Westinghouse Non-Proprietary Class 3

Westinghouse Presentation
for
Fuel Performance Update Meeting
BWR / PWR Fuel Update
(Slide Presentation for April 11, 2006)

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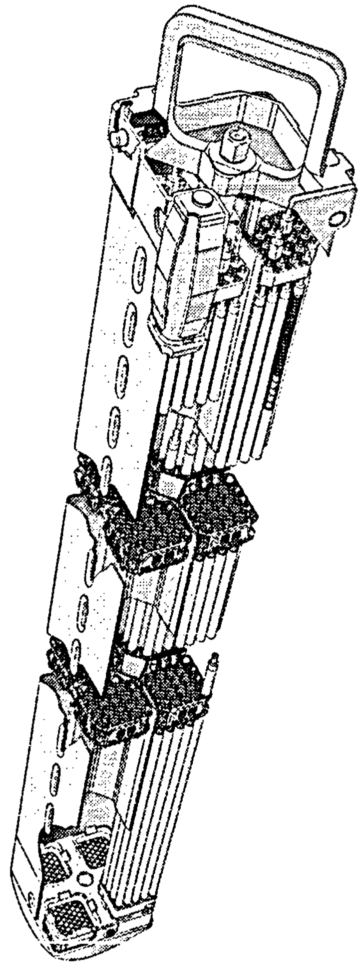
AGENDA
Westinghouse Fuel Performance Update
April 11, 2006
Westinghouse Office
Rockville, Maryland

Tuesday, Apr 11	**BWR Fuel Update**	
8:00 – 8:15 am	Introductions and welcome	[ ]^{a,c}
8:15 – 9:30 am	Fuel Performance Update	[ ]^{a,c}
9:30 – 10:00 am	Westinghouse BWR Containment Methodology using GOTHIC	[ ]^{a,c}
10:00 – 10:15 am	Break	
10:15 – 11:00 am	BISON Model and Application for ATWS with Boron Insertion	[ ]^{a,c}
11:00 – 11:50 am	Containment Back Pressure Application to Westinghouse U.S. BWR ECCS Evaluation Methodology	[ ]^{a,c}
11:50 – Noon	Wrap-up	[ ]^{a,c}
Noon – 1:00 pm	Lunch/Informal Discussion between NRC, Customers & Westinghouse	
	PWR Fuel Update	
1:00 – 1:15 pm	Introductions and welcome	[ ]^{a,c}
1:15 – 3:00 pm	Fuel Performance Update [Watts Bar Fuel Failure]^{a,c} Damaged Fuel Pellet	[ ]^{a,c}
3:00 - 3:15 pm	Break	
3:15 – 3:45 pm	CE16 NGF Fuel & DNB Correlation Update	[ ]^{a,c}
3:45 – 4:30 pm	SRWM Interpretation of Topical Report Test Results at Vogtle	[ ]^{a,c}
4:30 – 4:50 pm	ANC/BEACON Integration	[ ]^{a,c}
4:50 – 5:00 pm	Wrap-up	[ ]^{a,c}

Dress is Business Casual

BWR Fuel Performance Update



Fuel Performance Update Meeting
Rockville, Maryland
April 11, 2006

Outline of Presentation

❶ Statistics

- Deliveries
- Burnup
- Failures

❷ In-Reactor Performance

- Cladding
- Channel
- Pellet

BWR Fuel Deliveries

a, c

BWR Fuel Burnup Experience, 2005

a, c

Fuel Performance

a, c

[ ]

Primary Fuel Failures Westinghouse BWR Experience

a, c

Westinghouse 10x10 Fuel Primary Failures

a, c

Westinghouse 10x10 Fuel Primary Failures

[ ] a, c

Outline of Presentation

❶ Statistics

- Deliveries
- Burnup
- Failures

❷ In-Reactor Performance

- Cladding
- Channel
- Pellet

Cladding

- Improve high burnup performance
  - Development of LK2+ and LK3

- Verify high burnup performance
  - Pool-side and hot-cell examinations of leading fuel rods
    - Corrosion
    - Rod growth
    - Hydriding

a, c

[ ]

