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July 12, 2010 BW-JAH-2010-219

U.S. Nuclear Regulatory Commission (NRC)
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Babcock & Wilcox Company, Nuclear Energy
Docket No. PROJ0776
Project No. 776

Subject: Submittal of Presentations for Planned July 29, 2010 Meeting with NRC Staff(NP)

On July 9, 2010, Babcock & Wilcox Nuclear Energy (B&W NE) transmitted two sets of slides to the NRC for planned presentations at a closed meeting with NRC staff to be held on the 29th of July. The slides contained B&W NE Confidential Commercial Information that was requested to be withheld from public disclosure. At that time, B&W NE committed to provide a non-proprietary version of the presentations under a separate letter.

Accordingly, enclosed is a non-proprietary version of the slides.

Questions concerning this submittal may be directed to T. J. Kim at 434-382-9791 (email: tjkim@babcock.com) or to J. A. Halfinger at 434-316-7507 (email: jahalfinger@babcock.com).

Jeffrey A. Halfinger
Vice President, B&W mPower Development
B&W NE

JAH/

cc: Joelle L. Starefos, NRC, TWFN-9-F-27
Stewart L. Magruder, Jr., NRC, TWFN-9-F-27

4601
NRC

Add: J. L. Starefos
S.L. Magruder

[CCI per Affidavit 4(a)-(d)]

B&W

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B&W *mPower*™ Core Design

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Core DesignTopics

- Mechanical Design
- Nuclear Design
- Thermal-Hydraulic Design

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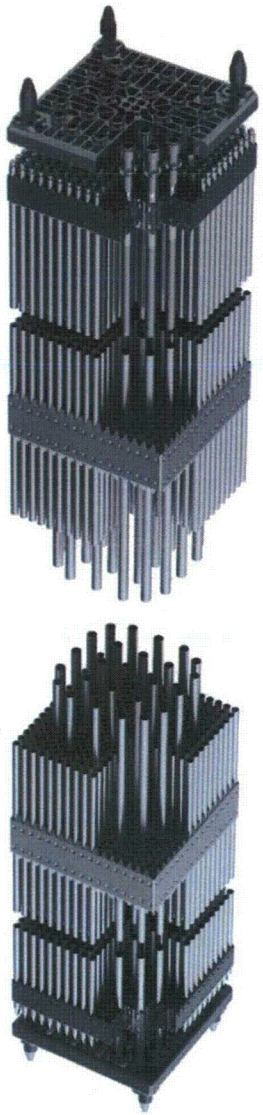

B&W mPower™ Core Mechanical Design Features

Eric Barringer
Core Design Group

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General Core Mechanical Design Objectives

- **Ensure That Fuel Elements, Control Devices, Burnable Poisons And Fuel Assemblies Are Designed -**
  - ★ To Remain Leak tight For All Operational States Throughout Their Lifetime
    - Designs Shall Account For The Effects Of Temperature, Pressure, Irradiation, Fission Products, Static And Dynamic Loads, And Changes In The Chemical Characteristics Of The Constituent Materials
  - ★ To Provide A Means For Their Safe Handling And Structural Integrity During Transport, Storage, Installation, And Refueling Operations
- **Ensure That The Core And Other Vessel Internals Are Designed -**
  - ★ To Maintain Their Structural Integrity Under All Operational States And Design Basis Accident Conditions
  - ★ To Be Compatible Under The Effects Of Irradiation, And Chemical And Physical Processes
  - ★ To Minimize Any Potential Obstruction To Coolant Flow Due To The Release Of Loose Parts

