2010 Fuel Performance Meeting

Ronnie L. Gardner
Manager, Corporate Regulatory Affairs

Rockville, Md.
April 22 and 23, 2010

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Agenda – April 22, 2010

- *8:30 Welcome and Introductions – Ronnie L. Gardner*
- *8:45 Description of AREVA NP Inc. – Ronnie L. Gardner*
- *9:00 Current & Future Topical Reports – Ronnie L. Gardner*
- *9:30 PWR Fuel Designs – Patrick F. Williams*
- *10:00 Break*
- *10:15 PWR Core Engineering Methods – Thomas L. Lotz*
- *11:30 Lunch*
- *12:30 BWR Fuel Designs – Norman L. Garner*
- *1:30 BWR Core Engineering Methods – Douglas W. Pruitt*
- *2:30 Break*
- *2:45 PWR Operating Experience – John H. Strumpell*
- *3:45 BWR Operating Experience – Norman L. Garner*
- *4:15 Spent Fuel Plan – Robert Einziger (NRC)*
- *4:45 Wrap Up – Ronnie L. Gardner*
- *5:00 Adjourn*

Agenda – April 23, 2010

8:30 PWR Fuel Exams – John H. Strumpell

9:30 BWR Fuel Exams – Norman L. Garner

10:30 Break

10:45 M5 Guide Tube Growth – Brian V. Haibach

11:15 PWR Fuel Assembly Distortion Update – Brian V. Haibach

11:45 Lunch

12:45 PWR Fuel Rod Bow – Brian V. Haibach

1:15 PWR Cruciform Holddown Spring Performance Update – Brian V. Haibach

2:15 Break

2:30 Pellet Thermal Conductivity – John H. Strumpell

3:30 Crud Risk Management Tools – Christopher J. Lewis

4:30 Wrap Up – Ronnie L. Gardner

4:45 Adjourn

Objectives

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Objectives

Outline of discussion

- ◇ *Fuel designs*
- ◇ *Core engineering methods*
- ◇ *Operating experience*
- ◇ *Fuel exams*
- ◇ *Observations and solutions*

Objectives

- ◇ *Understanding AREVA's fuel designs and methods*
- ◇ *Exchanging ideas and expectations on fuel issues*
- ◇ *Open communication; ask questions*

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Description of AREVA NP Inc.

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AREVA NP

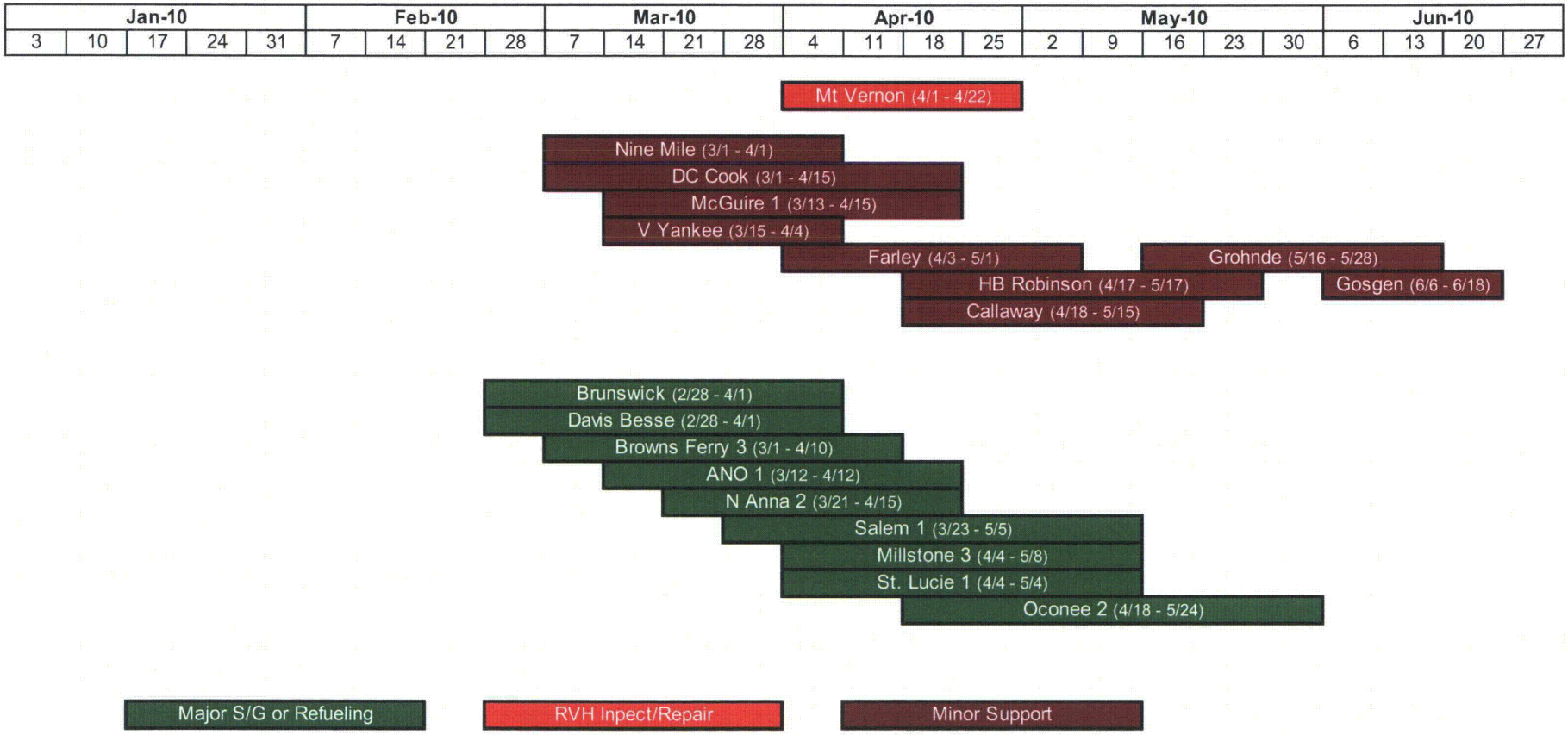
AREVA NP is the largest nuclear solutions employer in the U. S.

- ◇ 3,500 employees

AREVA NP is a market leader in nuclear outage services and component replacement

- ◇ Performed 12 of 19 steam generator replacements in the U.S. since 2003
- ◇ Currently supplying fuel to 30 percent of U.S. nuclear plants
- ◇ Completed reactor coolant pump and motor refurbishment work at 40 percent of U.S. plants
- ◇ Provided license renewal engineering for more than 50 percent of U.S. nuclear plants extending the life of existing reactors
- ◇ Performed 100 percent of the lower reactor head penetration repairs and approximately 90 percent of reactor vessel upper head nozzle repairs since 2000

Spring 2010 Outages



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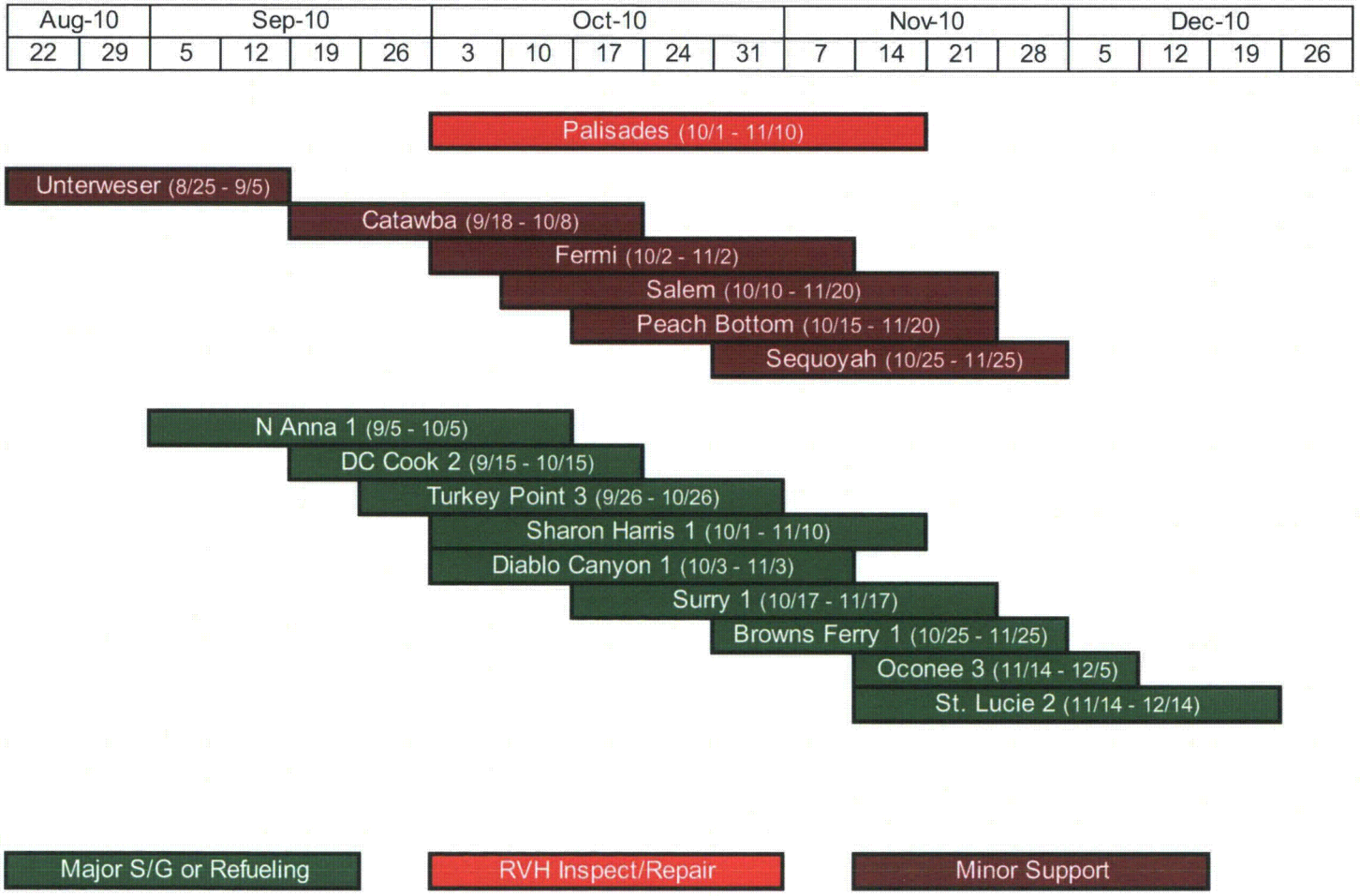
Fall 2010 Outages



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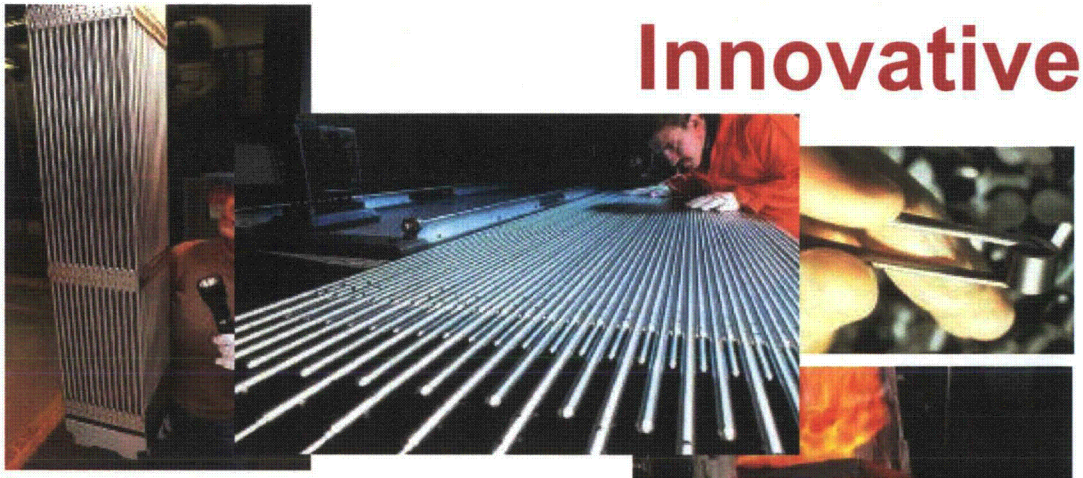
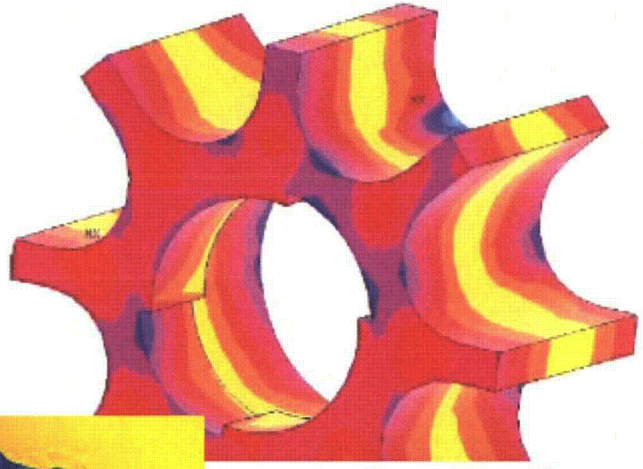
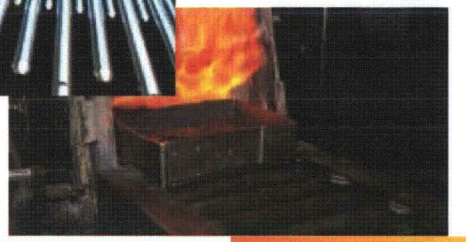
Global Integration & Innovative Solutions

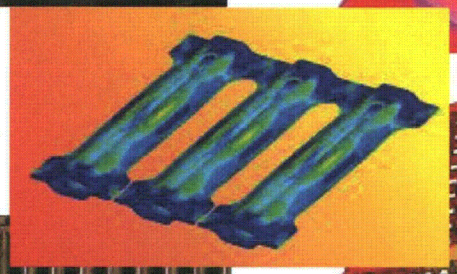


>*Engineering*

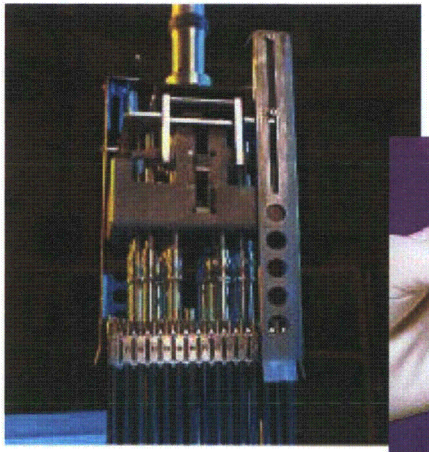


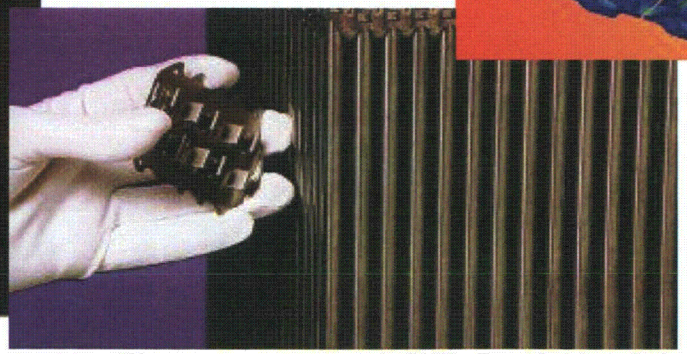
>*Manufacturing*

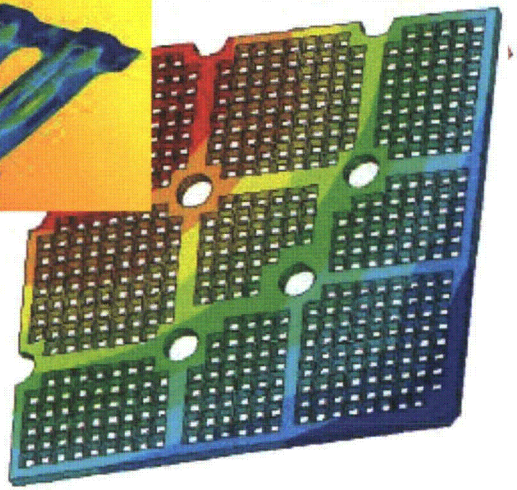




>*Field Services*







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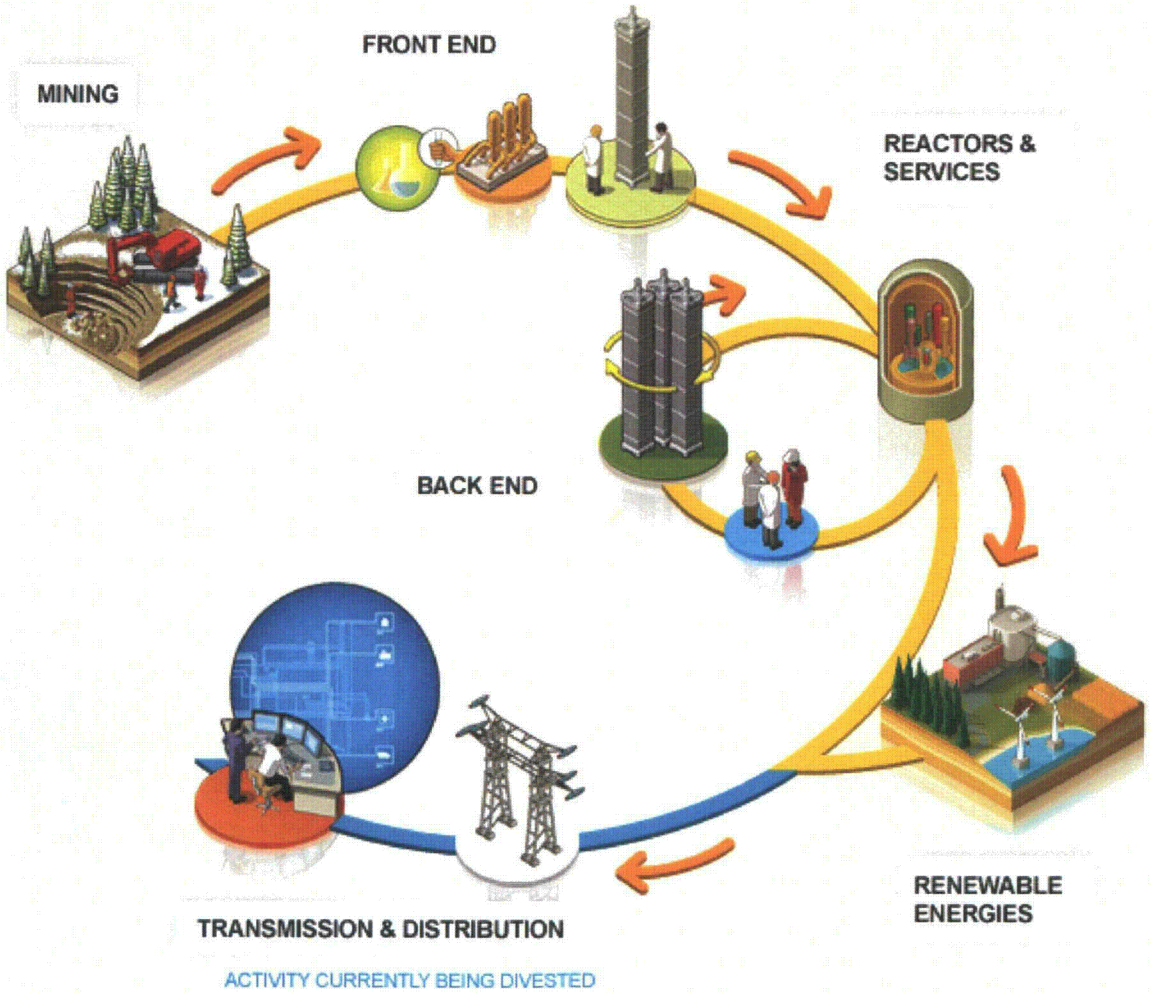
Creation of Business Group



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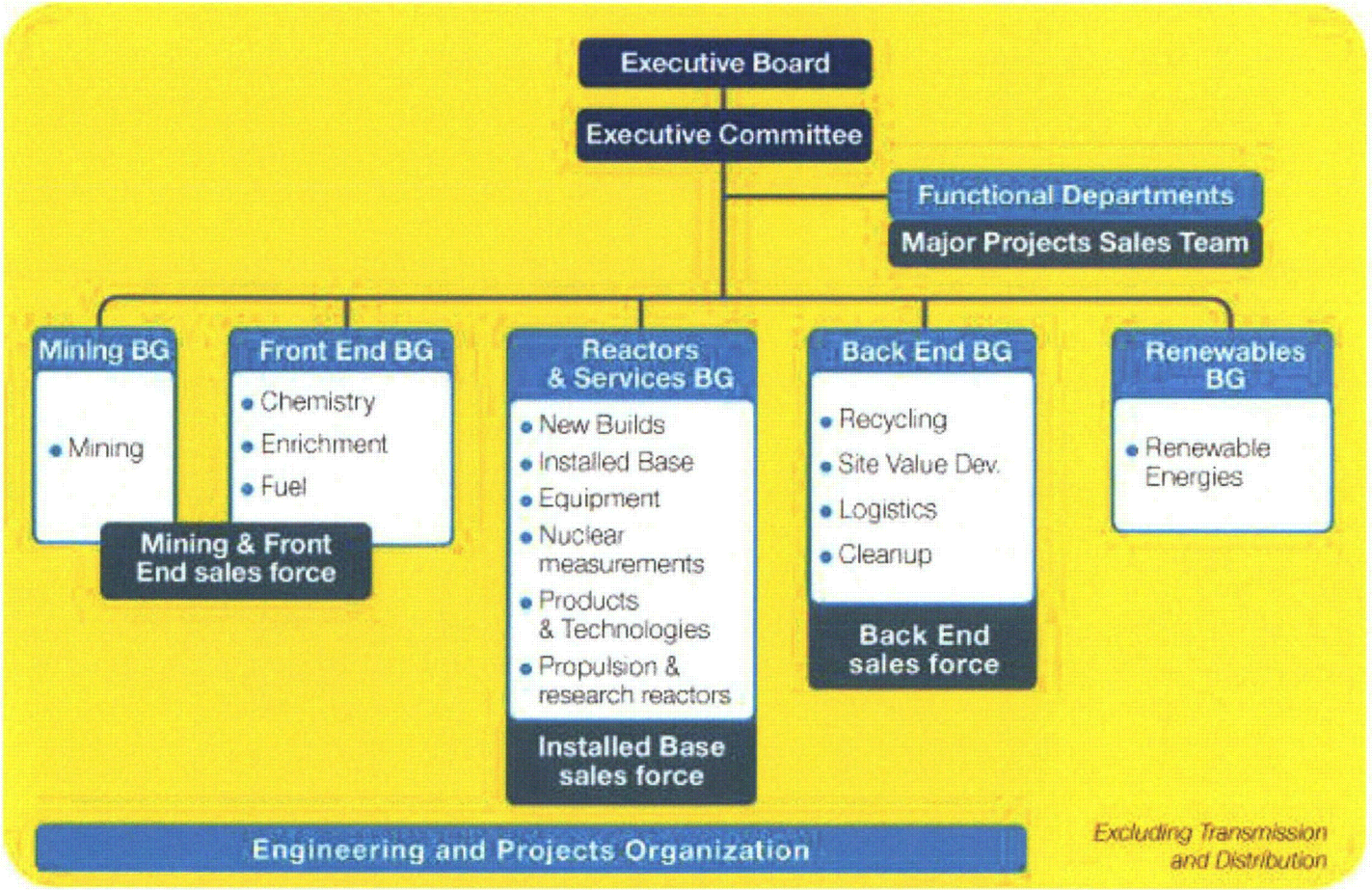
Creation of Business Group