LK3 - Achieved Burnup

a, c

Two-Life Rods [ ]a, c

[ ]a, c

Two-Life Rods Power History

a, c

Cladding Corrosion
Mid-span Oxide Thickness by Cladding Type

a, c

Rod Growth By Cladding Type

a, c

BWR Fuel Performance Channel

a, c

Channel Corrosion

a, c

Hydrogen Pick-Up

a, c

Hydrogen Pick-Up Outer Zry-2 Channel at 45 MWd/kg U

a, c

Channel Bow & Irradiation Induced Growth

a, c

Channel Growth

a, c

Channel Bow in Asymmetric Lattice

a, c

Channel Bow in Symmetric Lattice

a, c

Fission Gas Release

a, c

Westinghouse BWR Containment Methodology Using GOTHIC

Fuel Performance Update Meeting
Rockville, Maryland
April 11, 2006

Outline

- Objective
- Introduction
- Overview
- Events to be Modeled
- GOTHIC
- Sample Results
- Submittal Plan

Objective

- The objective is to present a high-level overview of Westinghouse plans about the BWR Containment methodology

Introduction

- Currently, containment calculations are done in plant-specific manner and submitted individually (no referenced generic methodology)

- Westinghouse is developing BWR containment methodology for generic approval

Overview

- The methodology will use the GOTHIC code
- Base models for Mark-I, Mark-II, and Mark-III containments are developed
- Designs will be in separate appendices to a base submittal
- The submittal will be for approval of the methodology for all the containment events

Events to be modeled

- The events for Mark I containment includes:
  1. An instantaneous guillotine rupture of a recirculation suction line
     - 1.1. Short-Term Containment Pressure and Temperature Response
     - 1.2. Long-Term Containment Pressure and Temperature Response
       - 1.2.1. Offsite Power Available
       - 1.2.2. Loss of Offsite Power
       - 1.2.3. Various equipment/ECCS configurations
  2. An instantaneous guillotine rupture of a main steam line
     - 2.1. Short-Term Drywell and Containment Pressure and Temperature Response
     - 2.2. Long-Term Containment Pressure and Temperature Response
       - 2.2.1. Offsite Power Available
       - 2.2.2. Loss of Offsite Power
       - 2.2.3. Various equipment/ECCS configurations

Events to be modeled (cont.)

3. An intermediate size liquid line rupture
4. A small size steam line rupture
5. Steam Bypass of the Suppression Pool ($A/\sqrt{K}$)
6. Vacuum Relief Analyses
    6.1. Between drywell and wetwell/suppression chamber and between wetwell/suppression chamber and reactor building
7. Station Blackout
8. Anticipated Transient Without Scram (ATWS)
9. Suppression Pool Hydrodynamic Loads

GOTHIC

from NAI 8907-06 Rev 12

- GOTHIC is a state-of-the-art program that solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow

- The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime

- Conservation equations are solved for up to three primary fields:
  - Steam/gas mixture
  - Continuous liquid
  - Liquid droplet

GOTHIC (cont.)

- GOTHIC includes an extensive set of models for operating equipment:
  - Pumps and fans
  - Valves and doors
  - Heat exchangers and fan coolers
  - Vacuum breakers
  - Spray nozzles
  - Coolers and heaters
  - Volumetric fans
  - Hydrogen recombiners
  - Ignitors
  - Pressure relief valves
  - Etc...

GOTHIC (cont.)

- Examples of the types of problems that can be modeled include:
  - Pressurization and heating due to blowdown
  - Suppression pools
  - Equipment qualification
  - Hydrogen distribution/combustion
  - Jet impingement loads
  - Jet induced mixing
  - Ice condensers
  - Etc…

Sample Results

a, c

Sample Results

a, c

Sample Results

a, c

Sample Results

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Sample Results

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Submittal Plan

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BISON Model and Application for ATWS with Boron Insertion

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BISON: Topical to NRC 1989

- Topical Report RPA 90-90-P-A, Rev. 0, "BISON – A One Dimensional Dynamic Analysis Code for Boling Water Reactors"
  - The original topical describing the BISON code
    - Thermal Hydraulics
    - Fuel Model
    - Kinetics
    - Basic Code validation

