscillation magnitude.

Fuel Rod Radial Temperature Profiles
for LaSalle Instability Event (1988)



Two reactor pump trip

- Initial conditions:
  - •84% of rated power
  - •76% of rated flow
  - •Fuel temperatures for highest power bundle (Red - 1)
- Final condition after RPT
  - •40% of rated power increasing to 45% with feed water temperature reduction
  - •28% of rated flow
  - •Fuel temperatures for highest power bundle (Brown - 2)
- Preliminary analysis shows that if boiling transition were to occur, an average increase in the cladding temperature of 100K would have a negligible impact on the cladding properties. This would also increase the average fuel temperatures by 100K
  - •Fuel temperatures for highest power bundle (Green - 3)
- Fluid saturation temperature (Blue)

Average Temperatures Lower than Initial Operating Temperatures

Result

- No increase in fuel duty during event
  - No increase in cladding stress due to PCMI relative to initial steady-state condition

- Following considerations will be negligibly affected
  - Cladding stress and strain
  - Fuel centerline melting
  - Fission gas release and fuel rod internal pressure
  - Cladding fatigue

- Remaining considerations determined by peak cladding temperature obtained during event
  - Cladding creep deformation (collapse or ballooning)
  - Cladding oxidation

- For expected range of event peak cladding temperatures, cladding creep and oxidation expected to be negligible

- Remaining consideration is annealing of irradiation hardening

Annealing of Irradiation Hardening

- Exposure of cladding to fast neutrons causes damage to Zr crystal lattice
  - Increased cladding strength
  - Decreased cladding ductility
  - Occurs over a period of months then essentially saturates
- Significant cladding temperature increase anneals (heals) irradiation damage
  - Mechanical properties change from that assumed in standard analyses
  - Lower strength
  - Increased ductility
  - Decreased creep resistance
- Desire to minimize annealing of irradiation hardening to maintain applicability of current analyses
- With no significant annealing of irradiation hardening, event reduces to essentially no difference from normal reduction in power and return

Primary consideration is peak cladding temperature to avoid significant annealing