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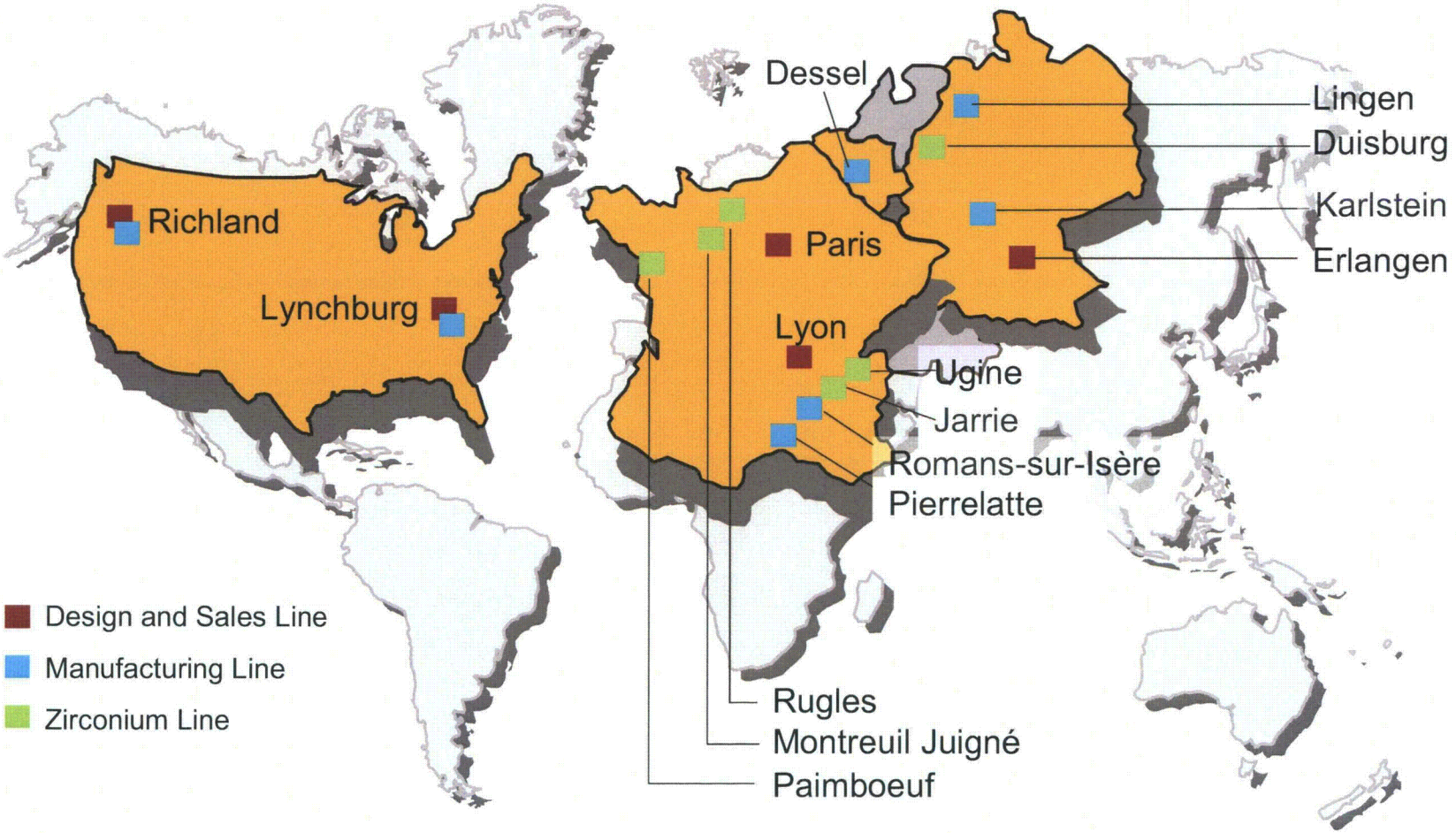
Global Nuclear Fuel and Fuel Component Facilities

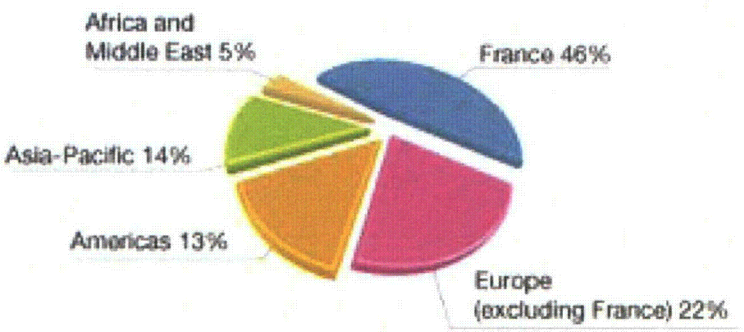


Global Workforce

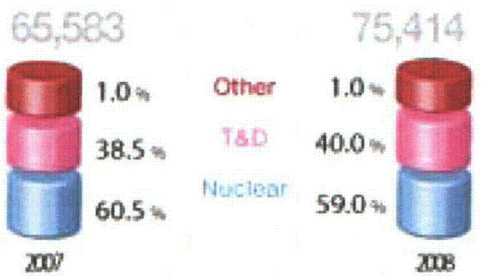
Workforce

75,414 employees

Workforce by region



Workforce



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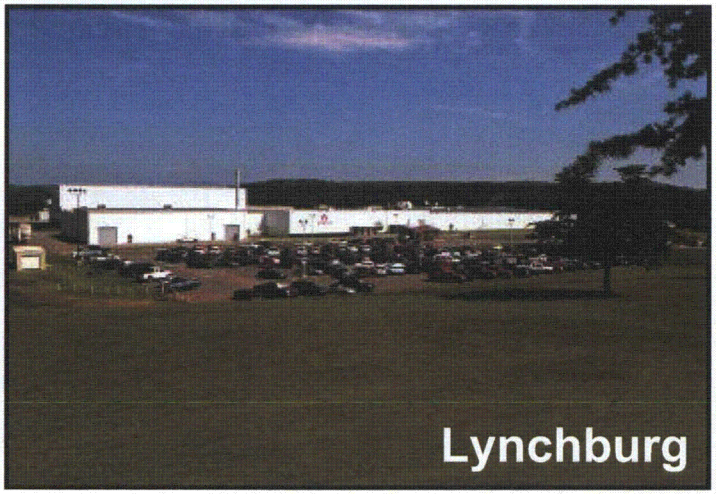
Nuclear Fuel



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AREVA US Fuel Customers

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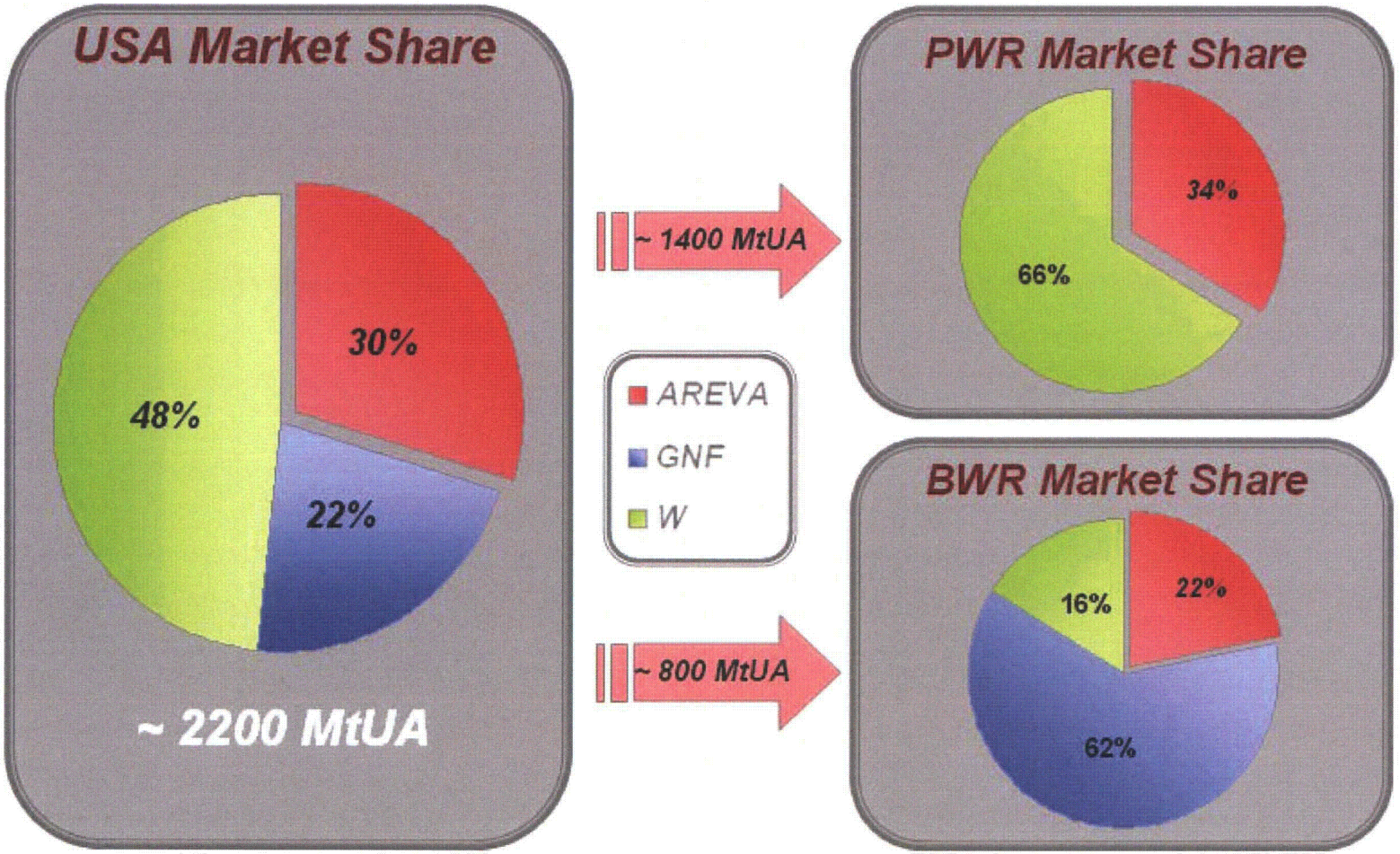
US Fuel Market Share



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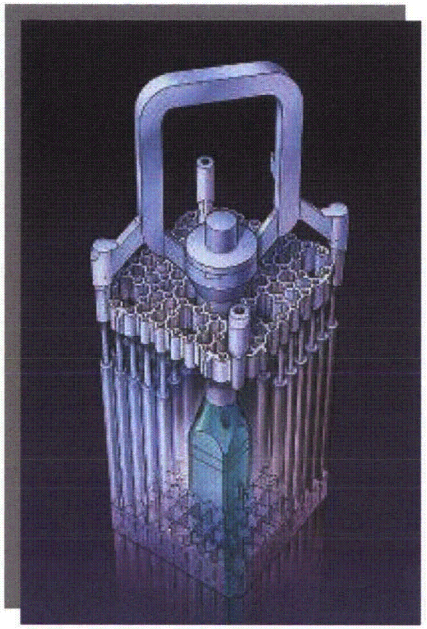
Fuel Products



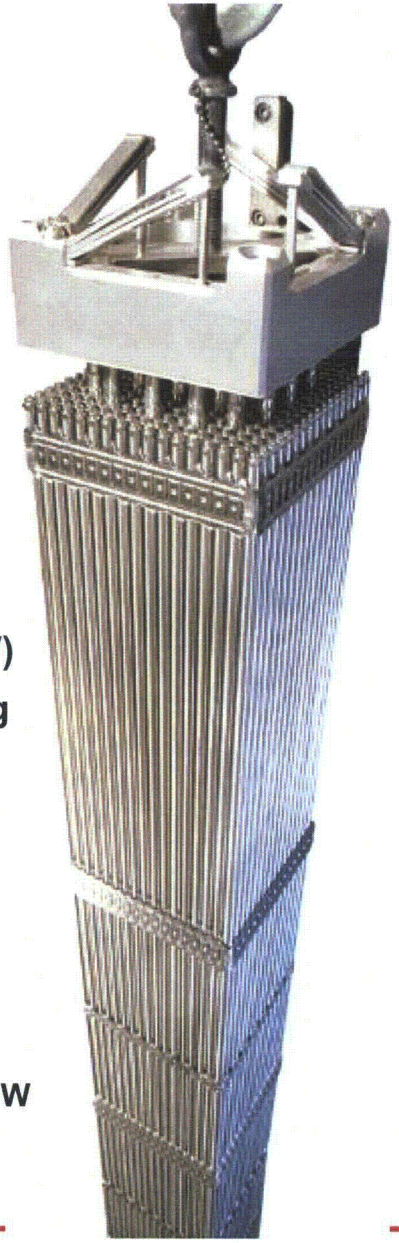
ATRIUM™ 10XM

Boiling water reactor (BWR) fuel – Plant Designs

- GE
- Siemens
- ASEA

Pressurized water reactor (PWR) fuel – Plant Designs

- Babcock & Wilcox (B&W)
- Combustion Engineering (C-E)
- Westinghouse (W)
- Siemens
- AREVA NP



Advanced Mark-BW

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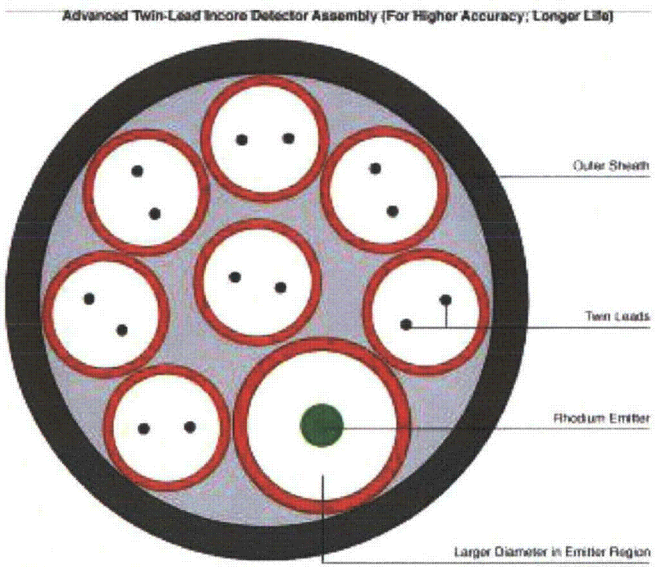
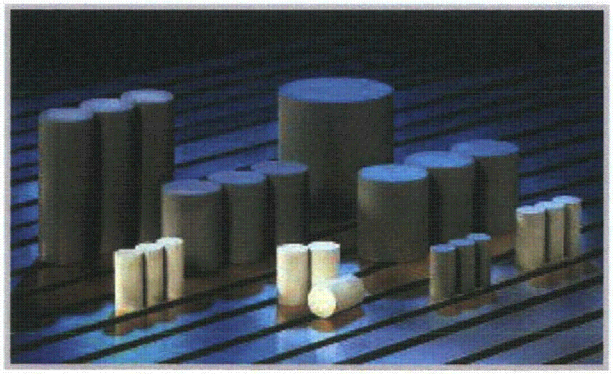
Fuel-Related Components

- Control components for B&W, C-E and W plants
- Burnable poison rod assemblies for B&W, C-E and W plants
- Incore detectors for B&W, C-E and W plants
- Flux thimble thermocouple tubes for W plants



Incore Detector



Burnable Poison Pellets

BPRAs

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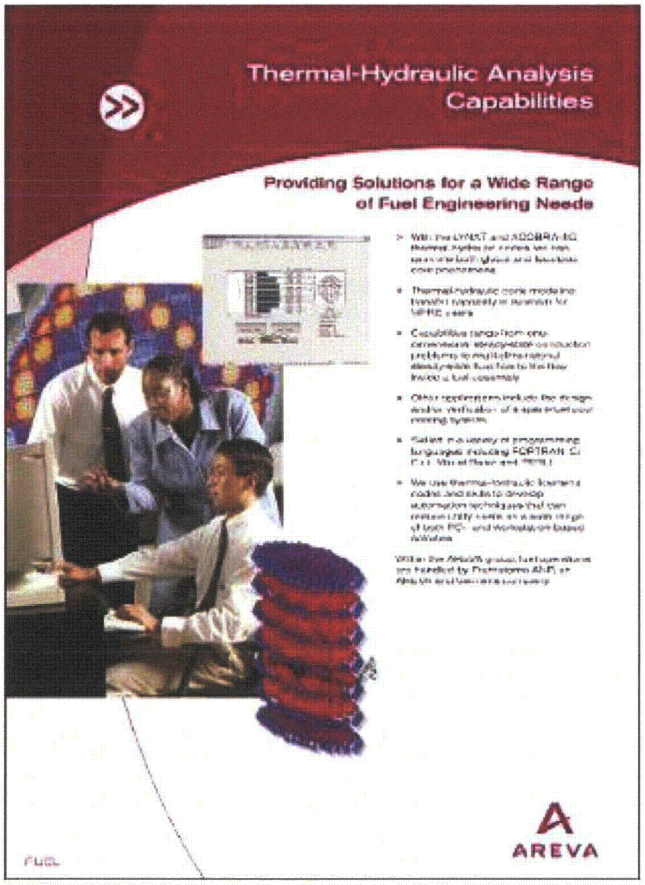
Engineering Services

- Core design and licensing
- Licensed power increases
- Reactor engineering analysis and on-site support
- Criticality analysis for spent fuel
- Structural analysis for fuel and related products
- Technology transfer for reload licensing and related codes and methods
- Advanced realistic large-break LOCA methodology
- Special studies



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Fuel Services

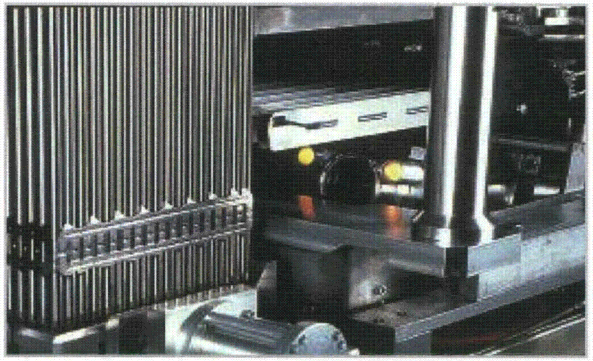
Fuel Inspections

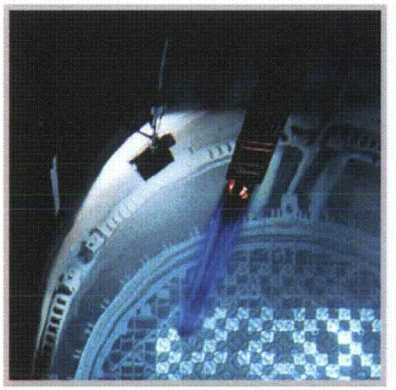
- Ultrasound
- Eddy Current
- Fuel Sipping

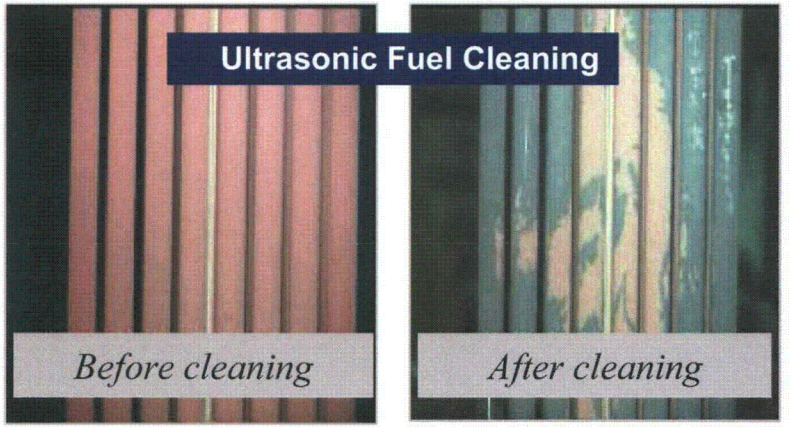
Fuel Cleaning

Fuel Repair & Reconstitution

Specialty Tooling







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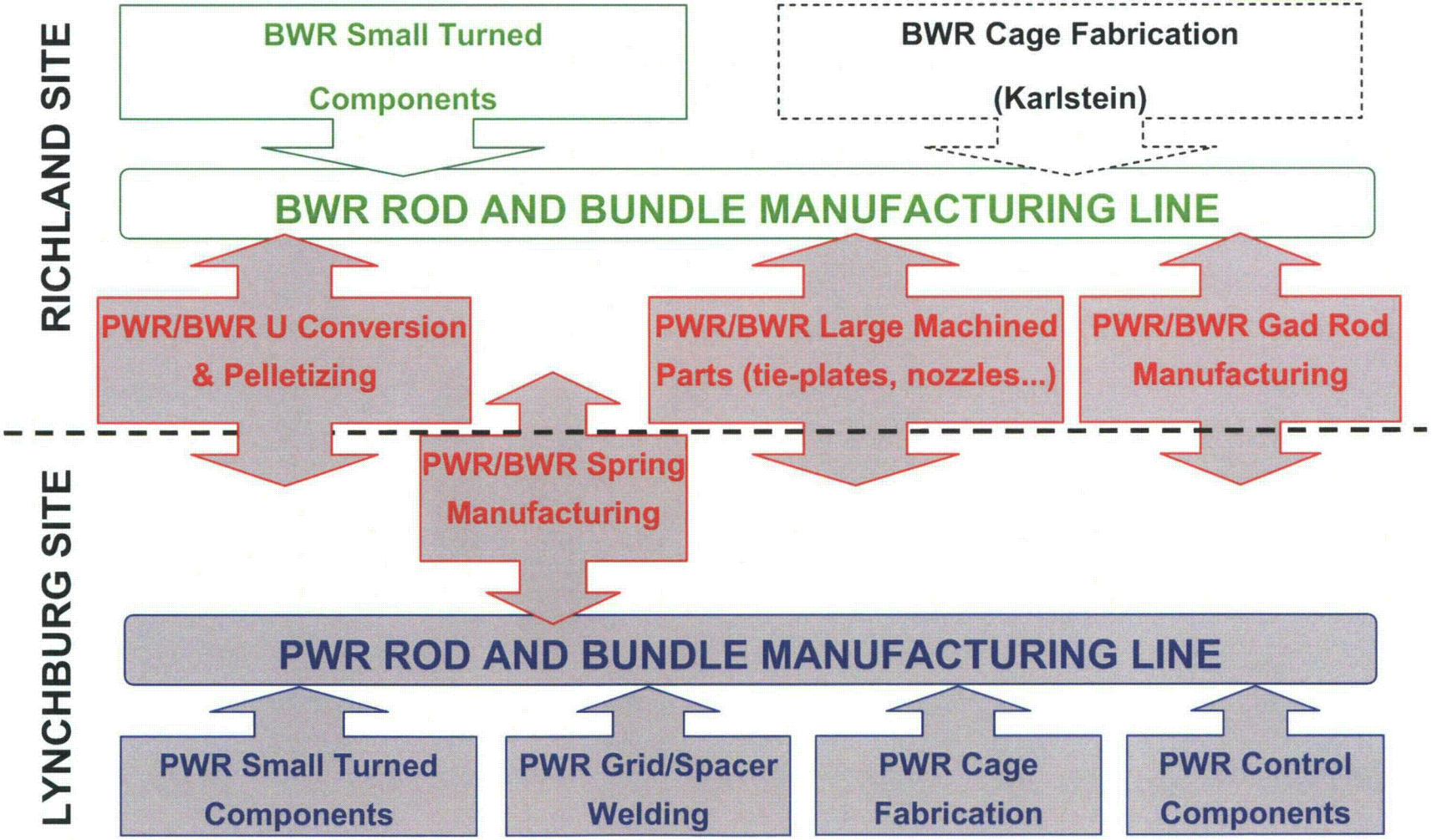
Centers of Excellence