BISON: Supplement to Topical to NRC 1996

- Topical Report CENPD-292-P-A, "BISON – A One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification," July 1996
  - New steam line model
  - Double Drive loops for Single Loop Operation
  - New void modeling (AA78/EPRI)
  - Validation of new models

BISON: Supplement to Topical to NRC 2006

- "Supplement to RPA 90-90-PA. BISON Model and Application for ATWS with Boron Insertion," Pre-submittal Meeting in May, 2006 followed by Topical Submittal in June, 2006
  - Increased Pressure range for AA78 void correlation
  - Boron reactivity model
  - Boron model validation (versus POLCA)
  - ATWS Application

Pressure Range for AA78 slip correlation

- Original AA78 correlation validated to 9.0 MPa
- Extended to 10.0 MPa in original Topical (RPA90-90)
  - Range extension by comparisons with EPRI slip
- Need for increased range for ATWS
  - Range increase to 12.0 MPa suggested
  - Extended comparisons equal to RPA 90-90
  - RMS and bias for range extension added

Pressure Range for AA78 slip correlation

a, c

Pressure Range for AA78 slip correlation

a, c

Pressure Range for AA78 slip correlation

Extrapolation accuracy:

a, c

[ ]

Conclusions:

- Extrapolation of AA78 slip correlation up to 12 MPa is shown to be similar with the EPRI correlation validated up to15 MPa
- Extrapolation can be made with essentially the same RMA and bias as for the previously validated pressure range

Boron Model

- Three different models included
  - Boron injection model – Boron Mass into the system
  - Boron concentration model – ppm in the system
  - Boron Reactivity model – Δk_{eff} in the core

- Implementation does NOT affect
  - T/H in Topical and Supplement
  - Kinetics model from Topical if Boron is not simulated

Example of ATWS scenario

- Pressure regulator failure – increasing demand
  - Fast depressurization
  - MSIV closure on low RPV pressure
  - Fast pressurization
  - ATWS signal on high pressure
  - Manual feedwater trip
  - Manual water level control at about 0 above top of active fuel
  - Recirculation stops due to lack of downcomer pressure head
  - Internal Core/Core bypass circulation maintains core cooling
  - Start of Boron injection in the lower plenum

Example of ATWS scenario – Dome Pressure

a, c

Example of ATWS scenario – Water Level

a, c

Example of ATWS scenario – Core Flow

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Boron Injection Model

- User specified:
  - Initiation time and mass flow rate
  - Total available Boron solution mass
  - Boron Concentration Fraction of the Boron solution mass
  - Conservativeness Factor
  - 3D correction factor

Boron Concentration Model

a, c

Boron Reactivity Model

- All cross-sections are calculated as described in the BISON topical report
- An additional set of cross-sections with Boron implemented is used for all fuel types.
- From this set the Boron worth is evaluated as a differential cross-section

[ ] a, c

Boron Model Validation

a, c

Boron Model Validation

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Example of ATWS scenario – Boron insertion

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Example of ATWS scenario – APRM

a, c

Westinghouse ATWS application

a, c

Containment Backpressure Application to Westinghouse U.S. BWR ECCS Evaluation Methodology

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April 11, 2006

Outline

- Objective
- Current Methodology
- Compliance with Appendix K
- Containment B.C. in LOCA Analysis
- Supporting Containment Analysis Method
- Biasing for Low Pressure
- Anticipated Impact
- Licensing Plan

Objective

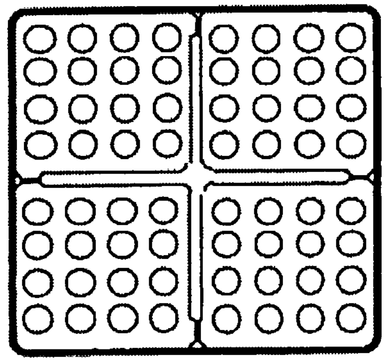
- The objective is to inform the NRC about Westinghouse plans for crediting containment backpressure in BWR ECCS evaluation calculations