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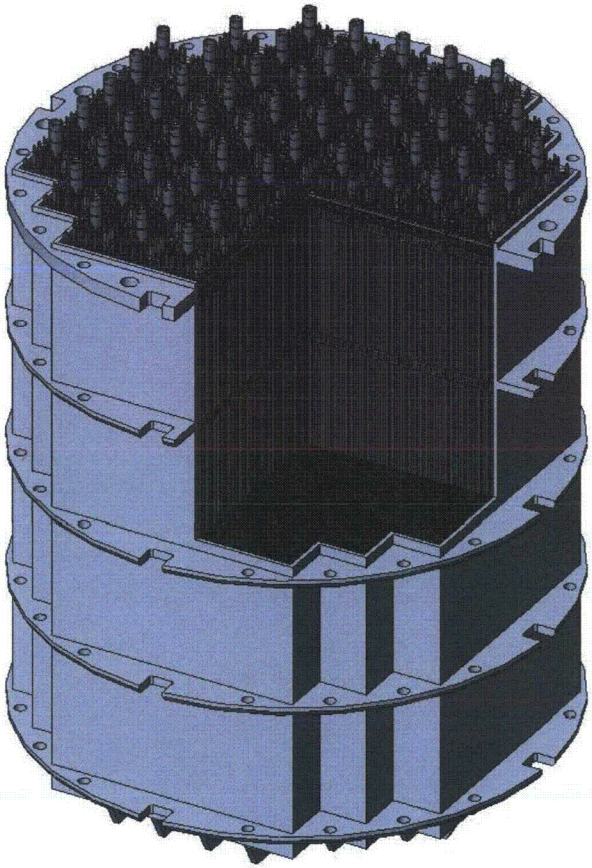
Reactor Core



[CCI per Affidavit 4(a)-(d)]

Conventional Core Configuration with Low Power Density

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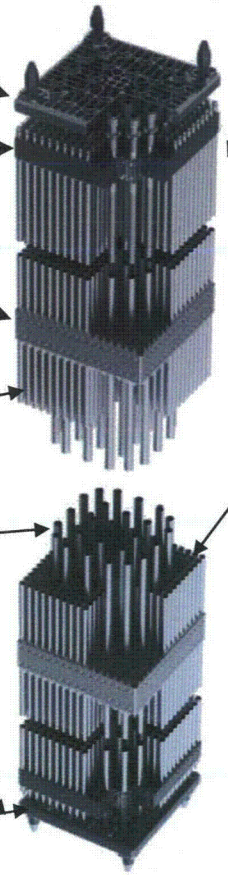
Fuel Assembly Arrangement



Fuel Assembly

- **Remotely Removable End Fittings**
- **No Holddown Spring**
- **Lifted Fuel Rods**
- **805-818 lb Weight**

Fuel Rod

- **Low Power Density**
- **Zircaloy-4 Cladding**
- **Large Plenum Volume (Low End of Life Pressure)**

[CCI per Affidavit 4(a)-(d)]

[CCI per Affidavit 4(a)-(d)]

Shortened and Simplified Conventional Fuel Assembly Design

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Upper End Fitting

[CCI per Affidavit 4(a)-(d)]

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[CCI per Affidavit 4(a)-(d)]

Robust, Low Pressure Drop End Grids

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[CCI per Affidavit 4(a)-(d)]

Robust, Low Pressure Drop Mid Grids

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Guide Tubes

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Lower End Fitting

[CCI per Affidavit 4(a)-(d)]

Simplified Lower End Fitting Design

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Joints & Connections

[CCI per Affidavit 4(a)-(d)]

Reliable Joint Design

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Control Rod Assembly

[CCI per Affidavit 4(a)-(d)]

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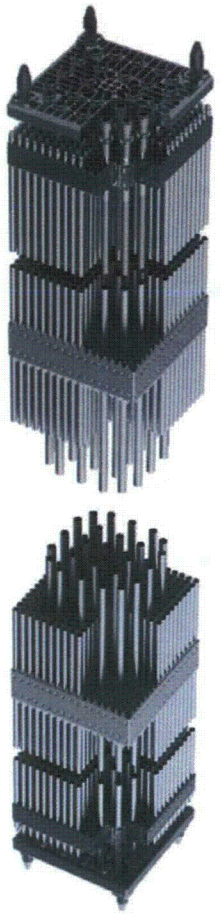
Mechanical Design Summary



[CCI per Affidavit 4(a)-(d)]

Unique, Simple Fuel Assembly Design

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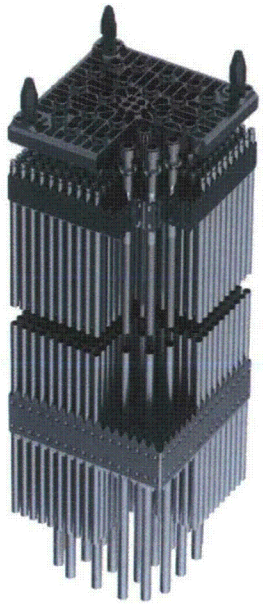