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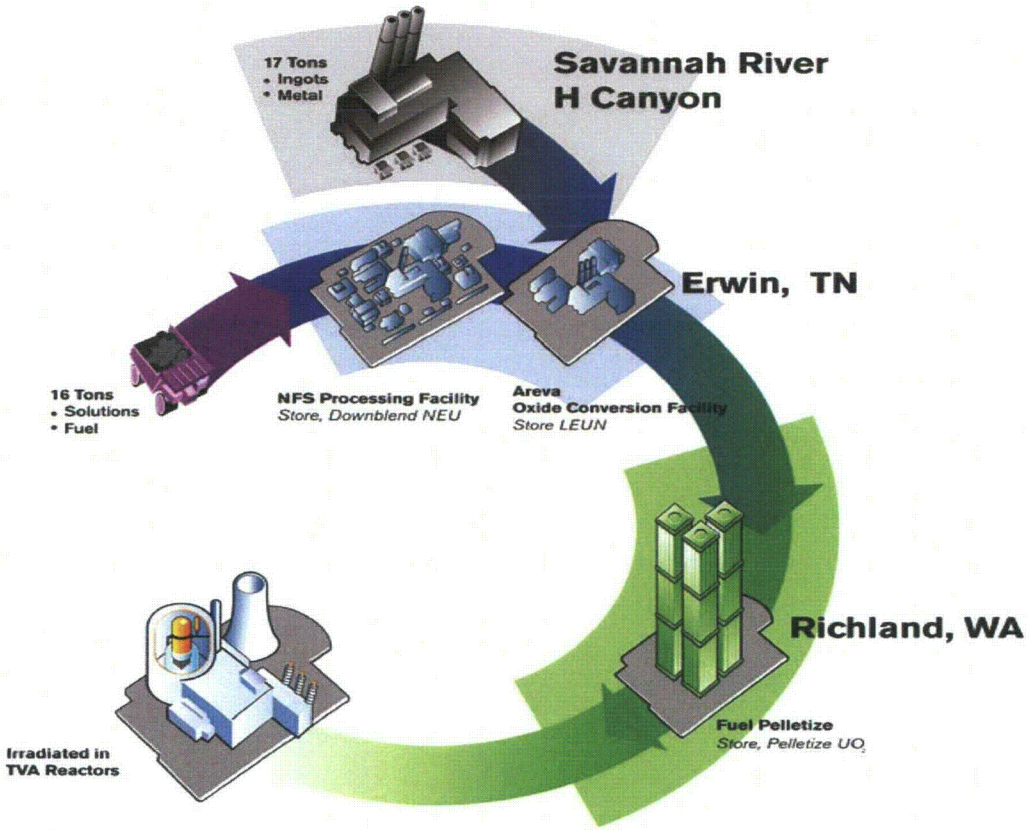
Operations
Mt. Athos Road Facility

- Constructed in 1968
  - Facility size: 200,000 square feet (Fuel & Nuclear Services)
  - Shipped over 15,000 fuel assemblies since 1971
  - Shipped over 2,500 incore detectors since 1983
  - Fabrication and supply of fuel related components : B_4C and B_4C – Alumina pellets, incore detectors, control components, machined components and springs
  - Average workforce experience: 12 years
- Nuclear Services Equipment & Refurbishment facilities on site
- Pump and Motor Services facilities on site
- Certifications: ISO 9001, ISO 14001 and OHSAS 18001

Operations
Horn Rapids Road Facility

- Constructed in 1971
- Facility size: 404,000 square feet
- Shipped over 43,000 fuel assemblies since 1971
- Over 200 powder shipments since 1990
- Over 350 pellets / rod shipments since 1989
- One-millionth ATRIUM™ 10 rods produced – September 2007
- Average workforce experience: 15 years
- Analytical and materials laboratories
- Advanced fuel design testing facilities
- Certification: ISO 9001

Operations
BLEU Conversion Complex



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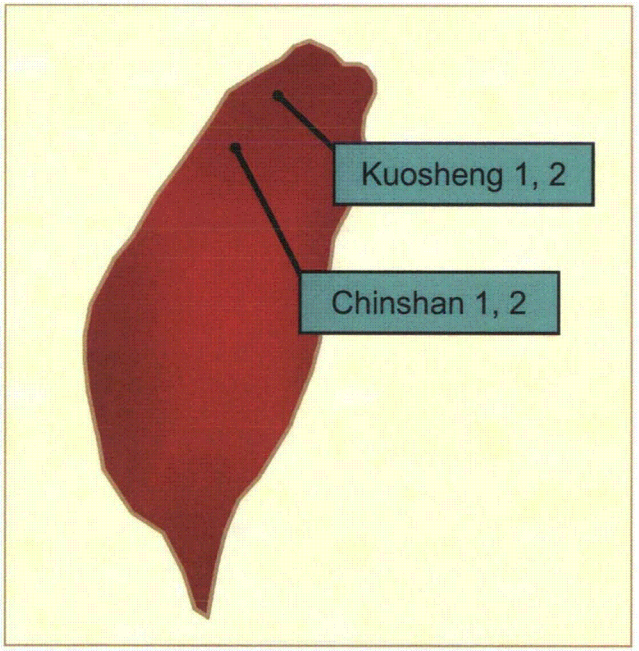
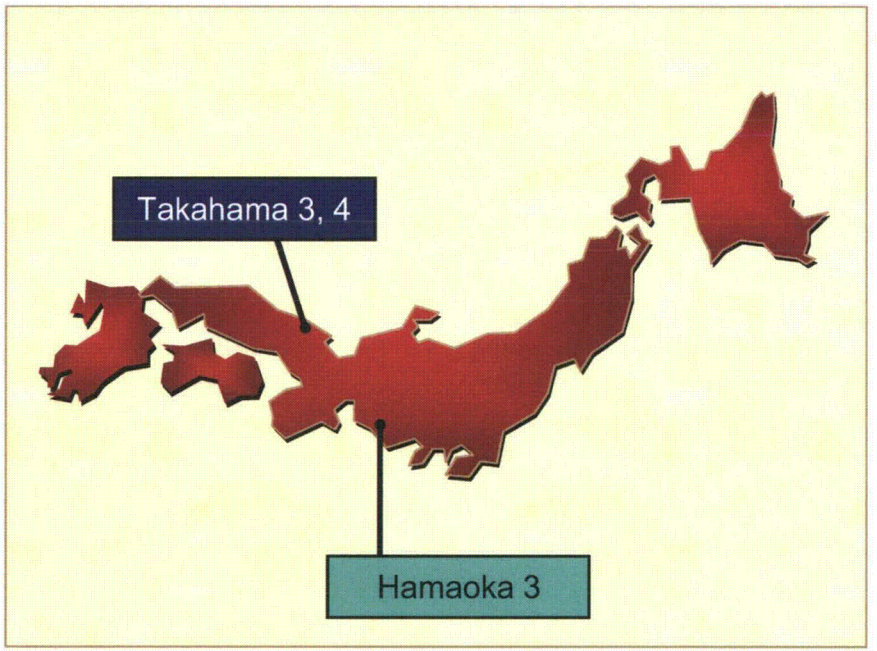
Fuel Customers in Taiwan and Japan



Taiwan



Japan

BWR

PWR

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Topical Reports Currently Under Review by the NRC

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Topical Reports Under Review

Reports Related to Thermal Hydraulics (LOCA & CHF)

- **ANP-10300P – AURORA-B: Evaluation Model for BWRs: Application to Transient and Accident Scenarios – Submitted December 2009**
  - Status
    - In acceptance review: draft acceptance review comments per LIC-190 verbally released on February 2, 2010; awaiting final acceptance review letter

- **ANP-10301P – COBRA-FLX: A Core Thermal-Hydraulic Analysis Code – Submitted March 2010**
  - Status
    - In acceptance review
    - Tentative May 17, 2010 post-submittal meeting

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Topical Reports Under Review

Reports Related to System Analysis

- **EMF-3028P - RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain, Volumes 1 & 2 – Submitted May 2008**
  - Status
    - Comments on draft SE submitted to the NRC on March 22, 2010; awaiting release of final SE

- **ANP-10303 – SIVAT: TELEPERM XS™ Simulation Validation Test Tool – Submitted June 2009**
  - Status
    - Draft RAI received March 10, 2010; awaiting release of final RAI

- **ANP-10301P – Statistical Universal Power Reconstruction with Fixed Margin Technical Specifications – Submitted September 2009**
  - Status
    - Acceptance review milestones verbally received on March 15, 2010; awaiting release of formal acceptance letter
    - Awaiting notice of necessity of post-submittal meeting

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Topical Reports Under Review

Reports Related to System Analysis

- **ANP-10307P – MCPR Safety Limit Methodology for BWRs – Submitted October 2009**
  - Status
    - Acceptance review; awaiting notification of milestones

- **ANP-10297P – The ARCADIA® Reactor Analysis System for PWRs Methodology Description & Benchmarking Results – Submitted March 2010**
  - Status
    - Acceptance review
    - Tentative May 17, 2010 post-submittal meeting

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Topical Reports Under Review

Reports Related to Fuel Rod Design and Mechanical Analysis

- **BAW-10133PA, Revision 1, Addendum 3 – Grid Testing Methodology Update – Submitted April 2009**
  - Status
    - Continuing RAI discussions

- **BAW-10247PA, Revision 0, Addendum 1 – Realistic Thermal-Mechanical Fuel Rod Methodology for BWRs – Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding – Submitted December 2009**
  - Status
    - Acceptance review; awaiting notification of milestones

- **EMF-93-177PA, Revision 1, Supplement 1 – Mechanical Design for BWR Fuel Channels – Advanced Methods for New Channel Designs – Submitted March 2010**
  - Status
    - Acceptance review

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Future Topical Reports

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Future Topical Reports

Reports Related to Thermal Hydraulics (LOCA & CHF)

- EMF-2103, Revision 2 – SRELAP5 RLBLOCA Methodology – June 2010
- Spay Heat Transfer Coefficients for the NGP 11x11 Fuel Design using F-COBRA-TF – December 2011

Reports Related to Fuel Rod Design and Analysis

- BAW-10133, Revision 1, Addendum 4 – Revised Seismic Analysis – August 2010
- BAW-10147, Revision 2 – Fuel Rod Bowing in B&W Fuel Designs – March 2011
- BAW-10231, Revision 3 – COPERNIC3 – June 2011

Reports Related to System Analysis

- ANP-10256, Revision 3 – Methodology for Analysis of Control Rod Withdrawal Error for BWR Plants with ARTS – July 2010

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PWR Fuel Designs

Patrick F. Williams
PWR Local Product Manager

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Presentation Outline

U.S. PWR Fuel Supply

Current PWR Fuel Designs and Planned Design Transitions

Summary of PWR Lead Assembly Programs

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Fuel Supply – Operating Plants by OEM

Babcock & Wilcox	Westinghouse	Combustion Engineering
Oconee 1,2,3	Sequoyah 1,2 (LAs)	Millstone-2
ANO1	North Anna 1,2	St. Lucie-1
TMI-1	Harris	Ft. Calhoun
Davis Besse	Robinson	Calvert Cliffs 1,2 (LAs)
Crystal River 3	Braidwood (LAs)	Palisades
		Palo Verde (LAs)
		SONGS (LAs)

Batch supply to 19 plants; lead assemblies in 5 plants

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Fuel Designs – Operating Plants by OEM

Babcock & Wilcox	Westinghouse	Combustion Engineering
All seven units –	SEQ 1,2: Mark-BW 17x17	MIL, SLU1: 14x14 HTP
Mark-BHTP 15x15	NAN 1,2: Adv. Mark-BW 17x17	FTC: 14 x 14 HTP
(Transition cores w/ Mk-B11 and B12)	Harris: 17x17 HTP	PAL: 15 x 15 HTP
	Robinson: 15x15 HTP	CCL 1,2: Adv. 14x14 HTP
	SEQ 1 & BRW: Adv. Mark-BW(A)	PVE1: 16x16 HTP
		SONGS: 16x16 HTP

14 PWR fuel bundle designs

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U.S. Fuel Zr-Alloy Deployment

OEM	Lattice	Plant	clad tubes	structural tubes/bars	spacers
B&W	15x15	Oconee-1,2,3			
		ANO1			
		Davis Besse			
		TMI-1			
		Crystal River-3			
W	17x17	North Anna 1,2			
		SEQ1 & BRW LAs			
		Sequoyah 1,2			
		Harris			
	15x15	Robinson			
CE	14x14	Ft. Calhoun			
		Milstone-2			
		St. Lucie-1			
		Calvert Cliffs-1,2			
	15x15	Palisades			
	16x16	Palo Verde			
		SONGS			

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Mark-B-HTP Design

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HTP Spacer Grid Design

- ◇ "Dual line contact" rod support system
- ◇ Robust construction
- ◇ Welded directly to guide tubes
- ◇ High fuel rod fretting resistance
- ◇ Curved flow channels for flow mixing

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HMP and IFM Grids

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Structural Components – cont'd

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Transition at B&W 177-Plants

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Mark-BW and W-HTP Designs

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Mark-BW and Adv. Mark BW Designs

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Comparison of Design Features – Mark-BW vs. Adv. Mark-BW

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Mark-BW Keyed Spacer Grids

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Mark-BW and W-HTP Lower End Fitting

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Robinson (W-15x15) & Harris (W 17x17)

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Robinson & Harris Transitions

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CE-HTP Designs

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CE Design General Characteristics

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CE 16x16 Structure

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CE 14x14 and 15x15 HTP Design Transitions

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Lead Assembly Programs

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CE Lead Assembly Programs

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Advanced Mk-BW(A)

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PWR Core Engineering Methods

Thomas L. Lotz
Manager, PWR Core Engineering

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Current Code Systems – Global Diversity

	Lynchburg	Richland	Paris/Lyon	Erlangen
Cross-Section	CASMO-3	CASMO-3	APOLLO-2	CASMO-3
Neutronic Simulator	NEMO	PRISM	SMART	PRISM
Kinetic Simulator	NEMO-K	---	SMART-K	PANBOX
Thermal-Hydraulics	LYNXT	XCOBRA-IIIC	FLICA-IIIF	COBRA-3CP
Fuel Rod	TACO3	RODEX2A	COPERNIC	CARO

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The Future – and Beyond!

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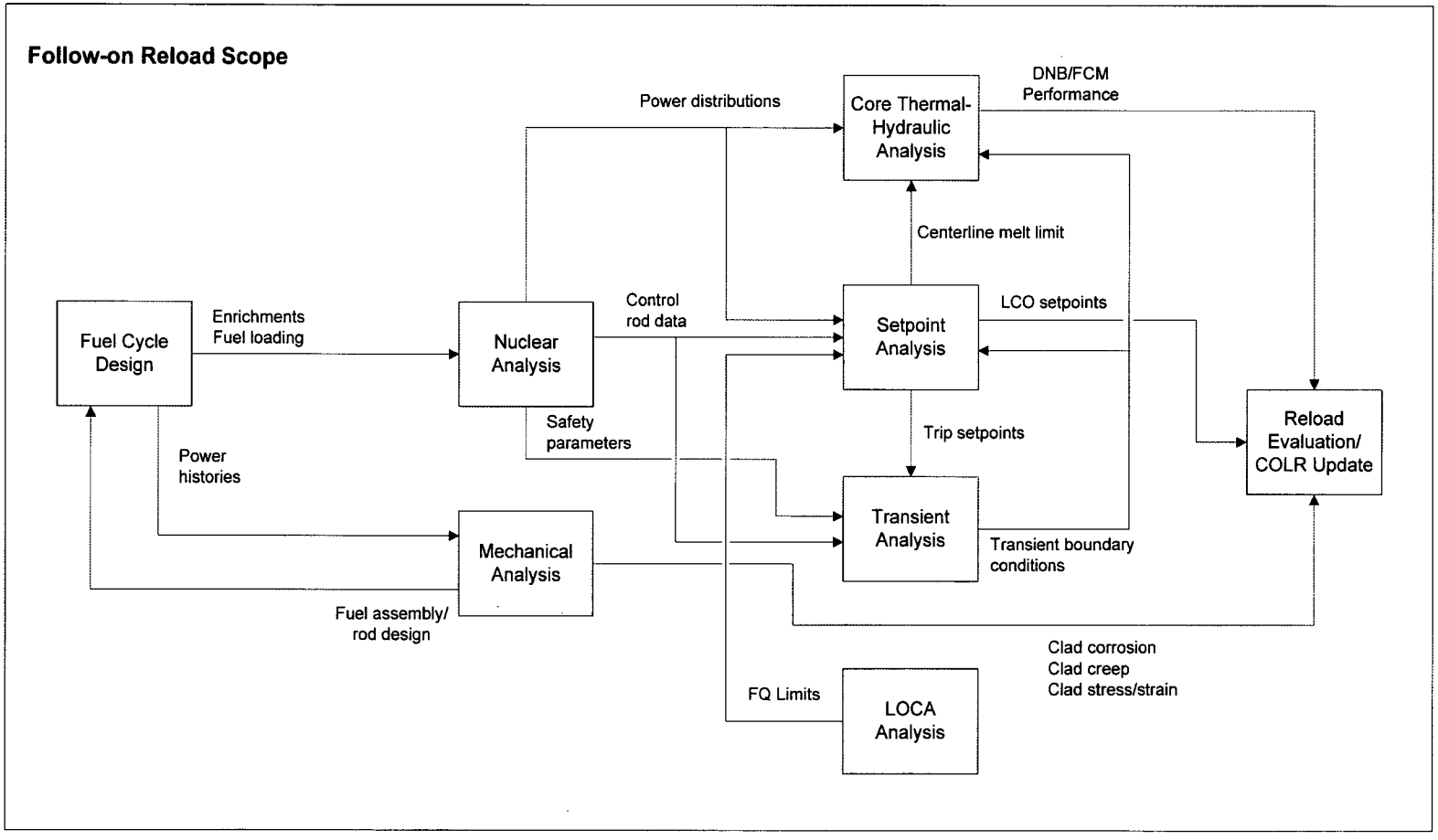
Reload Process – PWR Methods



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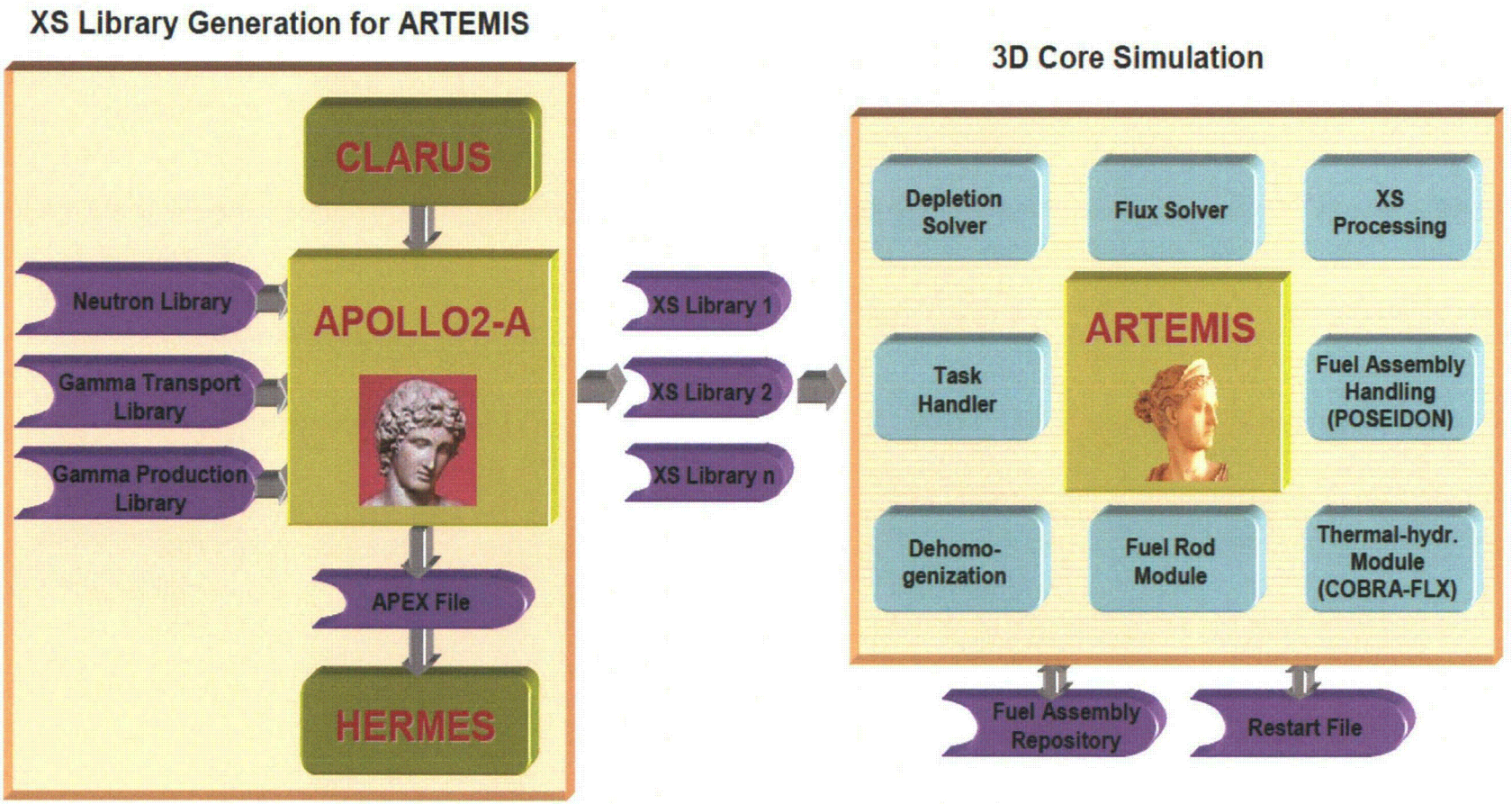
The ARCADIA® Calculation Scheme