Current Methodology

- USA1

  RPB-90-93-P-A and RPB-90-94-P-A

  SVEA-64

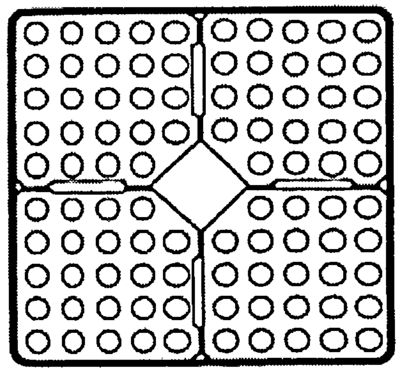


- USA2

  CENPD-283-P-A and CENPD-293-P-A

  SVEA-96 / SVEA-96+

  Applied to Columbia and Hope Creek



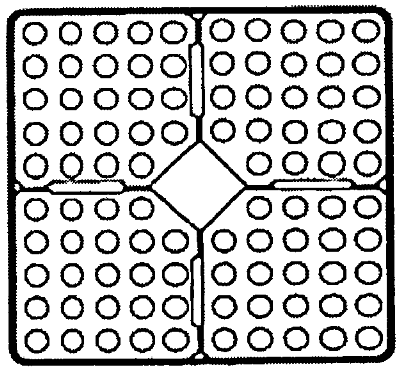
Current Methodology (cont.)

- USA4

  WCAP-15682-P-A

  Relaxation of rod-to-rod touching model

  also applied to Hope Creek



- USA5

  WCAP-16078-P-A

  Optima-2 with partial rods

  applied to Quad Cities/Dresden

Compliance with Appendix K

- 1.02 times the licensed power
- ANS '71 decay heat +20%
- Metal-water reaction using Baker-Just equation
- Moody break flow
- Consideration of most-limiting single failure
- Conservative containment pressure
- Zero heat transfer from uncovery until rated spray
- Prescribed spray and reflood HTCs

Containment B.C. in LOCA

- Current methodology conservatively assumes atmospheric pressure boundary condition in the containment:

I.D.2. Containment Pressure – GOBLIN analyses will conservatively assume atmospheric pressure in the containment volume throughout the LOCA transient. This assumption adequately addresses the requirements for this feature of Appendix K.

Containment B.C. in LOCA (cont.)

- 10 CFR 50, Appendix K states:

  2. *Containment Pressure.*

     The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

Supporting Containment Analysis Method

- Conservative containment pressure for LOCA will be calculated using GOTHIC model

- Both the LOCA blowdown M&E calculation and the containment model will be biased to give low containment pressure

Biasing for Low Pressure

- SRP Section 6.2.1.5 provides guidance for minimum containment pressure analysis for PWR. It will be followed as appropriate. The differences for BWR will be adequately addressed.
  - Initial containment internal conditions
  - containment volume
  - active heat sinks (spray, spilled ECCS)
  - pressure suppression
  - passive heat sinks

Biasing for Low Pressure (cont.)

- Conservative LOCA M&E release will be used

a, c

Anticipated Impact

- Comparison using containment pressure from NPSH

a, c

Anticipated Impact (cont.)

a, c

Anticipated Impact (cont.)

a, c

Anticipated Impact (cont.)

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Anticipated Impact (cont.)

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Anticipated Impact (cont.)

a, c

Licensing Plan

a, c

PWR Fuel Performance Update

Fuel Performance Update Meeting

Rockville, MD

April 2006

Outline

- Fuel Reliability Overview
- 17 OFA Root Cause Investigation Update & Recent PIE Results
- Future PIE & Hot Cell Program
- RCCA Update
- [ ] a, c Fractured Rod
- Damaged Axial Blanket Annular Pellets
- [ ] a, c Update
- LTA Program Status

Westinghouse PWR Fuel Reliability
(50 to 60 Plants per Year, Westinghouse and CE-NSSS Fuel)

a, b, c

Leakage Mechanisms in Westinghouse PWR Fuel: 2005

a, b, c

PWR Fuel Performance Trend

a, b, c

Leaking Rods by Major Product Family for 2004 - 2005

a, b, c

Grid-Rod Fretting Solutions Being Implemented

a, b, c

Current Status of RFA/RFA2

a, c

RFA/RFA2 Leakers

a, c

CE Improved Designs Implementation

a, c

Summary

a, c

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17x17 OFA Leaking Fuel Since 2002

a, b, c

Recent 17 OFA PIE Results – [ ]$^{a, c}$

[ ]$^{a, c}$

Recent 17 OFA PIE Results - [ ] a, c

a, b, c

[ ]