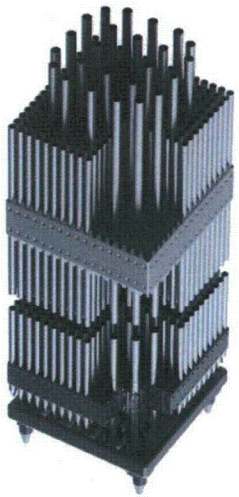

B&W *mPower*™ Core Nuclear Design Features

Gary Neeley
Core Design Group

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Primary Core Design Objectives

- **Load Enough Fuel Inventory To Ensure That -**
  - ★ The Core Excess Reactivity Is Sufficient To Operate At A Steady-State Power Level Of 425 MW(th) For 4 Years At A Capacity Factor > 90% Without Refueling
- **Limit the Core Peak Linear Heat Rate (PLHR) At The Most Reactive Time In Core Life To Less Than 8 kW/ft (~ 60 – 70% Of A Large PWR)**
  - ★ Maximum Core Peaking Factor < 2.3 (Includes An 8% Uncertainty)
  - ★ Maximum Enthalpy Rise Factor < 1.53 (Includes An 8% Uncertainty)
  - ★ Until CHF Data Is Obtained, The Minimum DNBR Will Be Set To 1.3 Under Assumed Limiting Operating Conditions, Namely:
    - 112% Overpower, Most Limiting Axial Power Distribution, Accompanied By An Elevated Inlet Core Coolant Temperature Of 4 °F And A Pressure Swing Of 30 psi
- **Ensure Core Shutdown Margin (SDM) Of > 1% $\Delta k_{eff}/k_{eff}$ Under Cold Conditions At The Most Reactive Time In Core Life, No Xenon, No Boron In Coolant, And The Highest Worth Rod Cluster Stuck Out**

Extended Core Life With Large Thermal Margins

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Core Loading Arrangement

[CCI per Affidavit 4(a)-(d)]

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Control Rod Assembly (CRA) Banks

[CCI per Affidavit 4(a)-(d)]

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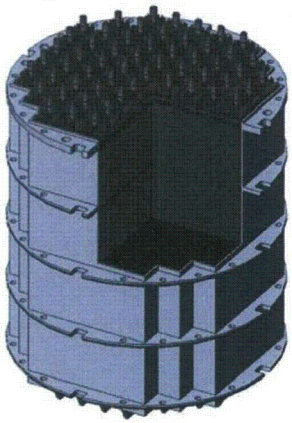
Core Design Parameters

Core Operating Conditions

- Core Thermal Power – 425 MW(th)
- Mass Flow Rate – 3208 kg/sec (25.4 Mlbm/hr)
- Core Inlet Temperature – 570.9 K (568 °F)
- Core Outlet Temperature – 593.4 K (608.5 °F)
- Core Outlet Pressure – 13.2 MPa (1914.5 psia)

Core Cycle Length Parameters

- Core Average Fuel Burnup – 34.5 GWd/MTU
- Core Cycle Length – 1360 EFPD



[CCI per Affidavit 4(a)-(d)]

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Core Excess Reactivity (ARO Burn)

[CCI per Affidavit 4(a)-(d)]

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Fuel Temperature Coefficient Of Reactivity

[CCI per Affidavit 4(a)-(d)]

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Moderator Temperature Coefficient Of Reactivity

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Shutdown Margin During Core Life

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Cycle Control Rod Patterns

[CCI per Affidavit 4(a)-(d)]

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Core Power Distribution - BOL

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Core Power Distribution - MOL

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Core Power Distribution - EOL

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Axial, Radial, and Nodal Peaking

[CCI per Affidavit 4(a)-(d)]

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Core Nuclear Analysis Methodology Verification