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CONVERGENCE

The Thermal Hydraulic Module (COBRA-FLX)

COBRA 3-CP

FLICA III F

LYNXT

XCOBRA

→ **COBRA-FLX**

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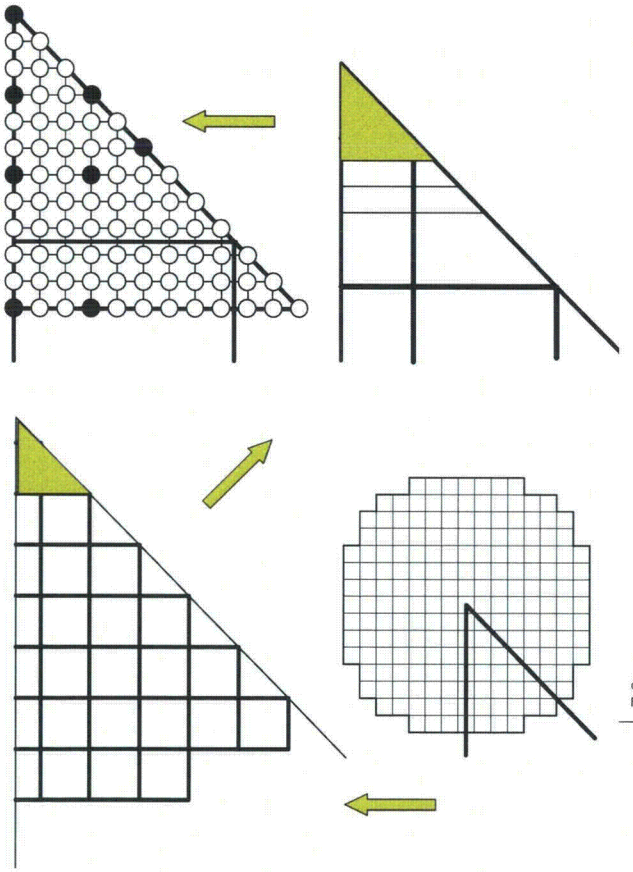
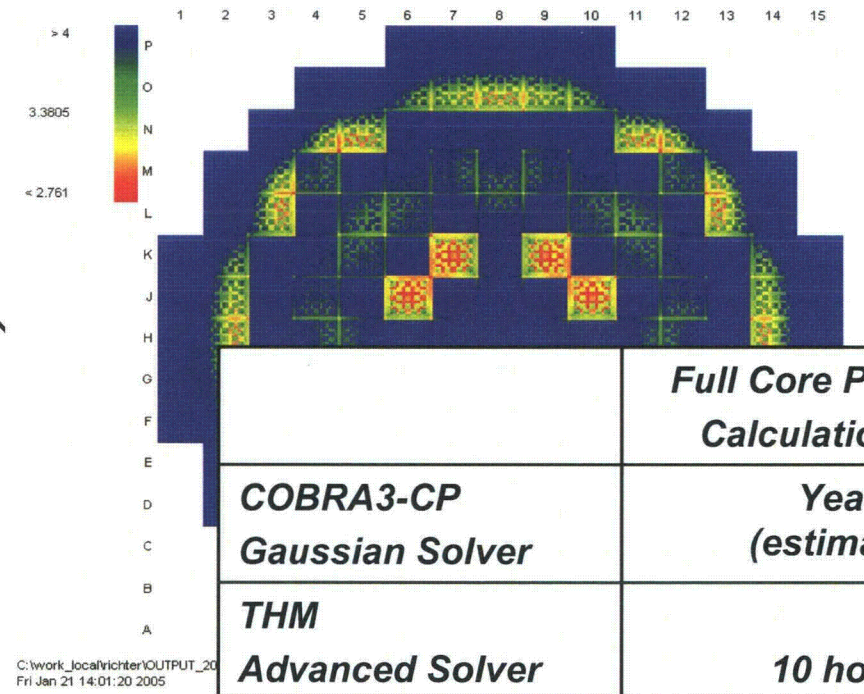
Thermal-Hydraulic Module Numerical Acceleration and Code Restructuring





	Full Core Pin-by-Pin Calculation Time
COBRA3-CP Gaussian Solver	Years (estimated)
THM Advanced Solver	10 hours
THM Advanced Solver + Code Restructuring	14 min

Typical example: 1/8 core (100 channels)

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COPERNIC3 Development

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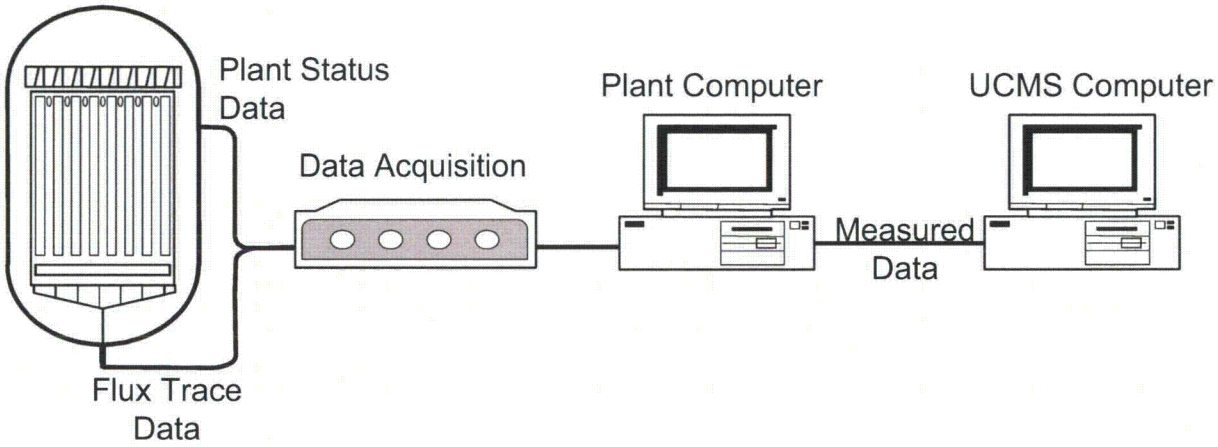
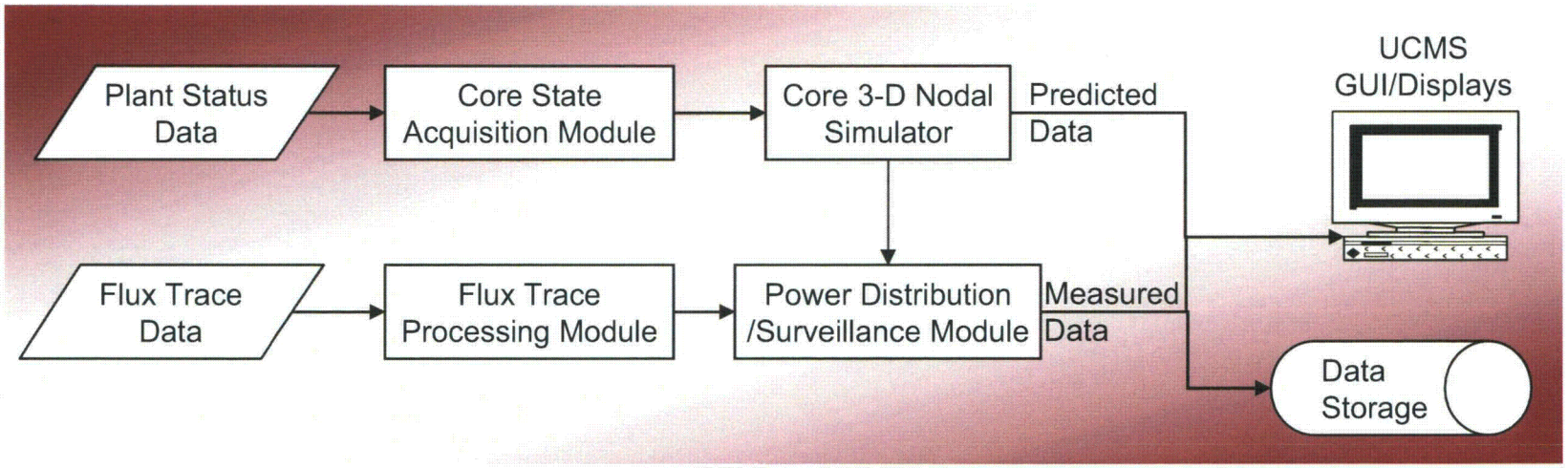
UCMS Modularity





Modularity is Key

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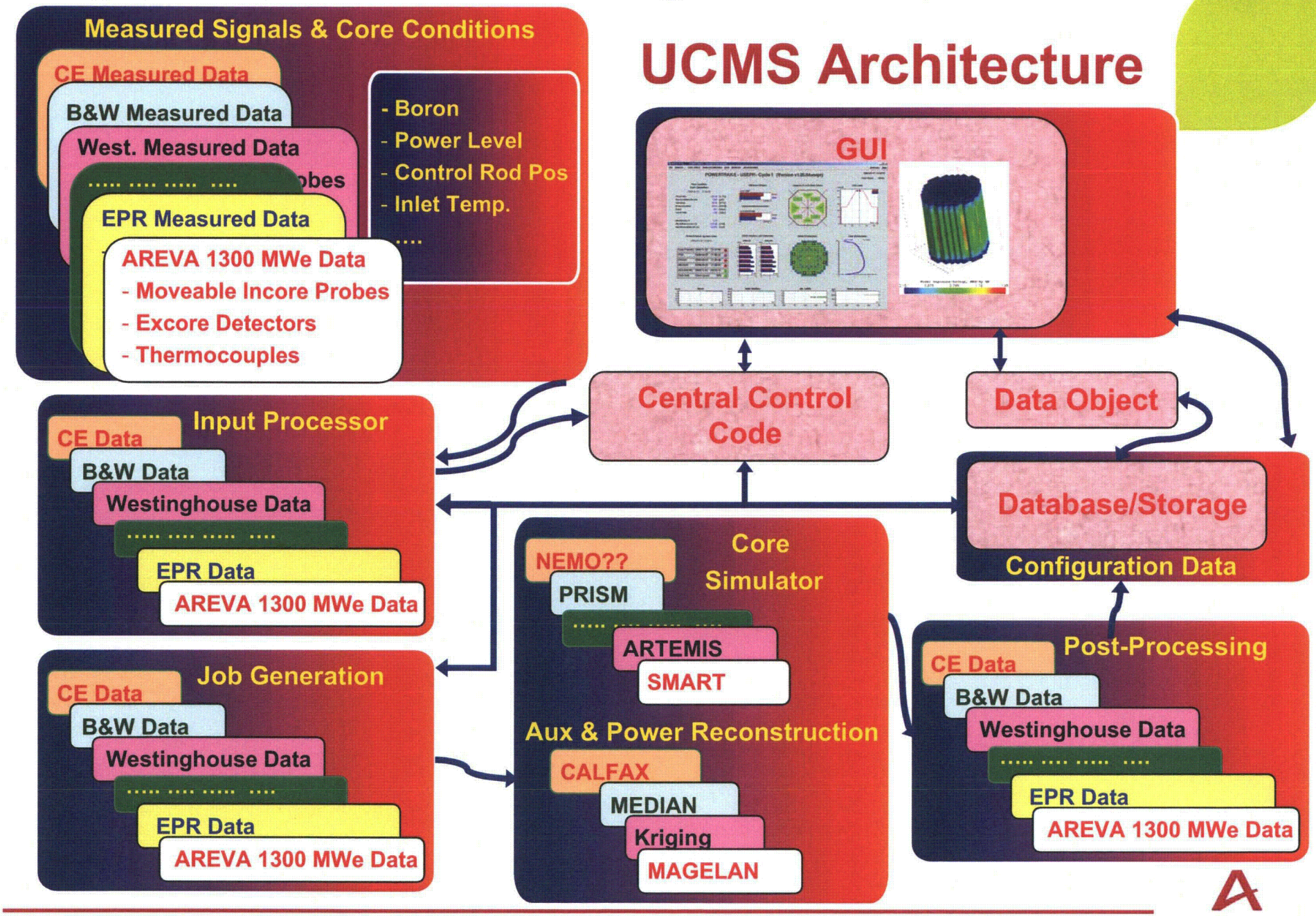
UCMS Architecture



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Future Topical Reports

Reports Related to Specific Applications & System Analysis

- Application Topicals (i.e., BAW-10179PA, etc.)
- Ejected Rod Method(s)
- BAW-10163P-A "Core Operating Limit Methodology for Westinghouse-Designed PWRs"
- Etc.

Reports Related to Fuel Rod Design and Analysis

- BAW-10231, Revision 3 – COPERNIC3 – June 2011
- BAW-10147, Revision 2 – Fuel Rod Bowing in B&W Fuel Designs – March 2011
- Etc.

Reports to Support Future Products

- POWERTRAX upgrade
- Fuel Product Mechanical Designs
- Etc.

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Open Discussion

Implementation Process – Next 2 years

- Evaluate Topical Report currently used
- Consolidate, Update, or Eliminate Topicals
- Execute analysis process with existing cycles

Analysis Challenges

- New fuel product materials
- Support additional operational risk analysis

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BWR Fuel Designs

Norman L. Garner
Technical Sales Manager – BWR Fuel

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BWR Fuel Design - Presentation Topics

Improving ATRIUM™ 10 Reliability

- ◇ Recent upgrades

ATRIUM™ 10XM Reload Introduction

- ◇ Design review
- ◇ Reload readiness
- ◇ Extended design validation

ATRIUM™ 11 Development Update

- ◇ Design review
- ◇ Prototyping and testing status

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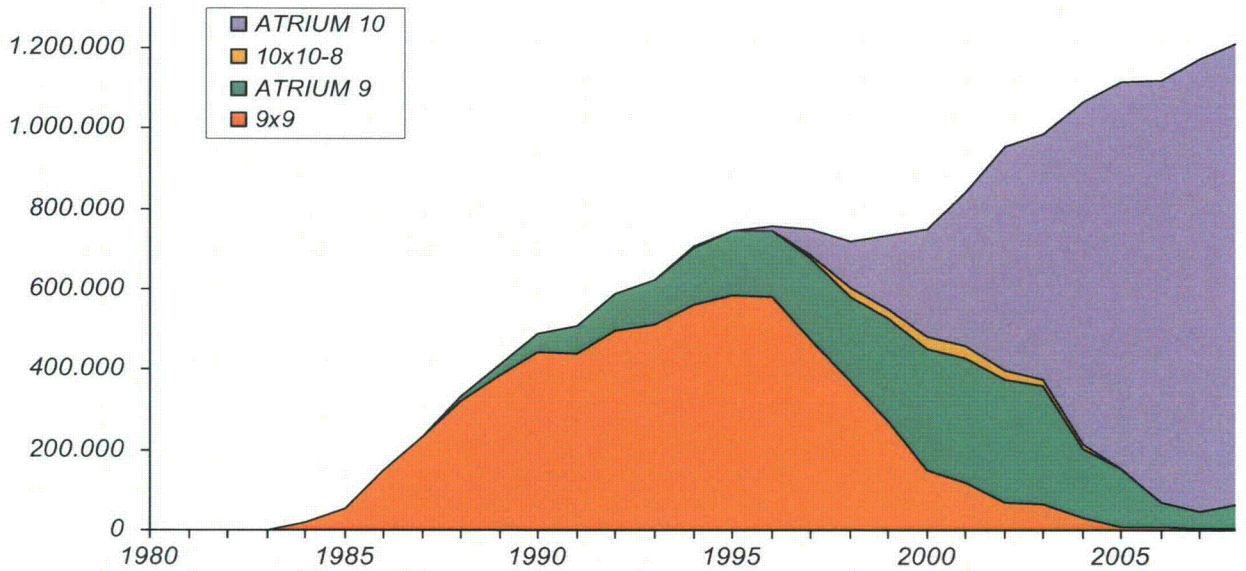
BWR Fuel Product Lifecycle

Number of Fuel Rods in Operation



9x9

ATRIUM 10

ATRIUM 10XP/XM

AREVA's BWR fuel R&D program is a sustained, continuous process of stewardship for existing products and new product development

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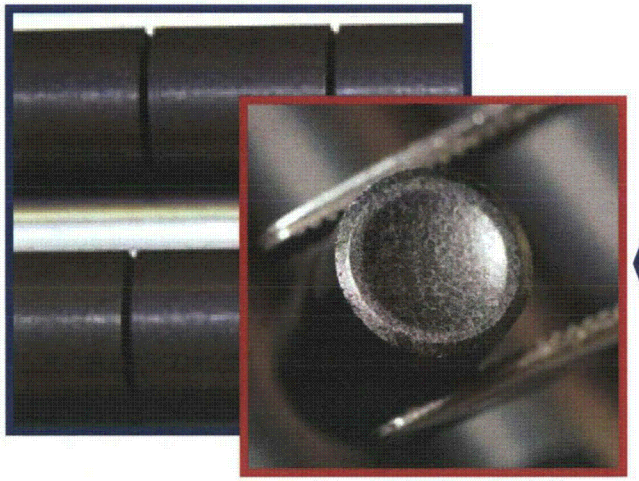
ATRIUM™ 10 Stewardship - Reliability Upgrades

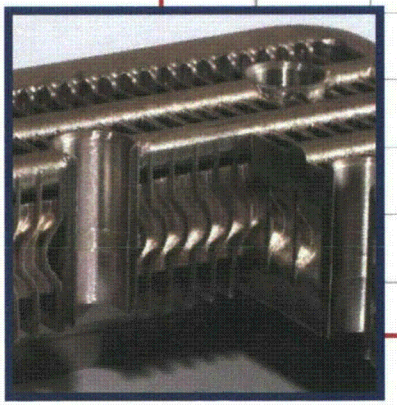


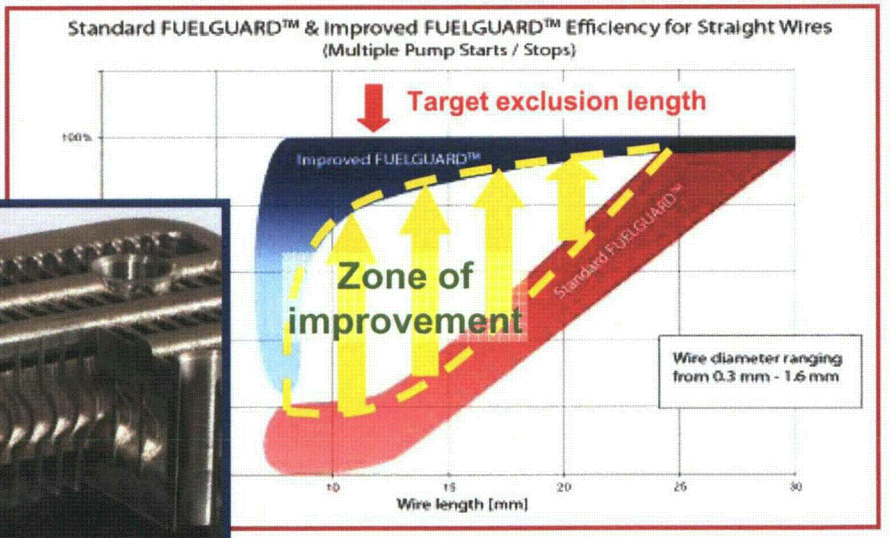
Chip resistant chamfered pellet reduces risk of missing pellet surface to enhance PCI margin

- now implemented for all US/Taiwan ATRIUM™ 10 reloads

Improved FUELGUARD™ filtering grid to enhance debris fretting protection

- First reload now in core





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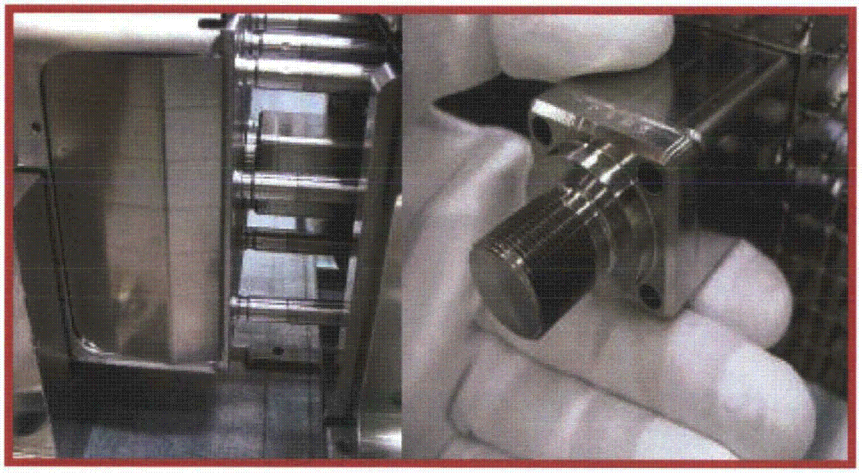
ATRIUM™ 10 Stewardship - Reliability Upgrades



Computational Fluid Dynamics calculations validated by flow tests were used to size inlet holes

Water channel inlet moved above FUELGUARD™ grid to enhance debris fretting protection

- **Coupled with Improved FUELGUARD™ upgrade**

Winged fuel channel fastener to enhance resistance to handling damage

- **Optional feature in select reload use**

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ATRIUM™ 10 Stewardship - Reliability Upgrades

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ATRIUM™ 10XM

Reload Readiness

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ATRIUM™ 10x10 Family

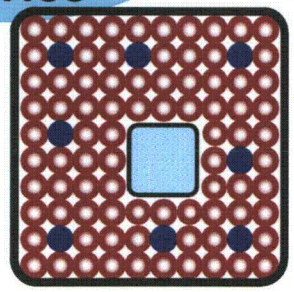
Continuity of proven features and components

>1.7M rods in service

improved stability

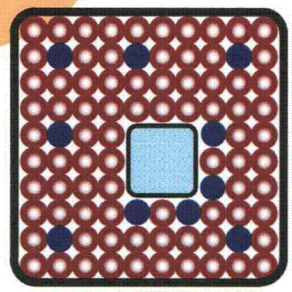
improved critical power

improved CSDM



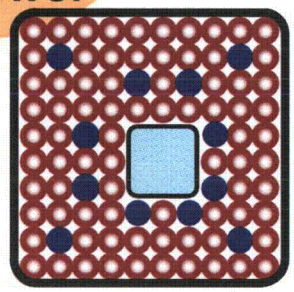
ATRIUM™ 10

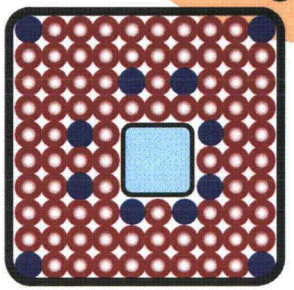
- ▶8 bi-metallic ULTRAFLOW™ grids
- ▶10.05 mm rod diameter
- ▶8 PLFRs