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Planned Inspections for 17x17 OFA

a, c

Leaking Rod Hot Cell Program

a, c

Hot Cell NDE Status

a, c

Fuel Rod R36S/05

a, c

Fuel Rod M16S/05

a, c

Hot Cell NDE Status (con't)

a, c

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[ ] a, c - Separated RCCA Rodlet

[ ] a, c

[ ]$^{a, c}$ - Separated RCCA Rodlet

[ ]$^{a, b, c}$

[ ] a, c - Separated RCCA Rodlet

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Outline

- Fuel Reliability Overview
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[ ] a, c Fractured Rod

[ ] a, c

[ ] a, c Fractured Rod

[ ] a, c

[ ] a, c Fractured Rod

[ ] a, c

[ ]$^{a, c}$ Fractured Rod

[ ]$^{a, b, c}$

[ ] a, c Fractured Rod

[ ] a, b, c

[ ]$^{a, c}$ Fractured Rod

[ ]$^{a, c}$

Outline

- Fuel Reliability Outline
- 17 OFA Root Cause Investigation Update & Recent PIE Results
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- [ ] ^{a,c} Update
- LTA Program Status

Issue

a, c

Potential Pellet Damage Mechanisms

a, c

Extent of Condition

a, c

Remedial Actions

a, c

Interim Compensatory Measures

a, c

Corrective Actions to Prevent Recurrence

a, c

Engineering Evaluation

a, c

Engineering Evaluation

a, c

Engineering Evaluation - Safety

- The issue was entered into the Westinghouse 10 CFR Part 21 process which concluded that :
  - The damaged pellet had no impact on safety analyses
  - The fuel continues to function as it was designed
  - The damaged pellet did not represent a substantial safety hazard and therefore 10 CFR Part 21 notification was not required
  - A NSAL will be issued

Outline

- Fuel Reliability Overview
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[ ] a, c

[ ] a, c

[ ] a, c

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[ ]a, c

[ ]a, b, c

Outline

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Status of Optimized ZIRLO™ LTA Programs

a, c

[redacted]

AXIOM^{TM} LTAs

a, c

Overall Summary

- Fuel performance has improved in many areas
- Programs and action plans are in place and being implemented
- Fuel designs susceptible to grid to rod fretting being replaced with improved products that are performing well
- Most pressing issue at this time is resolution of 17 OFA leakers
  - Taking proactive measures to address potential leakage mechanisms
  - Hot cell exam in progress

CE 16x16 Next Generation Fuel and DNB Correlation Update

Fuel Performance Update Meeting
Rockville, Maryland
April 11, 2006

Agenda

- CE 16x16 NGF Design
- General Program Status & Schedule
- Licensing Status
  - WCAP 16500-P "CE 16x16 Next Generation Fuel Core Report"
  - WCAP 16523-P "Westinghouse Correlations WSSV and WSSVT for Predicting CHF in Rod Bundles with Side Supported Mixing Vanes"
- Westinghouse DNB Correlation Development Status
- Summary

CE 16x16 NGF Assembly

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General Program Status

a, c

Overall Program Schedule of NGF

a, c

Licensing Status

- WCAP 16500-P CE 16x16 Next Generation Fuel Core Report
  - Submitted February 28, 2006

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Licensing Status (cont.)