- **Detailed 3-D Nuclear Model**
  - ✓ Cross-Sections Generated With ***CASMO-5*** Using ENDBF-VII
  - ✓ 3-D Core Simulations Performed With ***SIMULATE-3***
- **Core Design Methodology Verification**
  - ✓ Comparison of Calculated to Measured Core Data (Startup And Core Follow)
    - Cold Criticals
    - TMI Unit 1 Startup
    - TMI Unit 1 Cycles 1 and 2 Core Follow
  - ✓ Topical To Be Submitted To NRC

TMI-1 Cycle-1 Loading Pattern

	8	9	10	11	12	13	14	15
H	2.75	2.75 1.43	2.06	2.75 1.26	2.06	2.75 1.26	3.05	3.05
K	2.75 1.43	2.06	2.75 1.43	2.06	2.75 1.26	2.06	2.75 1.43	3.05
L	2.06	2.75 1.43	2.06	2.75 1.09	2.06	2.75 1.26	3.05	3.05
M	2.75 1.26	2.06	2.75 1.09	2.06	2.75 1.09	2.06	3.05	
N	2.06	2.75 1.26	2.06	2.75 1.09	2.06	3.05 1.26	3.05	
O	2.75 1.26	2.06	2.75 1.26	2.06	3.05 1.26	3.05		
P	3.05	2.75 1.43	3.05	3.05	3.05			
R	3.05	3.05	3.05					

BP	2.06	3.05	2.75	
0	56	52	1	
1.43			20	
1.26		8	24	
1.09			16	
	56	60	61	177

BP wt% B_4C	Fuel wt% ^{235}U
1.43	2.06
1.26	3.05
1.09	2.75

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Nuclear Methodology Verification – Cold Criticals

MEAS	0.000	1.102	0.986	1.003	1.015	1.005	1.045	1.079
CALC	0.000	1.095	0.997	0.988	0.991	0.995	1.034	1.067
DIFF	0.000	-0.007	0.011	-0.015	-0.024	-0.010	-0.011	-0.012
		0.995	0.907	0.961	0.943	0.924	1.027	1.061
		1.017	0.918	0.951	0.954	0.925	1.007	1.060
		0.022	0.011	-0.010	0.011	0.001	-0.020	-0.001
			0.000	0.865	0.864	0.000	0.961	1.044
			0.000	0.868	0.868	0.000	0.950	1.053
			0.000	0.003	0.004	0.000	-0.011	0.009
				0.878	0.820	0.896	1.007	1.045
				0.884	0.848	0.882	1.006	1.072
				0.006	0.028	-0.014	-0.001	0.027
					0.000	0.931	1.053	1.093
					0.000	0.938	1.052	1.097
					0.000	0.007	-0.001	0.004
						1.028	1.095	1.147
						1.027	1.093	1.122
						-0.001	-0.002	-0.025
							1.118	1.151
							1.134	1.144
							0.016	-0.007
								1.158
								1.157
								-0.001

MAX DIFF	0.028
AVE DIFF	0.000
RMS	0.013
STDEV	0.013

UO2 rod

Water rod

Pyrex rod

Vicor rod

Al2O3 rod

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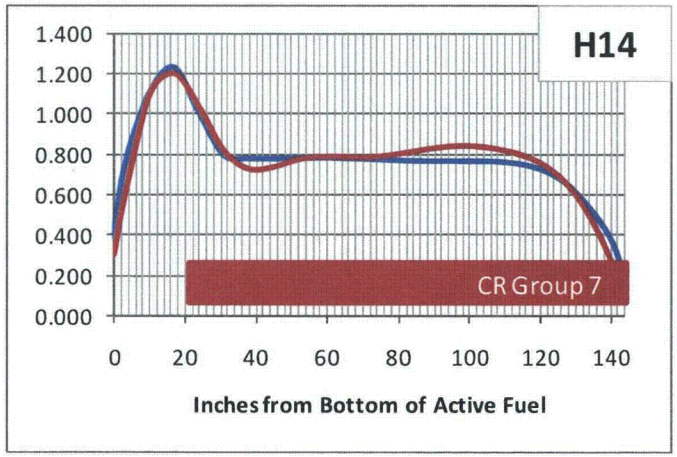
Nuclear Methodology Verification - TMI Unit 1 Cycle 1