ATRIUM™ 10XP

- ▶8 Ni-alloy 718 ULTRAFLOW™ grids
- ▶10.28 mm rod diameter
- ▶3 water channel crowns
- ▶10 PLFRs





ATRIUM™ 10XM

- ▶9 Ni-alloy 718 ULTRAFLOW™ grids
- ▶10.28 mm rod diameter
- ▶5 water channel crowns
- ▶12 PLFRs

1992 — 2002 — 2005 (EU) — 2009 (US)

Deliberate continuity retains successful operating experience

● *Denotes Part Length Fuel Rod*

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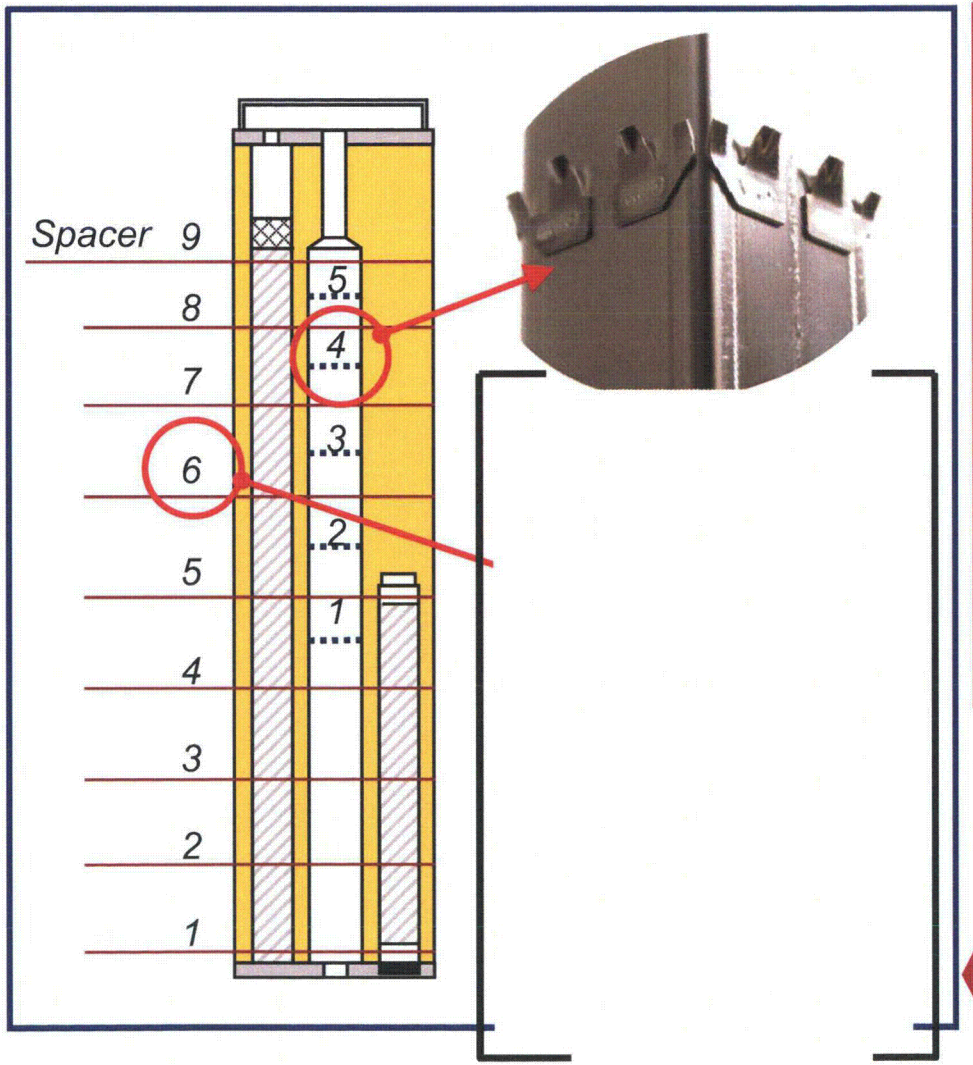
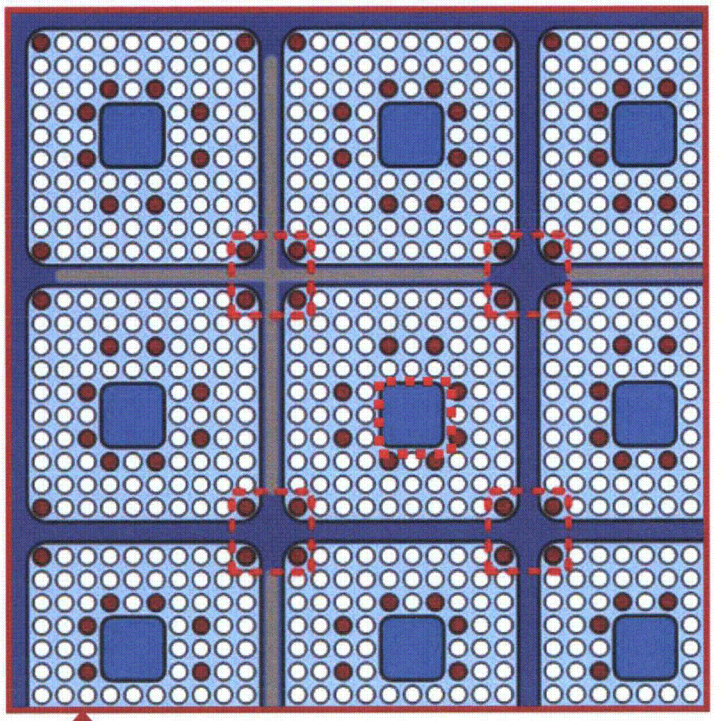
Performance-related Features of the ATRIUM™ 10XM





12 part-length fuel rods, including corner positions, and central water channel enhance cold shutdown margin

8 vaned ULTRAFLOW™ spacer grids in active region, progressively reduced spacer spans, and water channel crowns increase critical power capability

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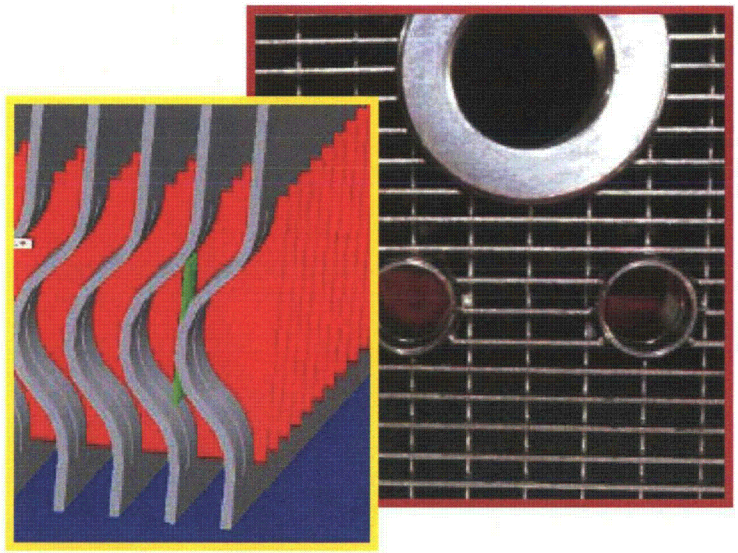
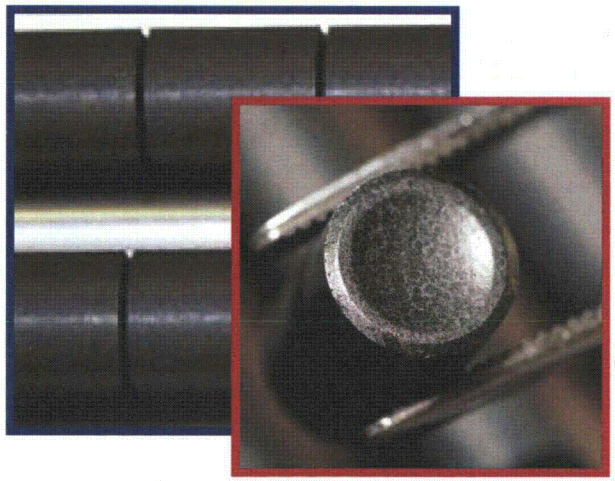
Reliability-related Features of the ATRIUM™ 10XM

Parameter	ATRIUM™ 9	ATRIUM™ 10XM	ATRIUM™ 10
Clad OD	11.00 mm	10.28 mm	10.05 mm

Fuel rod thickness-to-diameter ratio maintained consistent with benchmark data for mechanical methods to ensure excellent PCI margin



Improved FUELGUARD™ provides enhanced inlet filtering of chips and wires to reduce debris fretting risk



Chip resistant chamfered pellet and Automated Pellet Inspection System reduce risk of missing pellet surface to enhance PCI margin

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Spacer Grid Details

The ATRIUM™ 10XM retains the proven rod support and coolant mixing features of the preceding ATRIUM™ 9 and ATRIUM™ 10 ULTRAFLOW™ spacer grids:

Rod is supported at six points within each spacer grid by alloy 718 clips to provide stable rod constraint throughout life

Vane directions alternate across grid to assure no risk of flow induced vibration from vanes

- ◇ No net loads to cause spacer vibration
- ◇ No net spin forces on fuel rods from flow-induced drag

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ATRIUM™ 10XP/10XM

- Extensive experience base is growing rapidly

Plant	Plant Type	Power Density (kWt/l)	Maximum Core Flow (kg/s/bundle)	ATRIUM™ 10x10 Type	First Insertion Year	Number Irradiated Bundles	Number Irradiated Rods	Peak Assembly Burnup (MWd/kgU)
C04	Siemens KWU	56.8	19.9	10XP	2002	316	28,756	**67**
C05	Siemens KWU	56.8	19.9	10XP	2006	192	17,472	34
C22	GE BWR6	**62.8**	18.1	10XP 10XM	2003 2005	16 252	1,456 22,932	59 50
C21	Siemens KWU	50.4	18.4	10XP	2004	340	30,940	51
C12	ASEA	49.8	17.8	10XP 10XM	2004 2009	4 118	364 10,738	41 3
C20	GE BWR6	58.6	17.9	10XP	2005	32	2,912	44
C11	Siemens KWU	51.1	19.8	10XP	2006	252	22,932	38
C24	ASEA	49.8	17.8	10XM	2006	30	2,730	29
C10	Siemens KWU	49.6	**20.2**	10XP	2006	20	1,820	11
C13	ASEA	54.0	18.7	10XM	2008	4	364	11
C17	ASEA	57.5	16.7	10XM	2009	8	728	8
A33	GE BWR5	48.7	17.4	10XM	2009	8	728	6
C16	ASEA	45.7	18.4	10XM	2009	8	728	2
Total as of October 2009						**1,600**	**145,600**	**67**

Annual PIE of lead introductions have provided a substantial volume of performance data, including high duty operations

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Initial US Reload Timeline

Activity	2009				2010				2011
	Q1	Q2	Q3	Q4	Q1	Q2	Q3	Q4	Q1
ACE Review	Respond to RAIs			Draft SER					
					Publication of Topical Report				
		NRC Audit (Richland)							
Regulatory Compliance				(AREVA)	Generic Compliance Doc.				
				(Licensee)	Methods Reference LAR				
Reload Licensing Analysis				FSDD Notice					
				Core Engineering					
Fabrication and Delivery	Reserve Starting Material			Order Long Lead Items					
					Fabrication				Startup

Delivery: December 2010

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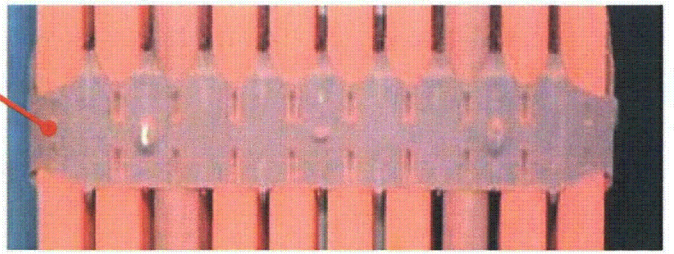
Generic Compliance Document

Per AREVA's NRC approved methodology for new fuel introduction, an ATRIUM™ 10XM Generic Compliance Document is provided to the NRC (information basis)

In addition to a generic licensing evaluation, the document provides:

- Mechanical Design Description
- Operating Experience Summary
- Post Irradiation Examination Results
- Provides basis for continued applicability of approved design and licensing methods

	# of Assemblies	Max Rod GWd/MTU
ATRIUM 10XP	1,172	67
ATRIUM 10XM	428	50
Total	1,600	67



Comprehensive source for ATRIUM™ 10XM design validation and compliance information

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Extended Design Validation

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Extended Design Validation

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ATRIUM™ 10XM Manufacturing Readiness

All manufacturing processes are qualified

- Qualification of manufacturing facilities via Proof of Fabrication and Lead Assembly campaigns
- EU facilities already support multiple reload campaigns

Chip free manufacturing is ready

Multiple customers are expected to witness fabrication activities

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ATRIUM™ 10XM Performance

Fuel cycle studies have been conducted for all AREVA customers demonstrating significant performance and operational benefits across a wide range applications

- **Allows high reactivity fuel to be moved inward for radial leakage reduction and more uniform depletion history**

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ATRIUM™ 10XM Performance

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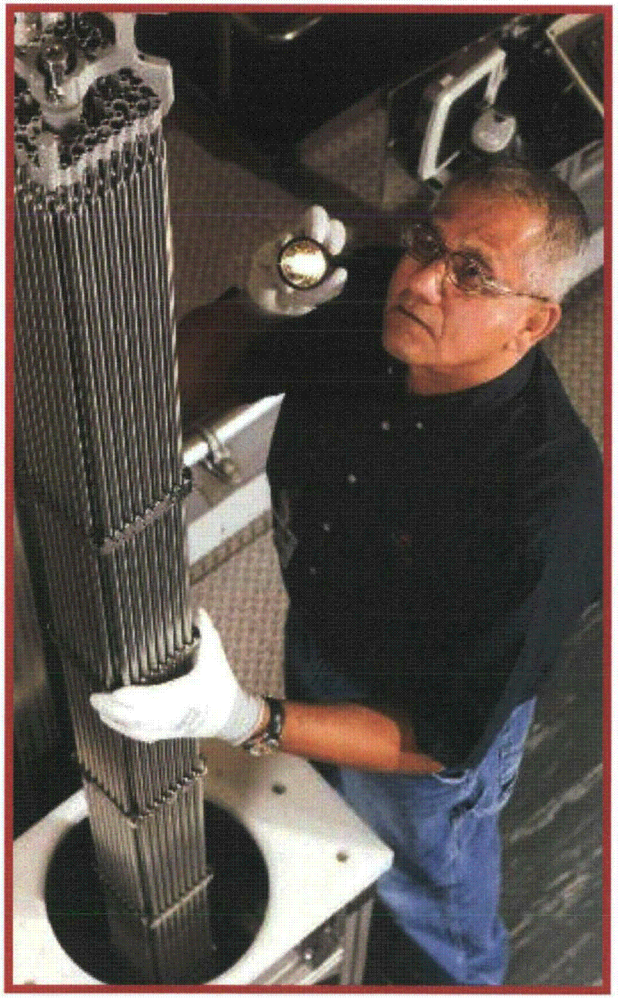
Summary



The ATRIUM™ 10XM delivers significant Fuel Cycle Cost benefits across a wide range of applications

Studies also illustrate the significant operational benefits of ATRIUM™ 10XM

- Reduced spent fuel inventory
- Reduced radial leakage to pressure vessel
- Lower burnup gradients in high reactivity fuel
- Reduced variation in hot excess reactivity

Deliberate continuity with proven features and specific well-tested enhancements assure the highest level of reliability

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ATRIUM™ 11

Development Progress

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Next Generation Product
Why ATRIUM™ 11?

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ATRIUM™ 11

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ATRIUM™ 11

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ATRIUM™ 11

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ATRIUM™ 11 Plan

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Summary

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BWR Core Engineering Methods

Douglas W. Pruitt
Manager, BWR Thermal Hydraulics

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Introduction
BWR Development Areas

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BWR Methodologies Current Status

Mechanical	Neutronics	Safety & Licensing

* Improved methods recently implemented

** These codes will be replaced by new methods

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Recent Submittals

SAFLIM-3D

- Submitted October 2009

[ ]

AURORA-B Deterministic AOO

- Submitted December 2009

[ ]

RODEX-4

- Submitted December 2009

[ ]

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BWR Methodologies Next Evolutions

Mechanical	Neutronics	Safety & Licensing

* Improved methods recently submitted

** New C&M currently being developed

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BWR Methodologies Future State

Mechanical | **Neutronics** | **Safety & Licensing**

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AURORA-B

AREVA's Realistic BWR Transient Simulator

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AURORA-B Platform

Advanced, realistic BWR transient simulator

- Eliminates conservatism from simpler physics → Operational Limits
- Wide range of applicability → Licensing and Operational Support
- Up-to-date code benchmarking and validation → Latest Understanding
- Modern methodology development principles

A best estimate multi-physics code system for simulating the coupled fuel, neutronic, and thermal hydraulic BWR system response

- RODEX4 best estimate fuel performance predictions
- MB2-K 3D kinetics built upon MICROBURN-B2
- S-RELAP5 modern two-fluid T/H system code

» **Consolidation of experience and a fundamental shift for the future**

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AURORA-B Development Focus

Immediate development activities [tory]
requirements and market forces

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AURORA-B Development Plan

Conclusions

The submittal of AURORA-B represents the first step in BWR transient and accident renovation

- ◇ Composed of well qualified and approved modules (S-RELAP5, MB2-K, RODEX4)
- ◇ Extensive qualification database

Incremental refinement and qualification of the base AURORA-B models will extend its applicability to additional transient and accident scenarios

- ◇ Reactivity Insertion Accidents
- ◇ Loss of Coolant Accidents
- ◇ Anticipated Transients Without Scram

BWR renovation facilitates the retirement of current BWR transient and accident methodologies

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PWR Operating Experience

John H. Strumpell
Manager, PWR Material and Thermal-Mechanics

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Topics

Operating Experience (OE)

Current Plant Status

Achievements in 2009

2010 Focus

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AREVA delivered its first M5® reloads

- In 1997 for M5 cladding (2000 in US)
- In 2001 for M5 cladding and guide tubes (US)
- In 2003 for M5 cladding, guide tubes and intermediate spacer grids (US)

In 2009, 60% of the PWR plants supplied by AREVA received M5 material

- 2/3 with M5 cladding

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M5® Alloy on the PWR Market (As of December 2009)

Components	*Plants*
M5 Clad Fuel Rods *(3,100,000)*	79
M5 Clad Fuel Assemblies *(12,600)*	79
M5 Clad Reloads *(290)*	66
M5 Clad and Guide Tube Assemblies *(6,300)*	42
All M5 Assemblies *(4,200)*	41

- Three main designs
  - AFA™, HTP™ (including Mark-B-HTP), Mark-BW
- Delivered in 13 different countries
- Six countries loaded more than 500 fuel assemblies
  - France
  - US
  - Germany
  - China
  - Belgium
  - Sweden

M5® is the Reference Alloy for PWR Fuel Assemblies Delivered by AREVA

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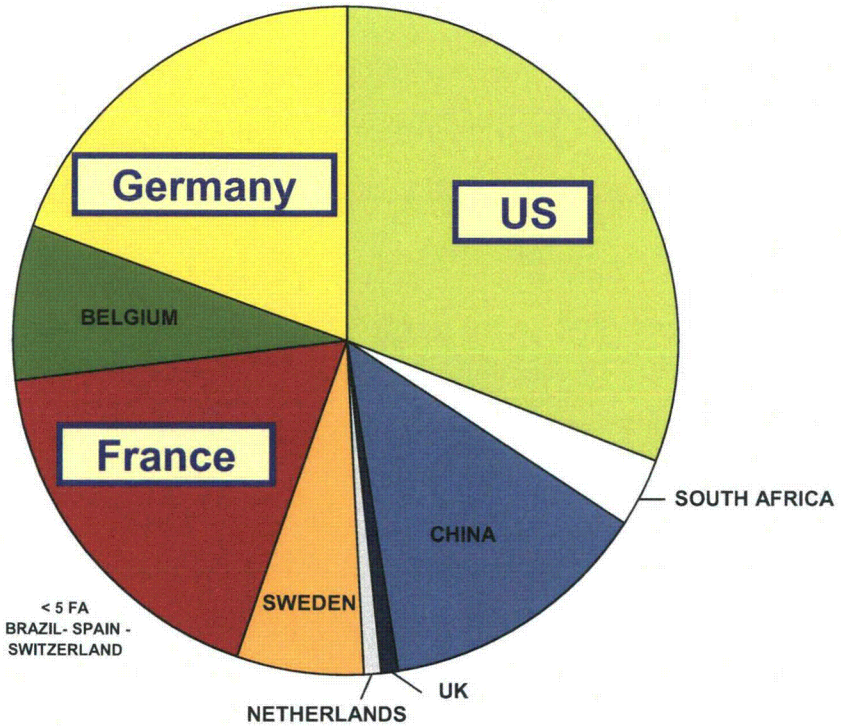
M5® Experience for Top 3 Markets

Fuel assemblies supplied per country



Country	Number of Reactors	T_{out} (°F)	Max fuel rod linear power integrated on a cycle (kW/ft)	Tested chemistry (Li content BOC - ppm)	Zn injection
Belgium	5	624	10.30	3.5	
Brazil	1	621	9.84	2.2	Yes
China	4	621	7.62	3,5	
France	30	624	7.25	2.2 & 3.5	
Germany	11	624	10.49	2	Yes
Netherlands	1	606	10.52	2.2	
South Africa	2	599	8.23	3.5	
Spain	1	619	6.74	3.5	
Sweden	3	617	9.91	5	
Switzerland	1	617	11.09	2.2	Yes
UK	1	617	7.53	2.2	
USA	16	622	7.04	3.5 to 4.9	Yes