- WCAP 16523-P "Westinghouse Correlations WSSV and WSSVT for Predicting CHF in Rod Bundles with Side Supported Mixing Vanes"
  - Submitted March 20, 2006

[ ]a, c

DNB Correlation Development Status

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Summary

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Low Power Physics Testing
Subcritical Rod Worth Measurement (SRWM)

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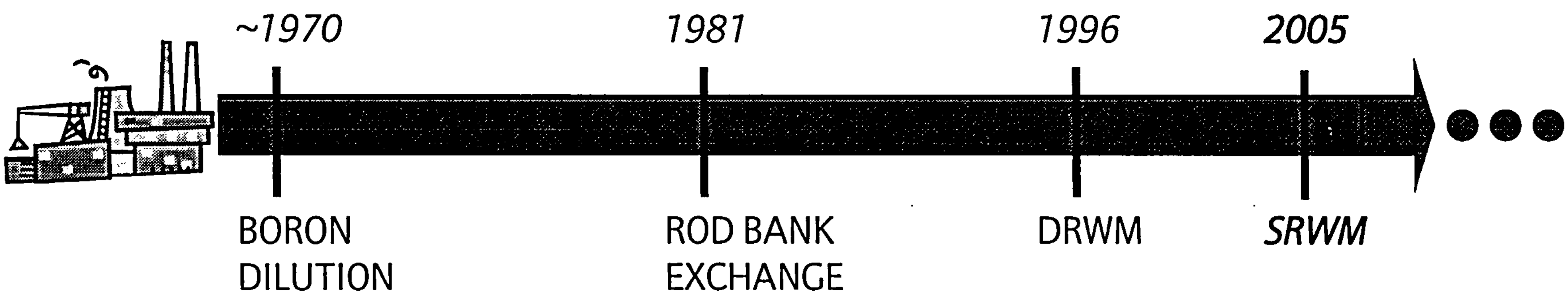
Basis for Low Power Physics Testing (LPPT)

- To verify that the reactor core will operate as designed prior to achieving full power operation following a core reload or other significant plant alterations
- Required by ANSI 19.6.1 and some utility technical specifications
- Control rod worth testing is one means of achieving these goals
- What kind of tests are required?
  - RCCA/CEA Drop Time (Hot Rod Drop)
  - Critical Boron Concentration
  - Control Rod Worth Measurement
  - Isothermal Temperature Coefficient Determination
  - Others

Evolution of Control Rod Worth Measurements

- SRWM is the new state-of-the art in Rod Worth Measurement



SRWM Theory

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SRWM Theory (cont.)

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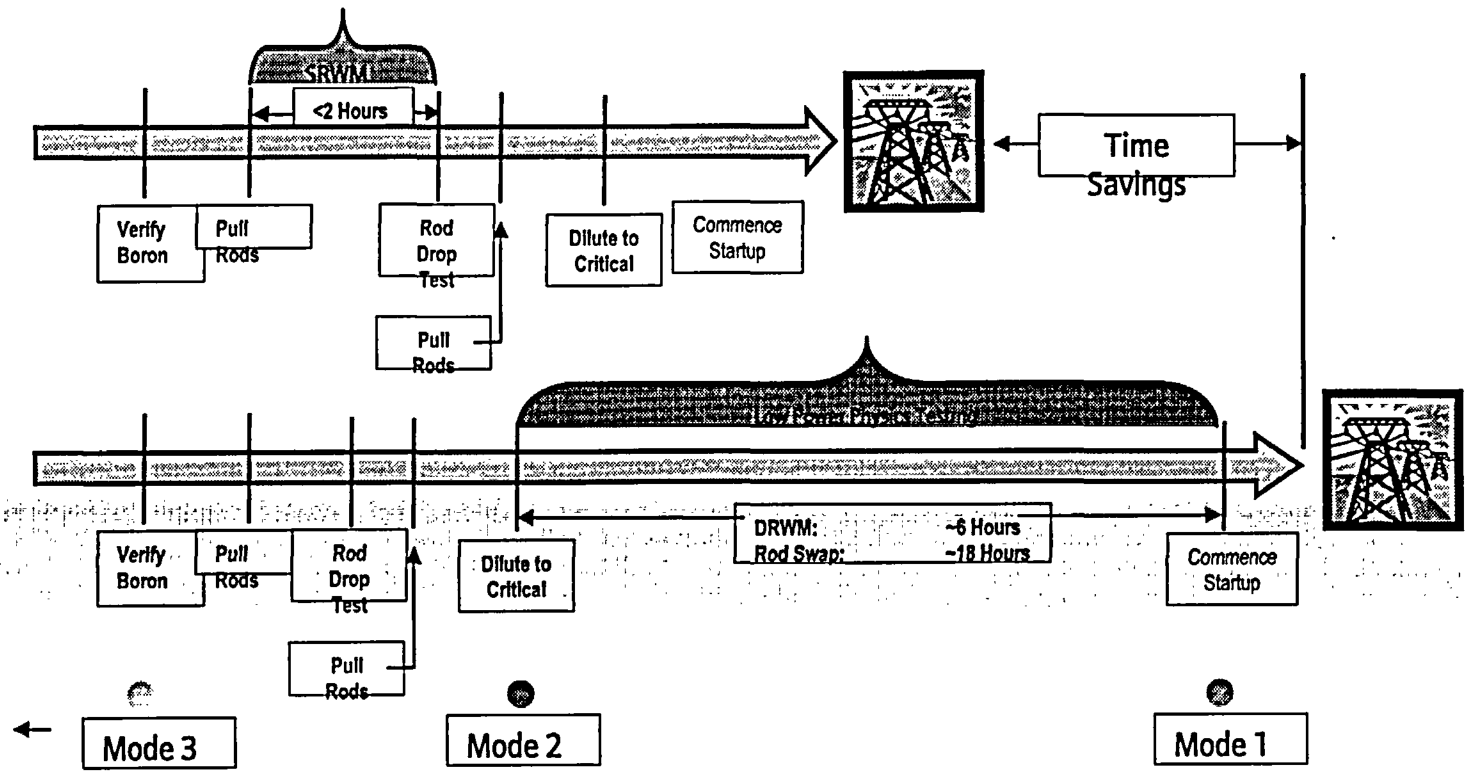
SRWM Path to Startup is Reduced