	H-8	H-9	H-10	H-11	H-12	H-13	H-14	H-15
Measured	0.943	1.173	1.234	1.186	1.173	1.114	0.800	0.707
Calculated	0.939	1.202	1.171	1.263	1.123	1.133	0.788	0.697
Difference	-0.004	0.029	-0.062	0.077	-0.051	0.019	-0.013	-0.010
		K-9	K-10	K-11	K-12	K-13	K-14	K-15
		1.144	1.259	1.194	1.172	1.149	1.042	0.751
		1.150	1.264	1.157	1.218	1.105	1.051	0.758
		0.006	0.005	-0.036	0.046	-0.044	0.009	0.007
			L-10	L-11	L-12	L-13	L-14	L-15
			1.154	1.160	0.999	1.168	1.209	0.703
			1.156	1.201	0.983	1.196	1.222	0.671
			0.002	0.041	-0.016	0.028	0.013	-0.032
				M-11	M-12	M-13	M-14	
				0.974	1.045	1.004	0.909	
				1.021	1.029	0.967	0.910	
				0.047	-0.016	-0.037	0.001	
					N-12	N-13	N-14	
					0.617	0.779	0.560	
					0.609	0.794	0.556	
					-0.008	0.015	-0.004	
						O-13		
						0.533		
						0.536		
						0.003		

- Highlighted Values Indicate Peak Radial Power

RMS = 0.0308



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Core Nuclear Design Summary

- **Methods Qualification**
  - ✓ Nuclear Methodology Verification Using ***CASMO-5/SIMULATE-3*** In Progress
  - ✓ Topical Submission To NRC In 2010
- **Core Nuclear Design Status**

[ ]

[CCI per Affidavit 4(a)-(d)]

- **Future Core Nuclear Design Improvements**
  - ✓ Design Is Progressing With Next Iteration Employing

[ ]

[CCI per Affidavit 4(a)-(d)]

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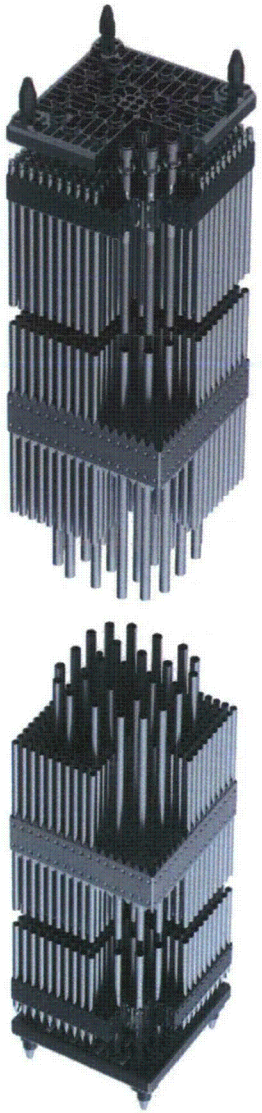

B&W *mPower*™ Core Thermal-Hydraulic Design Features

Steve Scoles
Core Design Group
B&W

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Core Thermal-Hydraulic Design Objectives

- **Core Will Be Designed With Sufficient Margin To Ensure That The Hot Fuel Rod In The Core Does Not Experience A Departure From Nucleate Boiling (DNB) Phenomenon During Normal Operation And Anticipated Operational Occurrences (AOOs)**
  - ★ A Critical Heat Flux (CHF) Correlation That Is Applicable To The B&W ***mPower***™ Core Operating State And Fuel Design Will Be Developed Based On Critical Heat Flux Tests Performed At Stern Labs.
  - ★ Statistical Analysis Will Be Used To Obtain The Minimum Departure From Nucleate Boiling Ratio (DNBR) Based Core Uncertainties With Plant Operating Parameters, Nuclear And Thermal Parameters, Fuel Fabrication Parameters, Computer Codes, And Departure From Nucleate Boiling (DNB) Correlation Predictions.
  - ★ Until CHF Data Is Obtained, The Minimum DNBR Based on the EPRI Correlation Will Be Set To 1.3 Under Limiting Operating Conditions, Namely:
    - 112% Overpower, Most Limiting Axial Power Distribution, Accompanied By An Elevated Inlet Core Coolant Temperature Of 4 °F And A Pressure Swing Of 30 psi
    - Since It Is Very Unlikely That All Of These Conditions Would Occur Simultaneously, This Conservative Assumption Provides Sufficient Thermal Margin To Withstand A Full Spectrum Of Transients/Accidents