M5® Irradiation Experience Covers a Wide Range of Demanding Operating Conditions

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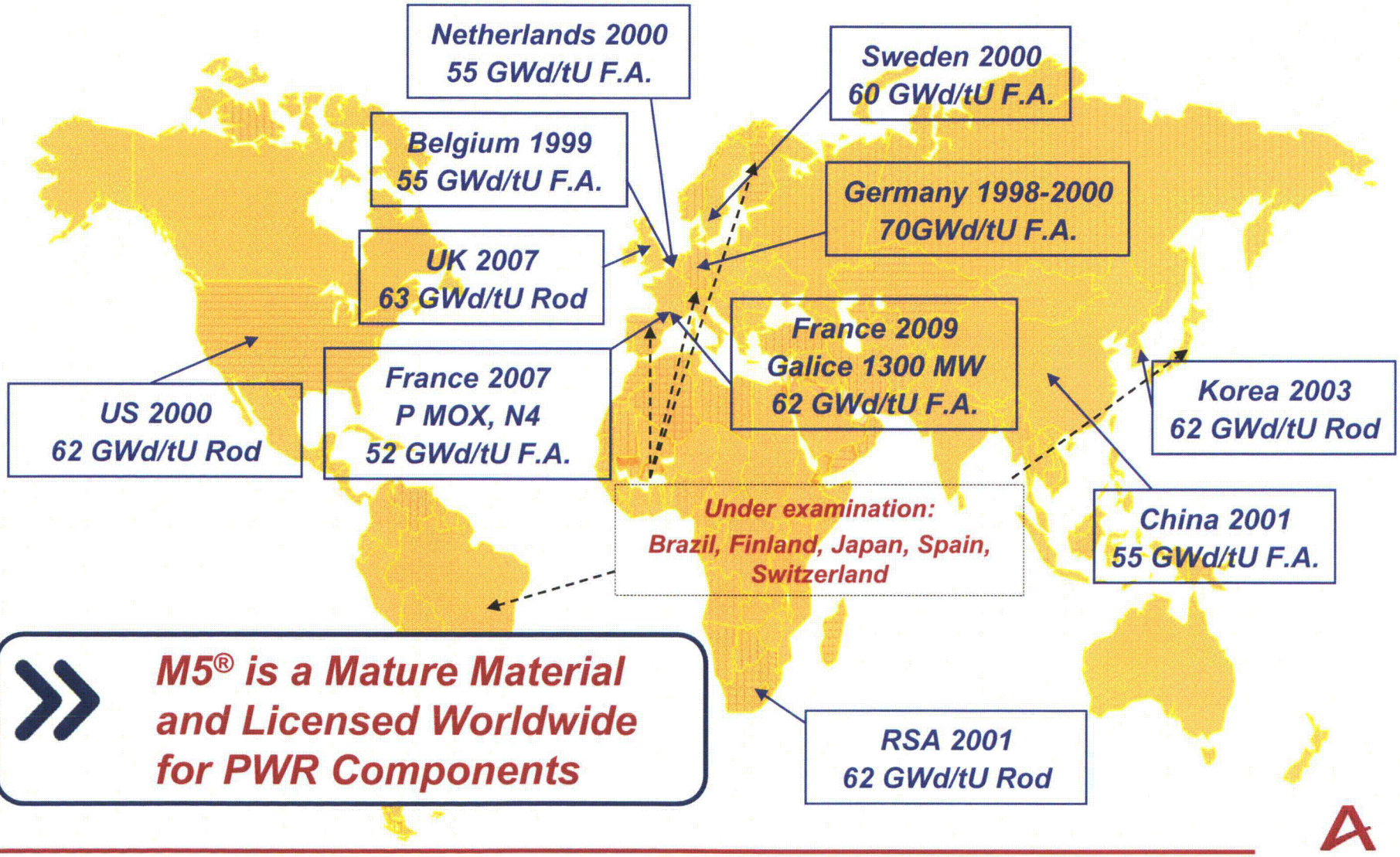
Licensing Status of M5®



M5® is a Mature Material and Licensed Worldwide for PWR Components

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M5® Implementation

First PWR Fuel Rod irradiations with M5 precursor (pre-licensing) alloys started in [ ] in Europe (3 plants) and continued in parallel in the US at [ ] 1 in 1991

Irradiation of M5 structural components has been performed since 1997

M5 cladding was fully industrialized for batch reloads by the end of 1999

First full reloads with M5 cladding and guide tubes were first introduced in 2001 in a US reactor

Fuel rod growth has totally conformed to established design limits

Fuel exams reveal that additional inspections were needed to bound assembly growth for M5® fuel designs

AREVA responded to first detections of issues (2007) via expansion of PIE programs

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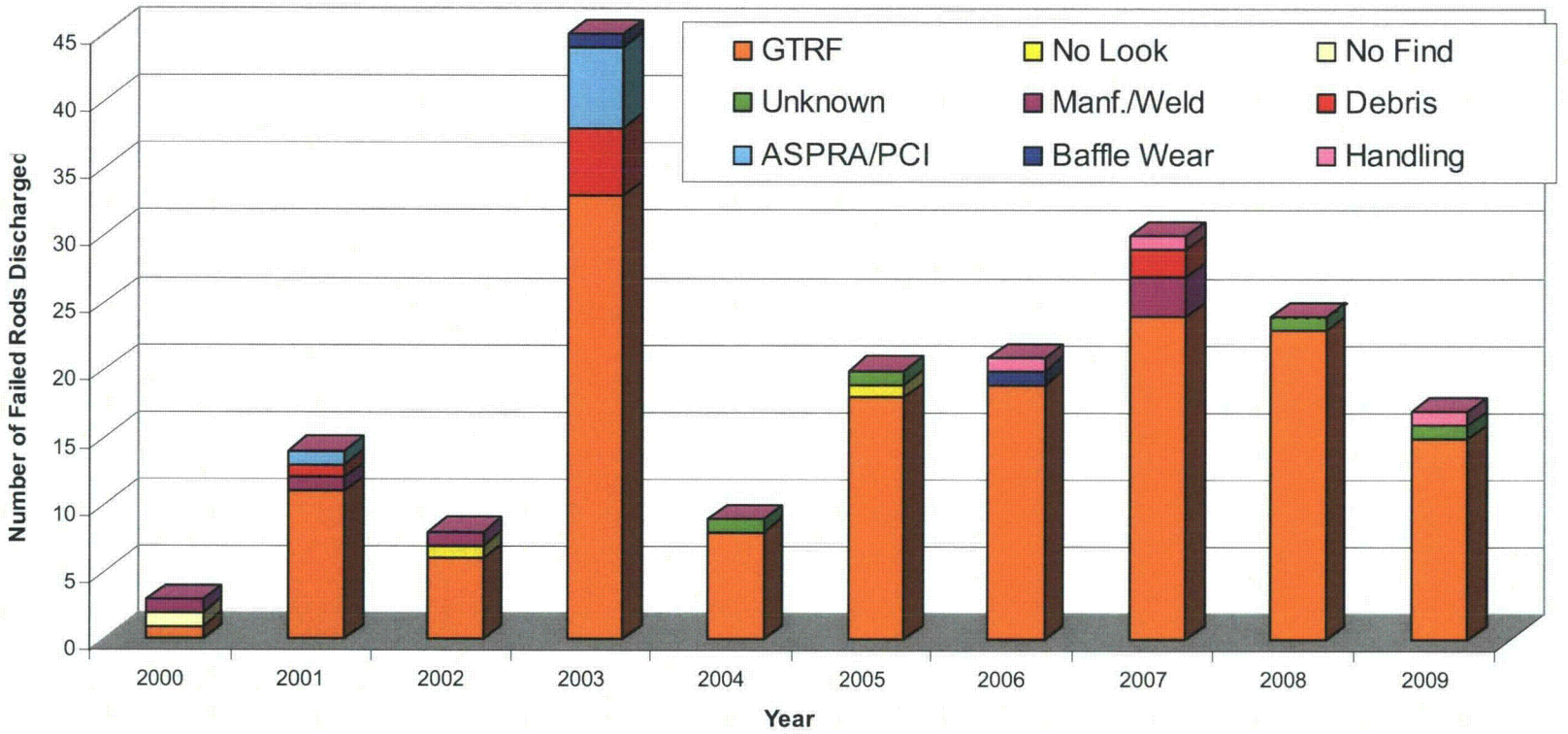
Plant History
PWR Fuel Manufactured In US



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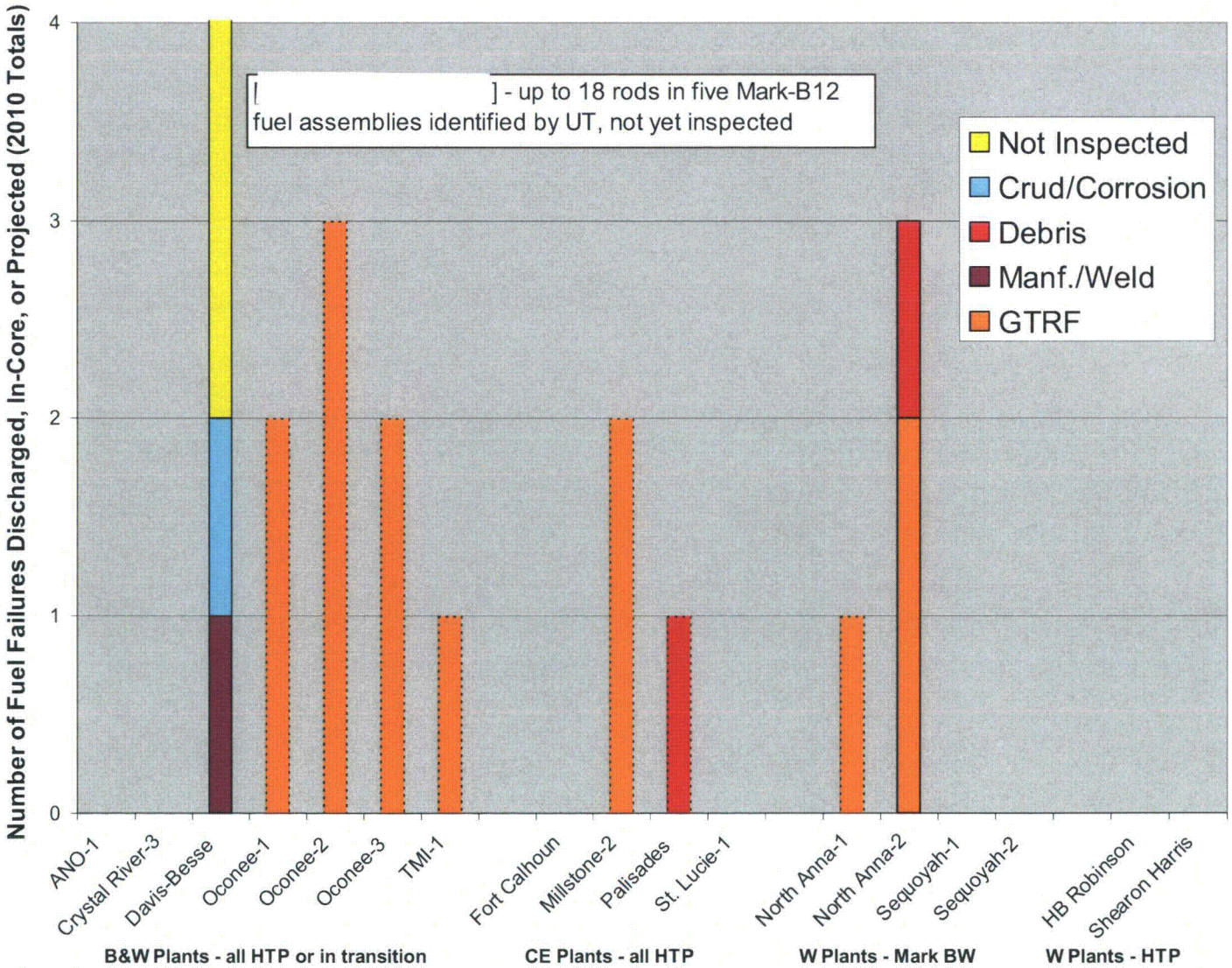
2010 Plant Status
PWR Fuel Manufactured In US



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2009 Achievements from OE

Elimination of spacer grid fretting failures

- **Reactors having successfully completed the transition**
  - St Lucie 1 – bi-metallic to HTP™
  - Fort Calhoun – CE to HTP™
  - Crystal River 3, ANO 1, Davis-Besse – Mark-B to Mark-B-HTP™
- **Reactors in transition**
  - TMI 1 – Mark-B to Mark-B-HTP™, complete 2012
  - Oconee 1, 2, & 3 – Mark-B11 to HTP™, complete 2012
  - Millstone 2 - HMP™ bottom grid, complete 2011

HTP is effective eliminating GTRF failures

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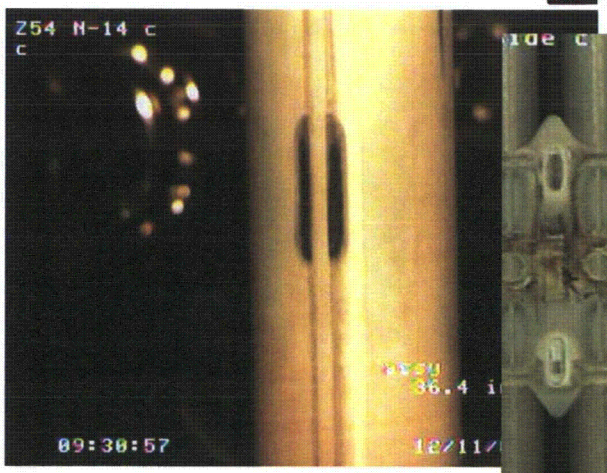
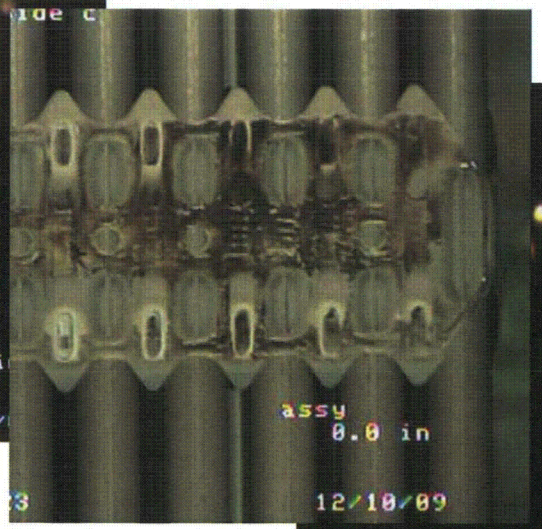
Baseline Fuel Inspection Results
Grid-to-Rod Fretting

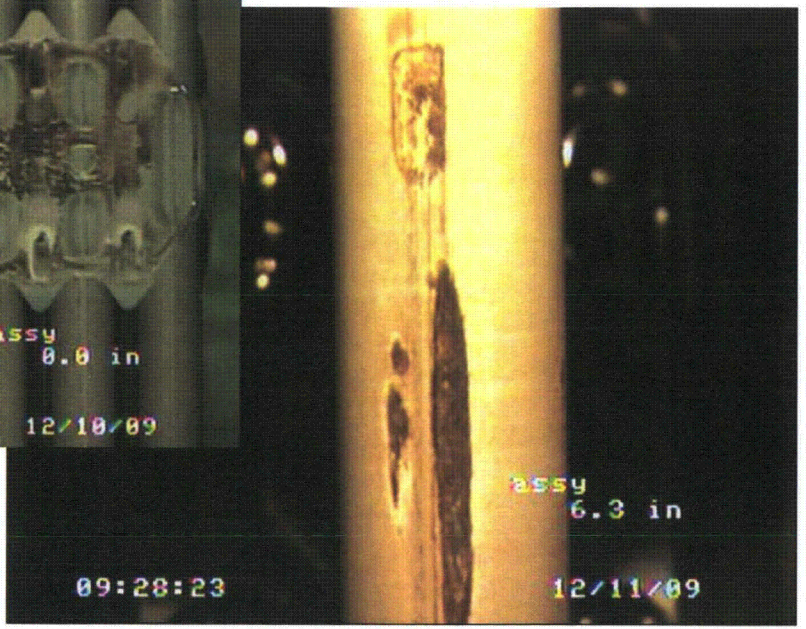
[ ]

- Four assemblies (3rd-burn, discharged), core periphery
- 74 total rods
- Range of wear depth: 0 to [ ]



1 mil (4%)





6.2 mils (22%)

Maximum Fuel Rod Wear In Presence of Minor Grid Wear – Large Structural Margin Exists

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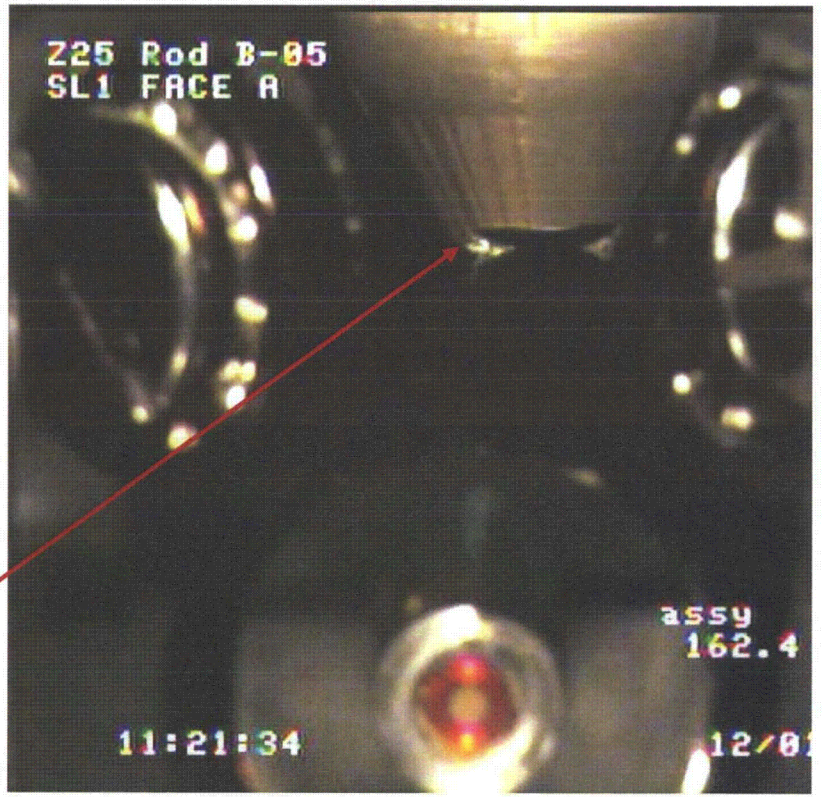
Baseline Fuel Inspection Results
Grid-to-Rod Fretting

[ ](CE-14 HTP™)

- Similar to Millstone 2
- End cap wear observed
- Indicates lateral rod vibration without spinning
- Continued confirmation of HTP™ fuel design against grid-to-rod fretting

[ ]



HMP™ end grid proven to be an effective counter rod failure mechanism in Millstone 2

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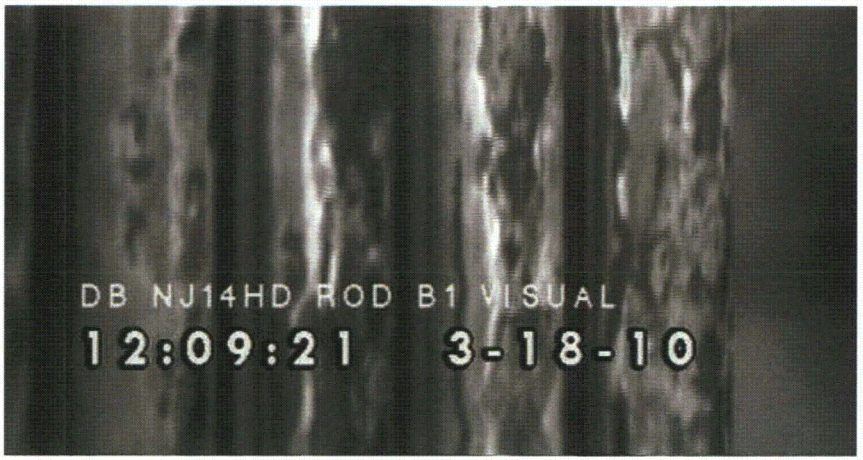
Davis-Besse Cycle 16 Fuel Failures

Manufacturing contamination: single first-cycle failure at startup

Crud-induced localized corrosion (CILC): single twice-burned failure

- Measured degradation (~20% through-wall) of neighboring fuel rods
- No degradation in reinsert fuel based on a population of bounding fuel rods
- Reducing crud inventory for Cycle 17 with ultrasonic cleaning and discharge of twice-burned fuel



Additional failures in third-burned Mark-B12 fuel

- Up to 18 failed/suspect rods identified by UT in five assemblies, not yet inspected

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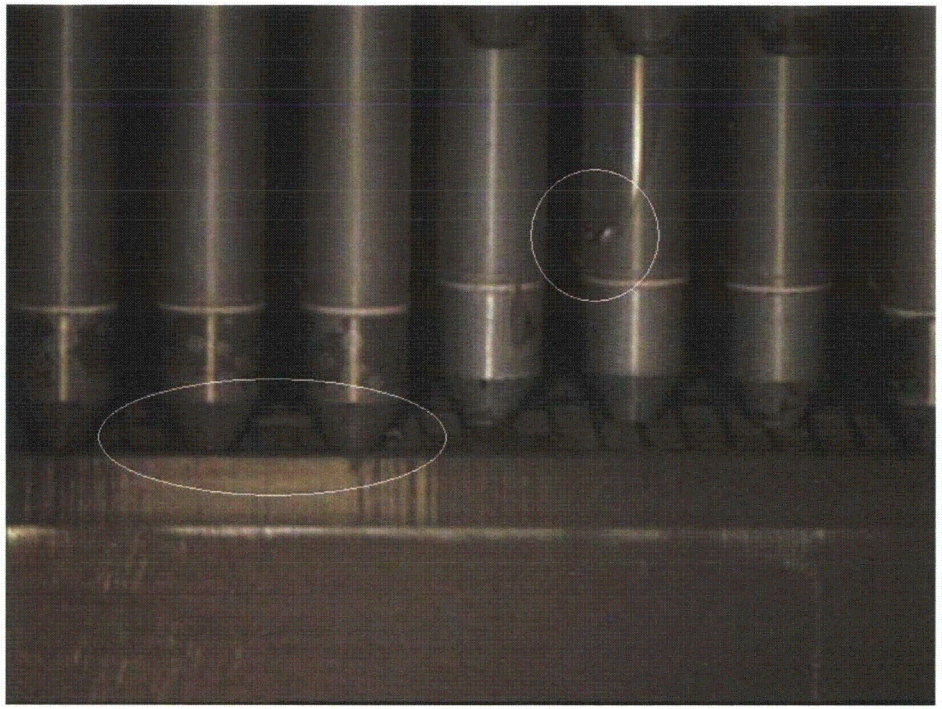
North Anna 2 Cycle 20 Fuel Failures

One twice-burned failure from debris



Two third-cycle discharge assemblies from the baffle

- Same rod location as Unit 1 last cycle (G-11)
- Suspected GTRF

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2010 Focus

Current failures other than Grid-to-Rod Fretting

- **Assessed as debris pending ACA after outage**
  - Palisades, out in Sep 2010
- **Common cause analysis looking at most recent failures**
  - Palisades RCA identified potential contamination
  - Corrective actions in place to exclude foreign material (MATCON)
- **Fuel fabricated prior to corrective actions**
  - Davis-Besse, one manufacturing and one corrosion
  - North Anna 2, one debris and two suspected GTRF

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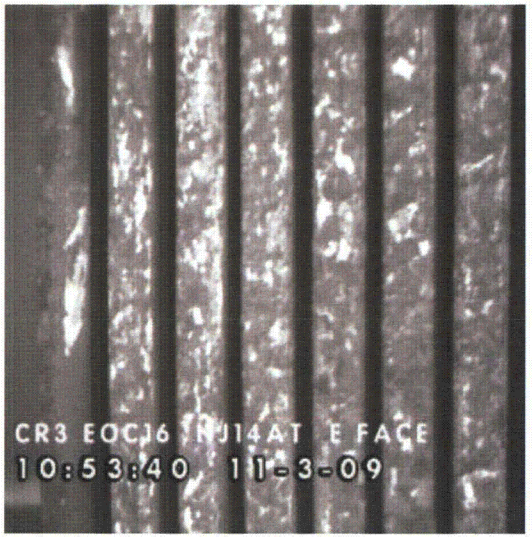
2010 Focus