SRWM Licensing Status

- Licensing submittals and NRC review
  - WCAP-16260-P was submitted to NRC on April 30, 2004
  - The final SER was received on August 11, 2005
- Licensing interactions identified to clarify compliance with:
  - ANSI/ANS-19.6.1 Standards
  - B_4C reactivity checks - (WCAP-8846-A, approved September 1977)
  - EOL MTC Measurements - (WCAP-13749-P-A, approved March 1997)
- Westinghouse has drafted letter for NRC information/acknowledgement
  - LTR-NRC-06-15 submittal on March 30, 2006
  - Conclusion is EOL MTC and B_4C requirements are met if using SRWM

SRWM Licensing Status

- ANSI/ANS – 19.6.1 Standard
  - Reactivity balance - ARO boron endpoint
  - Capability for shutdown margin - Sum of all bank worths from ARI to ARO
  - Power distribution - Individual rod bank worth via the MD/RMS values
  - Reactivity Control - ITC

- B_4C control rod material - MD/RMS comparison same as rod swap

- EOL MTC Measurement - ITC
  - Sub-critical ITC only differs from critical ITC by 100 to 150 ppm
  - Adjustment based on well known boron worth at these conditions

- EOL MTC Measurement - Individual bank worths needed
  - Same as above - via the MD/RMS values obtained during testing

SRWM Implementation Status

a, c

SRWM Implementation Status (cont.)

- SRWM will be implemented at the following units:

[ ][a, c]

- An SRWM demonstration test is planned for [ ][a, c]
  - DRWM will be performed as the official LPPT Rod Worth test
- An additional 10 units are considering implementing SRWM in the fall of 2006

SRWM Improvements in Human Performance

- Plant employees experience less schedule pressure and fewer human performance challenges
  - Eliminates the need to check nuclear instruments just before testing
  - Equipment hookup was very simple and did not require any instruments to be removed from service
  - Startup from the refueling outage can be performed entirely using existing plant operating procedures, eliminating the need for special, additional test procedures

Update on APA Development Activities

Fuel Performance Update Meeting
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NEXUS Project - Description

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NEXUS Project - Description

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ANC and Related Technology Development

a, c

NEXUS/ANC 9 Project – Status and Actions

a, c

ANC9/SPNOVA Merge Project– Status and Actions

a, c

ANC9/SPNOVA Merge Project– Status and Actions

a, c

ANC 9 / SPNOVA Merge Project – Status and Actions

a, c