Recent occurrences of elevated crud in PWRs

- Recent occurrences but not above licensing limit (D-B, TMI, CR-3, SLU, SHA)
- Emphasis is on improving crud prediction capability

[ ]
(crud + oxide, not brushed)



[ ]
(oxide, after brushing)

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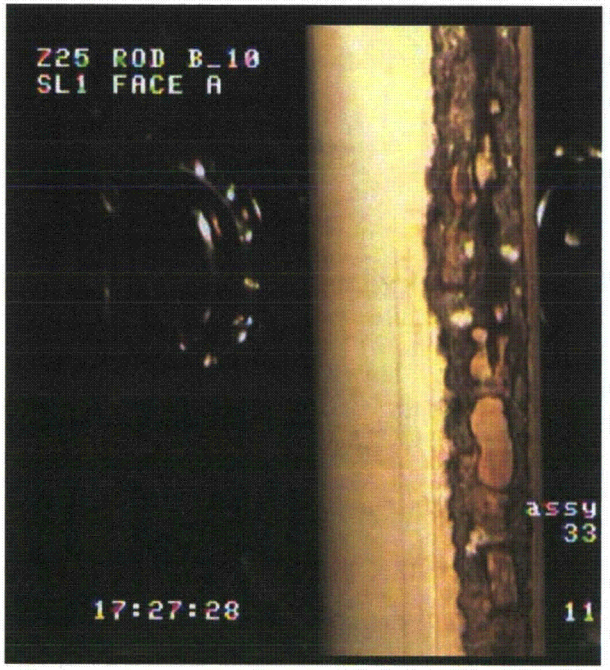
2010 Focus



[

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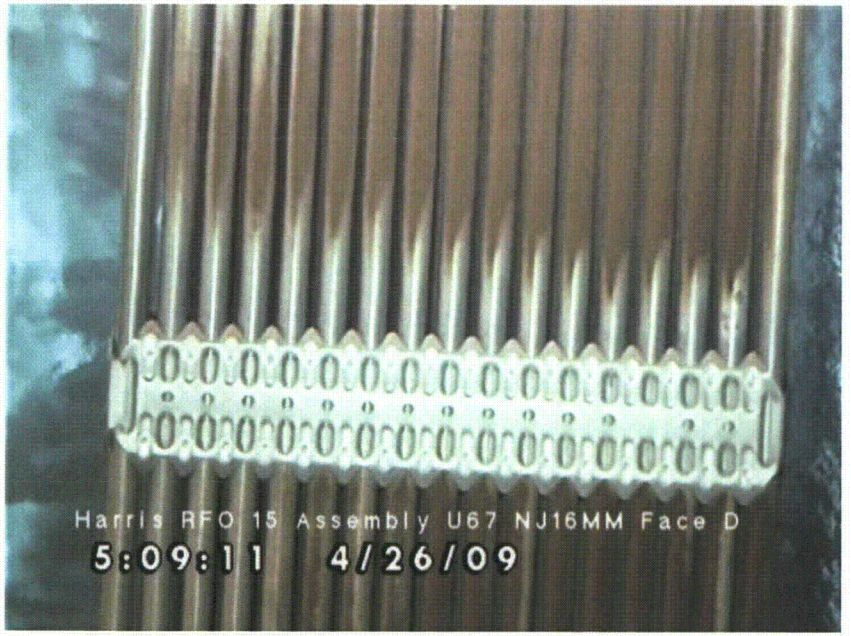
[Harris, W-HTP™
Elevated Li, 15-30µ, first-burn]



Level 2 PWR Assessment Tool in Place
Level 3, 4 PWR Capability in Place

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Summary

HTP™ continues to demonstrate successful performance against grid-to-rod fretting

Exceptional corrosion performance of M5® clad

Improvements to poolside measurement equipment and data analysis are underway to enhance crud evaluation techniques

The Goal To Eliminate Leaking Fuel Is In Sight And AREVA Has A Plan To Get There!

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BWR Operating Experience

Norman L. Garner
Technical Sales Manager – BWR Fuel

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BWR Operating Experience - Presentation Topics

Operating Experience

- ◇ Fuel rod failure
- ◇ Fuel channel distortion
- ◇ Lead product and material programs

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BWR Fuel Rod Reliability

Good progress toward Zero by 2010 objective for US plants with AREVA BWR fuel

- ◇ More than 36 months of failure free operations in US BWRs prior to June 2009
- ◇ No PCI/MPS failures in US fabricated fuel since 2004 pellet process improvements

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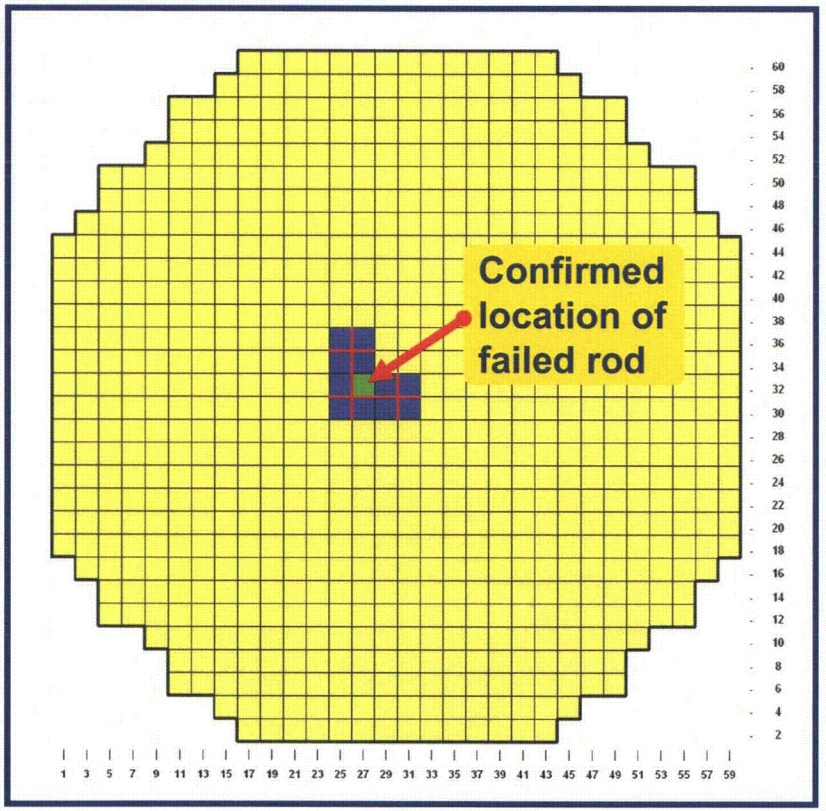
LaSalle 1 – Cycle 13
Status of Recently Discharged Failed Rod



Suspected single failed rod is only AREVA BWR rod failure in US in nearly 4 years – likely debris fretting

Activity increase observed ~15 months into 24-month cycle

- No significant power maneuvers immediately preceding activity

Power Suppression Test (PST) isolated probable location to cell adjacent to center cell

- Probable first cycle bundle based on response to PST
- 3 suppression blades inserted

No activity changes observed during sequence exchanges

- Cycle was able to run to planned EOC in 1Q10

Poolside exam – Summer 2010

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Control Blade Interference
Status of Fuel Channel Bow Issues

Observations of control blade interference are diminishing as volume of Zry-2 fuel channels in service declines

- ◇ Grand Gulf is the only US/Taiwan reactor with AREVA fuel currently reporting interference issues – 2 interior and 3 peripheral cells with high burnup AREVA Zry-2 fuel channels as of February 2010
- ◇ D-lattice: Surveillance testing at Browns Ferry showed no signs of settling issues in bundles with highest early control history

Zircaloy-4 fuel channels continue to show no susceptibility to elevated bow arising from high first cycle control history

Zircaloy-4 fuel channels are now reaching high burnups with significant numbers in high flux gradient positions

- ◇ Control blade surveillance maneuvers and poolside examinations are being used to assure no issues arise with break-away growth

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US In-reactor Qualification Programs

Two concurrent in-reactor qualification programs for advanced BWR materials are underway in the US to supplement extensive European programs

- ◇ AREVA is working with the host operator to effectively coordinate project planning for the initial poolside PIE campaigns following first cycle of irradiation in 2011

Insertion of β-Quenched Zry-BWR fuel channels planned for US in 2011 to supplement EU experience

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Global Progress of In-Reactor Advanced Material & Process Qualification Programs

Zry-BWR Fuel Channels

- A concurrent program of Zry-BWR lead fuel channel irradiation is underway in Europe
- Goals: Low uniform oxidation and low hydrogen uptake

β-Quenched Fuel Channels

- Irradiation programs in three European BWRs (to 56.5 GWd/MTU)
- Data shows desired low axial growth and low bow, even in cases of high early control history, without impact on corrosion resistance
- Goal: Industrial scale process to maximize dimensional stability

Chromia-doped Fuel Pellets

- A number of rods have been irradiated in European PWRs and BWRs
- Additional ramp testing scheduled in 2010 for BWR rods
- Goals: Enhanced PCI margin and reduced fission gas release
- Bonus: Greater chipping resistance and reduced steam degradation

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BWR Fuel Performance Conclusions

Improved pellet quality and refined maneuvering guidelines are yielding good results in driving PCI failures to zero

Channel management tools and declining Zry-2 inventory are yielding diminishing incidents of control blade interference

Availability of the Improved FUELGUARD™ directly addresses the key recurrent fuel failure mode for BWR fuel

Qualification of Zry-BWR fuel channels, β-Quenched fuel channels, and chromia-doped fuel pellets will yield further enhancements in fuel reliability in the next few years

» *Product improvements implemented over the past few years are directly enhancing fuel reliability*

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PWR Fuel Exams

John H. Strumpell
Manager, PWR Material and Thermal-Mechanics

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NRC Fuel Performance Update

PWR Program Objectives

Test Assembly Programs

Poolside PIEs

Experience Summary

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PWR Exam Objectives

Purpose of the latest PIE data is to specifically support the following fuel designs:

- ◇ **Mark-B (15x15, B & W Reactor)**
- ◇ **Mark-BW (17x17, W Reactor)**
- ◇ **W-HTP (15x15, 17x17, W Reactor)**
- ◇ **CE-HTP (14x14, 15x15, 16x16, CE Reactor)**

Comprehensive plan developed by product design

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PWR Test Assembly Programs

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PWR Test Assembly Programs

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PWR Test Assembly Programs

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PWR Test Assembly Programs

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PWR Test Assembly Programs

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2009 Poolside PIEs

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2009 Poolside PIEs
(Continued)

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2010 Poolside PIEs

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M5 Experience Summary

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M5 Experience Summary
(continued)

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M5 Experience Summary
(continued)

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M5 Experience Summary
(continued)

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M5 Experience Summary
(continued)

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M5 Experience Summary
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M5 Experience Summary
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M5 Experience Summary
(continued)

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M5 Experience Summary
(continued)

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M5 Experience Summary
(continued)

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Zircaloy-4 Experience Summary

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Zircaloy-4 Experience Summary (continued)

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Zircaloy-4 Experience Summary
(continued)

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Zircaloy-4 Experience Summary (continued)

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BWR Fuel Exams

Norman L. Garner
Technical Sales Manager – BWR Fuel

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BWR Fuel Exams - Presentation Topics

Fuel Examinations

- ◇ Recent exams
- ◇ Planned exams

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BWR Fuel Exams - General Programs

Absence of fuel rod failures has precluded need for failed fuel exams at US BWRs since mid-2006

Reduced incidence of abnormal fuel channel bow has slowed the need for cause investigation and remediation campaigns

- Measurement programs are continuing to keep close track on Zry-4 fuel channel performance as fleet inventory starts to reach high burnups

Focus has been shifting to supporting general surveillance goals, with several basic characteristics of interest:

- Fuel rod baseline corrosion behavior under various chemistry regimes
- Fuel rod profilometry
- Fuel assembly and fuel rod growth

Aligned with INPO Zero by 2010 and surveillance initiatives

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BWR Fuel Exams - Special Topics of Interest

Surveillance of lead products and new material programs

- ◇ Zry-BWR fuel channel corrosion and dimensional behavior
- ◇ Cladding behavior and fission gas release in Cr-doped fuel rods

Surveillance of non-standard fissile material

- ◇ Continued tracking of off-spec BLEU uranium in use at TVA reactors

Expanding reference data for crud structure and constituents

- ◇ Coordinated effort with rapidly evolving crud prediction tools

Surveillance of shadow corrosion on fuel rods

- ◇ Baseline data for ranking sensitivity of US plants against global data

Expanded reference data for plants with elevated Zn injection and online noble metals injection

- ◇ Tracking any changes in fuel performance as chemistry evolves

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2010-2011 BWR US PIE Plan

Failed fuel rod examination at LaSalle

INPO baseline inspections (crud / corrosion) at each site with current AREVA reload supply

ATRIUM™ 10XM performance validation and chromia-doped pellet behavior

Zr-4 and Zr-BWR channel performance

Cladding performance validation (length, diameter, oxide)

BLEU evaluation for mechanical performance and to confirm neutronics depletion model

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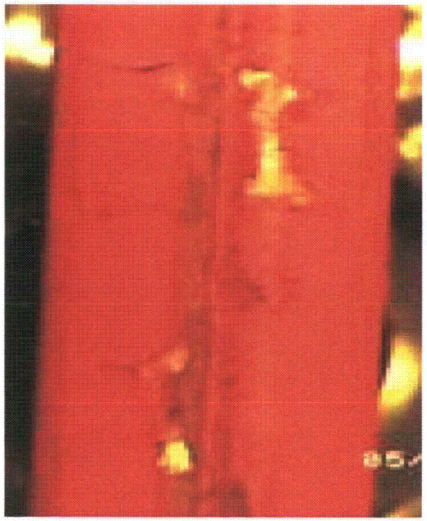
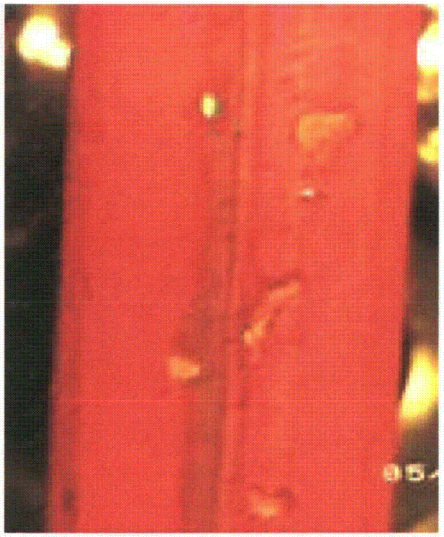
BWR Crud Evaluations

No cladding degradation has been observed in recent years

In one case, operation penalties were imposed

Emphasis is on understanding Zn impact on measurements





BWR crud assessment tool in place using crud scrapes for verification

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Recent BWR Fuel Exam Results
BWR Fuel Rod Corrosion

[ ]

- *Elevated crud / corrosion liftoff data in high Zn environment challenged historic limits*

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Summary

Success in eliminating fuel rod failures and fuel channel distortion is freeing resources to focus on healthy fuel baseline surveillance

Comprehensive 5-year PIE plan in place to gather and assess baseline data going forward

- ◇ Key to validating and improving performance and understanding failure modes (crud, debris, etc.)
- ◇ Tracking effects of changing chemistry on fuel performance

Poolside and hot cell data programs are coordinated with R&D efforts to build accurate prediction models

- ◇ Supports sophisticated tools to model complex interactions of fuel duty and coolant chemistry regimes

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M5 Guide Tube Growth

Brian V. Haibach
Supervisor, Fuel Mechanics

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M5 Fuel Assembly (Guide Tube) Growth

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Status Update

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Status Update (continued)

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Status of Long Term Actions

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MOGADOR
Fuel Assembly Growth Predictive Tool

Fuel assembly growth limits are empirically based requiring wide data bases covering different designs and operating conditions with limited capability of prediction in case of changes (design, material, operating conditions...)

MOGADOR, a fuel assembly growth predictive tool, continues to be used and further advanced to support robustness of calculated growth limits defined through parametric evaluations

Models guide tube dimensional changes attributed to two main mechanisms

- Irradiation-induced free growth
- Irradiation-induced creep

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Fuel Assembly Growth Prediction Ongoing R&D Actions

Define grid slip loads evolution under hot irradiated conditions:

- ◇ Addressing following aspects: temperature, fluence, sliding speed, surface condition changes due to irradiation, vibrations, fuel assembly bow, fuel rod bow

Grid cell modeling and analysis of fuel rod extraction loads performed in spent fuel pool

Friction experiments in laboratory

- ◇ ALFA and FANI experiments on grid cell components at Le Creusot

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Advanced Modeling of Zirconium Alloy Guide Thimble Response

Overview of continuous multi-year R&D programs to increase our basic knowledge of zirconium alloys dimensional changes

◇ **Main objectives are :**

1. Refining the accuracy of material behavior laws
2. Allow increased accuracy in predicting material response under complex conditions

◇ **Hot Cell Examinations on Guide Tube material (C-loop and hydrogen)**

- North Anna high burnup M5®
- Three Mile Island high fuel assembly growth
- Europe high, lower and upper band growth
- Crystal River Mark-B-HTP™

◇ **Specific analytical programs (including hydrogen and stress impact)**

- Proton/Ion Irradiation (feasibility first, and then hydrogen and stress effect)
- Fast neutrons Irradiation (hydrogen and stress effect)

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PWR Fuel Assembly Distortion Update

Brian V. Haibach
Supervisor, Fuel Mechanics

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Agenda

Describe events and current observations/data

Action Plan Overview/Status

Fuel Assembly Bow Modeling Activity

Summary

Q&A

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Fuel Assembly Distortion Observations

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Fuel Assembly Bow Database

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Fuel Assembly Distortion at Sequoyah-1 EOC16 (1/2)

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Fuel Assembly Distortion at Sequoyah-1 EOC16 (2/2)

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BOC to EOC Drag Increase – SEQ1 Cycle 16

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BOC to EOC Drag Increase – SEQ2 Cycle 14, 15, 16

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Full-Core EOC Drag Measurements at SEQ-2 EOC16

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Actions to Solve Bow Issue at Sequoyah

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Guidelines for SEQ Core Design to Prevent Slow-to-Settle

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Mark-BW Fuel Assembly Bow Modeling

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Fuel Assembly Distortion

FSI Model with Channel Network Approach

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Fuel Assembly Distortion Hydraulic Tests

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Propagation of Initial Bow During Several Cycles

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Conclusions

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PWR Fuel Rod Bow

Brian V. Haibach
Supervisor, Fuel Mechanics

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Meeting Agenda

Describe events and current status

Action Plan Overview/Status

Water channel database

Analysis of Influencing Parameters

Summary

Q&A

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Description of Fuel Rod Bow, Observations and Recent Events

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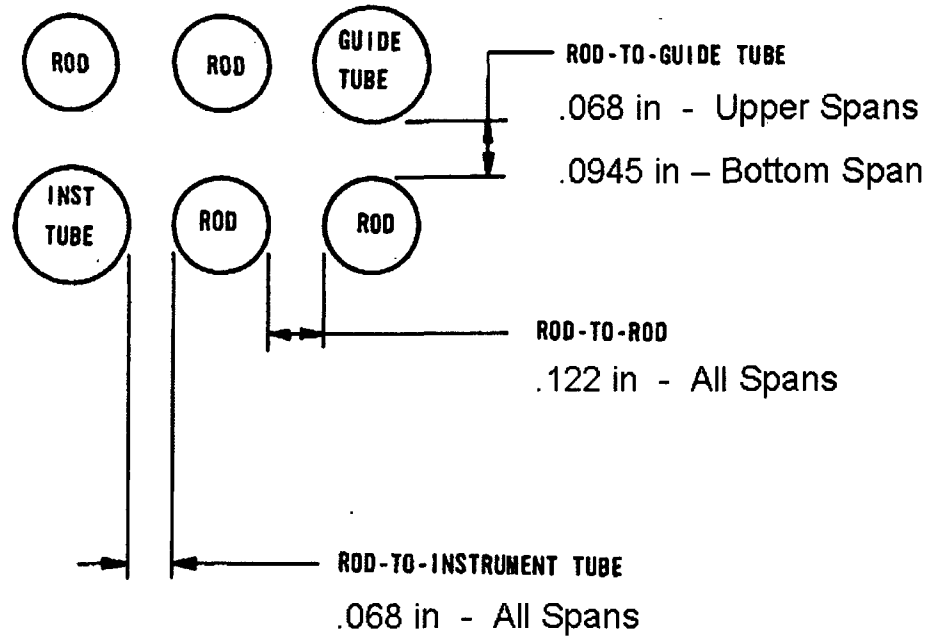
Rod Bow

Alters design pitch dimensions, or water channel gap, between adjacent rods within an Advanced Mark-BW fuel assembly array which affects:

Local nuclear power peaking

Local heat transfer to coolant



Consequence ➔ local excessive over-heating of fuel rod cladding which may lead to failure

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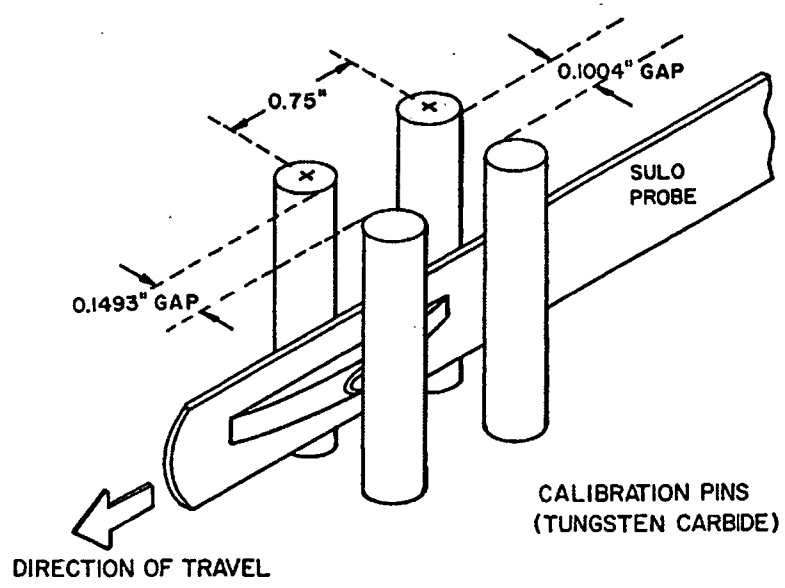
Rod Bow

Water channel gaps are measured with a Sulo probe.

- ***The probe is positioned between two fuel rods and inserted into the fuel assembly***
- ***Each gap along this path is measured by the probe.***

The standard deviation of the gaps from one span is the measured rod bow.



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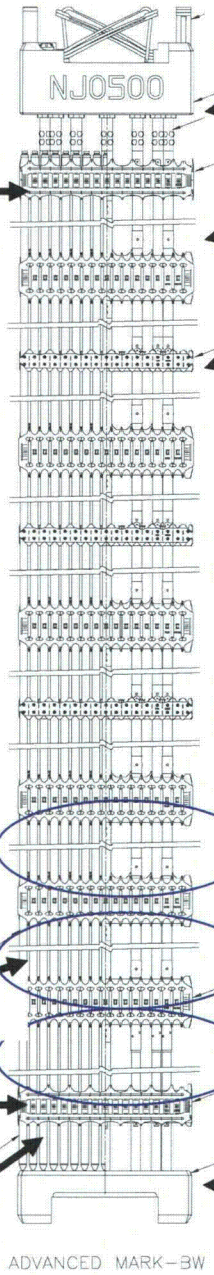
Advanced Mark-BW Fuel Assembly



3 Leaf Holddown Spring Assembly (Inconel)

Fixed Top End Grid (Inconel)

5 Floating Vaned Intermediate Grids M5™

M5™ Fuel Rod Cladding

Fixed Bottom End Grid (Inconel)

Seated Fuel Rods

Top Nozzle

M5™ Guide Thimbles

3 Mid-Span Mixing Grids M5™

Lower 3 spans observed with elevated rod bow

Floating Vaneless Intermediate Grid M5™

Coarse Mesh Debris Filter Bottom Nozzle

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Recent Observations

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Action Plan Overview/Status

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Actions Completed

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Fuel Rod Bow Experience U.S. Database

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Analysis of Influencing Parameters

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Fuel Rod Bow Modeling

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Fuel Rod Bow Modeling

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Evaluation of Fuel Rod Bow Behavior Moving Forward

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Summary

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PWR Cruciform Holddown Spring Performance Update

Brian V. Haibach
Supervisor, Fuel Mechanics

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Agenda

Previous Event

Second Observation

Status Update

Root Cause Analysis

Planned Activities

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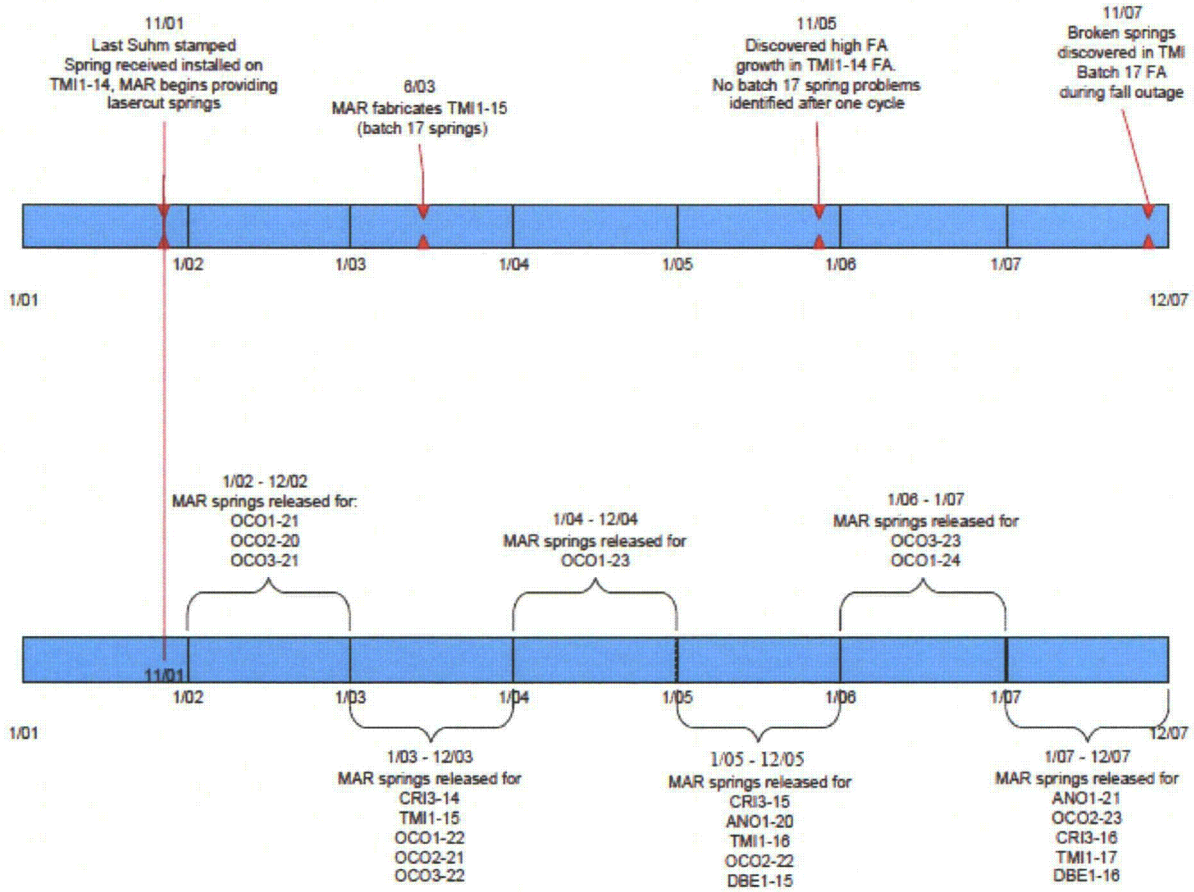
Previous Event - Timeline

Design Timeline



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Previous Event Summary

Following TMI-1 Observations in 2007

- ◇ RCA team concluded that spring failures caused by surface condition created due to manufacturing process change
- ◇ RCA team found extent of condition was generic for cruciform springs using laser cutting process leaving recast layer on surface
- ◇ After RCA team disbanded, un-machined recast layer was found on production springs using the previous process which supported RCA team conclusions

Manufacturing Changes

- ◇ Improvements were implemented in 2008
  - More sacrificial material on as-cut spring
  - Improved 2-pass machining operation to ensure recast layer removed

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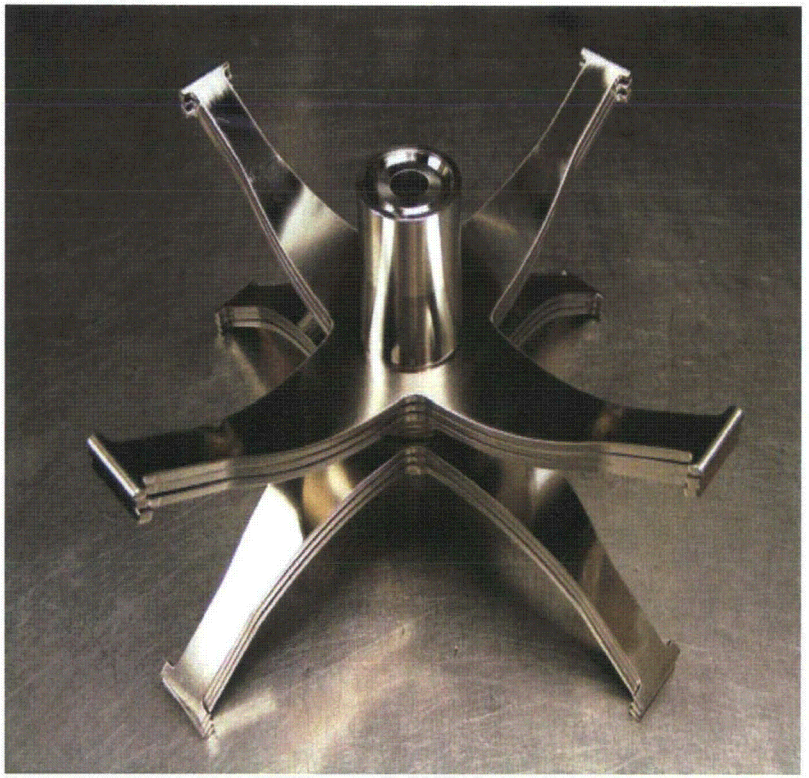
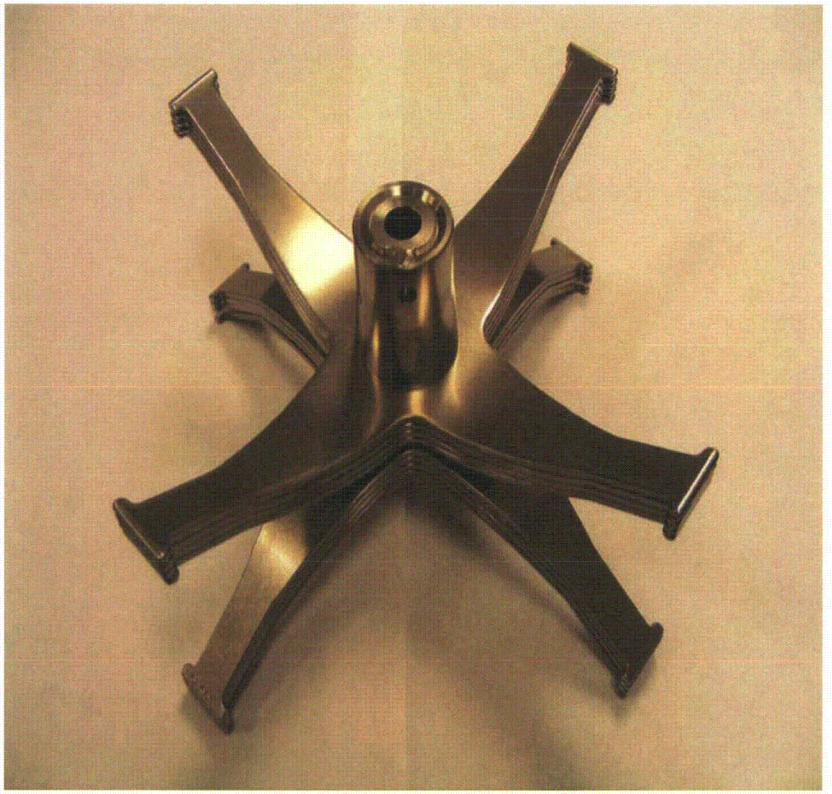
Issue Summary – Cruciform Spring



Current 6-Leaf Design



Initial Stamped 8-Leaf Design

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Previous Event Damage – at TMI EOC16

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Issue Summary – Second Observation

Crystal River Unit 3 End-of-cycle 16 Refueling Outage

- ◇ 19 assemblies with no holddown spring preload (out of 50 measured)
- ◇ 8 assemblies have visual indication of crack or damage to HD springs
- ◇ All Mark-B-HTP fuel assemblies

Follow-up Inspections at Oconee

- ◇ All Mark-B11 assemblies with functional springs - 62 assemblies inspected
- ◇ OCO1, and OCO2 preload test on 10 assemblies each (50 to 54 burnup)
- ◇ OCO3 preload test on 42 assemblies, visual inspection of 20 assemblies (38 to 55 burnup)

Subsequent at Three Mile Island Unit 1 EOC 17 Refueling Outage

- ◇ 3 Mark-B12 assemblies with no preload (out of 29 measured)
- ◇ 1 Mark-B12 with replaced upper end fitting had small preload

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Status Update

Immediate Actions (information provided to NRC)

- ◇ **Performed operability assessments for fuel in core**
- ◇ **Initiated inspections at TMI and OCO**
- ◇ **Developed recovery plan for impacted sites**
  - Remove springs made using old process from service
  - Replacement Upper End Fittings for Re-insert Mark-B-HTP and Mark-B12
  - Based on inspection results, no replacements for Mark-B11 at OCO

Current Status

- ◇ **All reinsert fuel has had springs replaced during outages (excludes OCO)**
- ◇ **Once burned springs with improved manufacturing process available for inspection at ANO-1 and OCO-2**
  - No pre-load or spring rate data taken at ANO due to outage limitations
  - ANO first burn spring video is being reviewed

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Crack Indication in Holddown Spring

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Inspections – Status
D-B Visual EOC 16

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Inspections – Status
D-B Visual EOC 16

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RCA Charter (CR 2009-7073)

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Planned RCA Activities

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Inspections - Status

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Inspections - Status

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Inspections – Status

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Inspections – Status

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Plant Water Chemistry

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Plant Water Chemistry (continued)

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Stress Corrosion Cracking - Testing

SCC Requires

- ◇ **Environment**
- ◇ **Tensile Stress**
- ◇ **Susceptible Material**

Test Setup

- ◇ **'Nominal-Aggressive' Environment**
  - Based on plant water chemistry survey
- ◇ **Loading to maximum predicted deflection during normal operations**
  - Held static during test
- ◇ **Samples Being Tested**
  - Spring Pack from Current Manufacturing Process
  - Stamped Spring Pack
  - Mixture Pack (TMI Retain, Visible Recast Layer, Intentional Damage)
  - Alloy 600 Control Sample

Samples Examined after 1000 and 2000 hours

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Stress Corrosion Cracking - Testing

1000 Hour Exposure – Just completed, under evaluation

- ◇ Visual inspections
- ◇ Force deflection test
- ◇ Metallurgical Inspections

2000 Hour Exposure – begin ~ end of May 2010

Expectations from testing:

- ◇ Establish robustness of different spring manufacturing processes.
- ◇ Is SCC the correct failure mechanism?

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Residual Stress Analysis Test

Test Samples

- ◇ **Current manufacturing process**
  - Determine variability among 4 legs
- ◇ **Current manufacturing process**
  - Before and after undergoing typical 'set'
- ◇ **Previous manufacturing process**
  - Measurements at visible recast layer locations

Expectations:

- ◇ **Understand spring stress state before operation**
- ◇ **Determine residual stress impact of recast layer**
- ◇ **Data to feed FEA models**

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Fuel Thermal Conductivity

John H. Strumpell
Manager, PWR Material and Thermal-Mechanics

Yusen Qi
Fuel Performance Analyst

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Presentation Contents

Background Material

Mechanism

Impact Assessment

Current Activity

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Background Material

Fuel thermal conductivity – one important parameter to predict the fuel rod performance

Generally, modeled as

$$\lambda = \lambda_{\text{phonon}}(t) + \lambda_{\text{electronic}}(t)$$

Low temperature - lattice phonon term

High temperature - electronic term

Thermal conductivity for 100% dense fuel is adjusted by the porosity factor

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Background Material

Example: TACO3 thermal conductivity model:

[ ]

[ ]

[ ]

[ ]

T = fuel temperature, K

λ = fuel thermal conductivity (100% dense), $W/(m \cdot K)$

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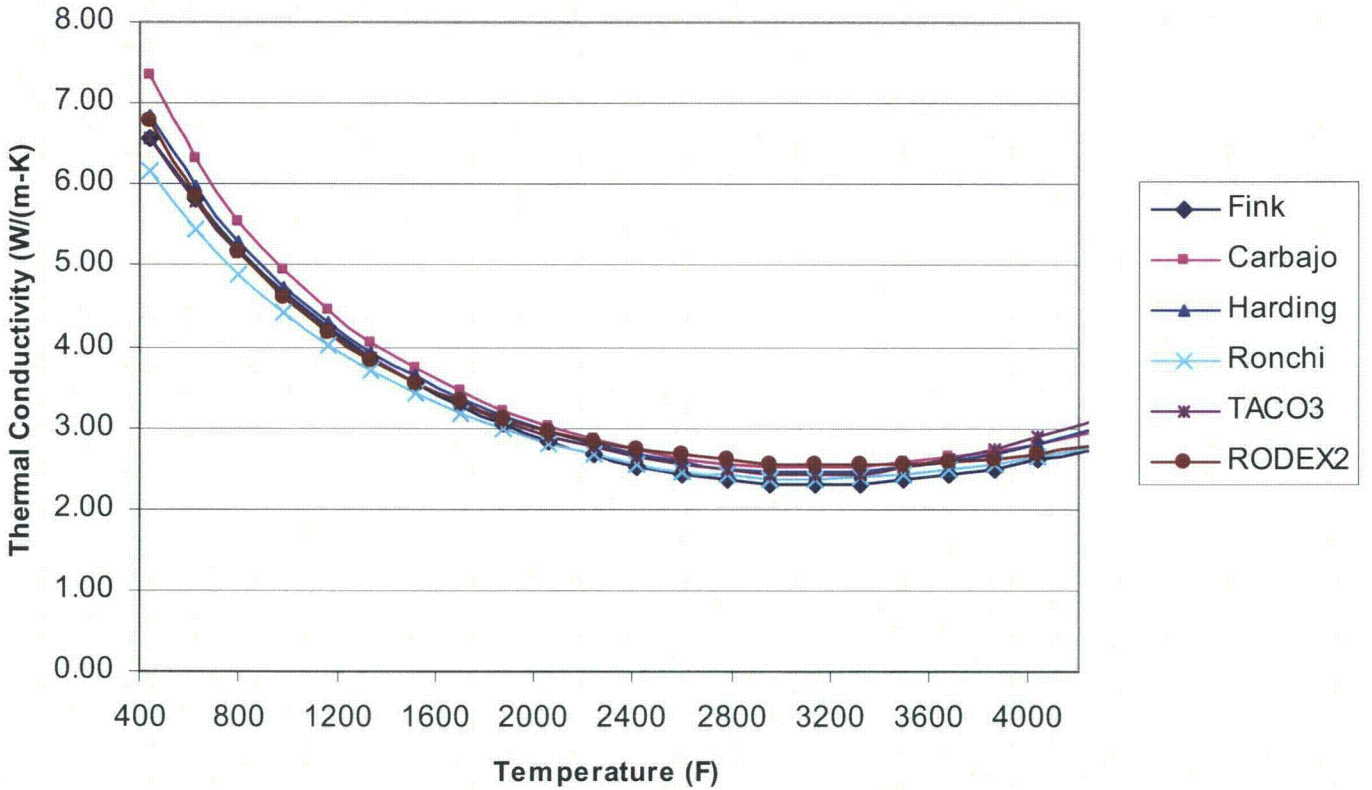
Background Material

Comparison for un-irradiated 100% dense UO_2 thermal conductivity models



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Background Material

Carbajo et al. [* states that "the Fink correlation clearly provides the to the data"

TACO3 thermal conductivity model used a quadratic representation for the lattice phonon term, as Fink's model does

TACO3 thermal conductivity model agrees with Fink et al.'s model very well

RODEX2/2A values is similar to TACO3's

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Mechanism

Well established in literature that fuel thermal conductivity decreases with burnup

Causes:

- **Fission Product Accumulation**
- **Irradiation Damage**

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Mechanism

In-reactor thermal conductivity data collection is complex

The formation and annealing of irradiation damage effects, therefore, was based on a relatively small data set

Current formulations based on empirical data

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Mechanism

Evolution of Fuel Performance Codes

Legacy codes

TACO3/GDTACO (1989/1993)

- For PWR
- Best-estimate predictions with uncertainties applied where appropriate
- Thermal conductivity for 100% dense fuel does not explicitly consider the irradiation effect

RODEX 2/2A (1984/1986)

- For both PWR and BWR
- Thermal conductivity for 100% dense fuel does not explicitly consider the irradiation effect

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Mechanism

Evolution of Fuel Performance Codes

Present codes

COPERNIC (2002)

- PWR thermal mechanical code
- Fully account for thermal conductivity decrease with burnup

RODEX4 (2008)

- BWR thermal mechanical code
- Fully account for thermal conductivity decrease with burnup

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Impact Assessment

SIMFUEL – Simulated high-burnup UO_2 fuel (1992)

The reported thermal conductivity of SIMFUEL is lower than that measured for UO_2

The effect of thermal conductivity degradation using SIMFUEL data was accounted

Appropriate uncertainty was applied to analyses after studying SIMFUEL data

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Impact Assessment

US NRC Information Notice (IN 2009-23, Oct. 8, 2009) requests fuel vendors to review the codes that were licensed before 1999

Analyses potentially impacted, such as:

- End of life fuel rod internal gas pressure
- Centerline fuel melt (CFM)
- Transient cladding strain (TCS)
- Loss of coolant accident (LOCA) fuel temperature initializations

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Impact Assessment

AREVA NP corrective action program includes the assessment on RODEX 2/2A, TACO3/GDTACO

Part 21 evaluations for these codes were performed

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Impact Assessment

Used modern codes to calculate the adjustment to be applied wherever appropriate

Example: assessment on TACO3 follows

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Impact Assessment

Rod Internal Gas Pressure

- ◇ Methodology imposes six condition I transients throughout the burnup range
- ◇ Conservatism in power history
- ◇ Conservative bounding value exists

Centerline Fuel Melt

- ◇ Determine a linear heat rate limit at which fuel melt begins
- ◇ Using COPERNIC to calculate the centerline temperature adjustment
- ◇ No need to modify the reactor trip setpoints

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Impact Assessment

Transient Cladding Strain

- ◇ Temperature increase affects fuel-clad contact and strain
- ◇ COPERNIC derived adjustment was applied to the TACO3 results
- ◇ Reactor operating limit was not affected

LOCA Initialization

- ◇ Temperature uncertainty is adjusted at high burnup
- ◇ Sufficient xenon gas is generated at high LHGRs with 1000 MWd/mtU hold to degrade the gap conductance and effectively account for the lack of a fuel thermal conductivity degradation model
- ◇ LOCA analysis fuel temperature input remain appropriate

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Impact Assessment

Conclusion

- ◇ Thermal conductivity models in the legacy codes do not explicitly account for irradiation effect
- ◇ The effect of burnup on thermal conductivity was considered previously
- ◇ Each fuel rod analysis application has unique methodology with different conservatism
- ◇ Adjustments have been applied on the affected applications
- ◇ Adjustments are absorbed by the existing margin
- ◇ No defect was concluded

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Current Activity

Interim approach is to apply adjustments to TACO3 and RODEX2

COPERNIC and RODEX4 are approved newer generation codes

- ◇Irradiation impact on thermal conductivity addressed

AREVA has initiated application studies directed at implementing COPERNIC and RODEX4

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Crud Risk Management Tools (Thermal-Hydraulic)

Christopher J. Lewis
Manager, PWR Thermal-Hydraulics

AREVA

Agenda

Current Status of Crud in AREVA Designed Cores

Status of AREVA's PWR TH Crud Risk Assessment Tools

Future Development Plans?

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Current Status of Crud in AREVA Cores

Contemporary PWR fuel failures in USA attributed to crud/corrosion

TMI-1 During Cycle 10 (EOC 1995) ←AREVA Fuel
Seabrook Cycle 5 (EOC 1997)
Palo Verde-2 Cycle 9 (EOC 2000)
Calvert Cliffs-1 Cycle 17 (EOC 2006)

Significant DCP indications noted in:

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Current Status of Crud in AREVA Cores

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Status of AREVA's PWR Crud Risk Assessment Tools

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Subchannel code enhancements in LYNXT for use in L2 and L2.5 analyses

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Status of AREVA's PWR Crud Risk Assessment Tools

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Status of AREVA's PWR Crud Risk Assessment Tools

Level 3 analyses

[ ]

- ◇ **Results from analysis used to direct changes to core loading strategy, i.e.**

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Status of AREVA's PWR Crud Risk Assessment Tools

Level 4 analyses

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Future Development Plans

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Crud Risk Assessment Tools

BACKUP Slides

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Crud Risk Assessment Tools - Application

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Crud Risk Assessment Tools - Application

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Crud Risk Assessment Tools - Application

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CFD Results for Full Span 3

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Crud Risk Assessment Tools - Application

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Closing

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