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Core Thermal-Hydraulic Design Objectives

- **Core Will Be Designed So That The Fuel Rod With The Most Limiting Linear Heat Rate Does Not Cause The Fuel Pellet To Melt During Normal Operation And AOOs**
  - ★ The Melting Temperature Of UO_2 Pellets Is 5072°F (2800°C) For Un-Irradiated Fuel And Decreases By 58°F (32°C) Per 10,000 MWd/MTU
  - ★ The Melting Temperature Of UO_2-Gd_2O_3 Pellets Is 4892°F (2700°C) At A Maximum Content Of 10 wt% For Un-Irradiated Fuel And Decreases By 58°F (32°C) Per 10,000 MWd/MTU
  - ★ Due To The Lower Melting Temperature And The Lower Thermal Conductivity Of The UO_2-Gd_2O_3 Fuel Rods
    - The ^{235}U Enrichment In The UO_2-Gd_2O_3 Pins Is Limited To A Maximum Of[ ] Less Than That Of The UO_2 Pins In The Bundle
      [CCI per Affidavit 4(a)-(d)]
    - Maximum Content Of Gd_2O_3 In The UO_2-Gd_2O_3 Pins Is Limited To ≤[ ]
      [CCI per Affidavit 4(a)-(d)]

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Core Thermal-Hydraulic Design Objectives

- **Core Will Be Designed So That The Peak Linear Heat Rate (PLHR) At The Most Reactive Time In Core Life Is Less Than 8 kW/ft**
  - ★ Maximum Core Peaking Factor < 2.3 (Includes An 8% Uncertainty)
  - ★ Maximum Enthalpy Rise Factor < 1.53 (Includes An 8% Uncertainty)
- **The Core Thermal-Hydraulic Design Will Be Based On The Effective Core Flow, i.e. Total Reactor Coolant Flow Minus The Design Core Bypass Flow**
  - ★ Design Core Bypass Flow Includes:
    - Flow Through CRA Guide Thimbles And In-Core Instrumentation Guide Tubes
    - Leakage Flow To The Core Cavity
    - Leakage Flow To Reactor Vessel Components
    - Reserve Margin For Steam Generator Tube Plugging And Uncertainties On Pressure Loss, Pump Performance, And Flow Measurement

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Core CHF Correlation

- A New CHF Correlation For The B&W ***mPower***™ Core Will be Developed
  - ✓ Described in B&W Technical Report Number 06-00000334-000, "CHF Test and Correlation Development Plan" (Submitted To The NRC In April 2010)
  - ✓ Based On Existing and Planned CHF Test Data
    - Selected Portions of the EPRI CHF Database
    - New Data From B&W Sponsored Testing At Stern Labs In Canada
- Until CHF Testing Is Complete, The Design Calculations Will Use The EPRI Correlation

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Existing and Proposed CHF Test Data

[CCI per Affidavit 4(a)-(d)]

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Existing and Proposed CHF Test Data

[CCI per Affidavit 4(a)-(d)]

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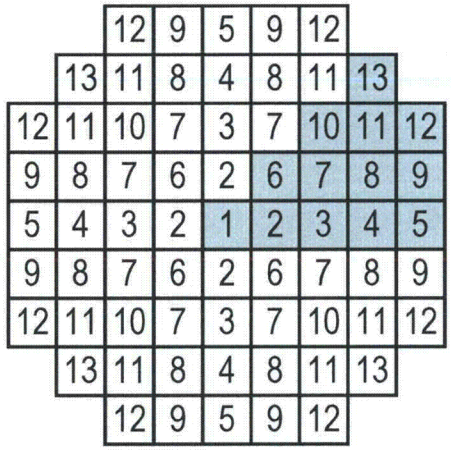
Core Thermal-Hydraulic Analysis

- **Detailed 3-D Thermal-Hydraulic Model**
  - ✓ 1/8 Core Model With Crossflow
  - ✓ 3-D Core Simulations Performed With ***VIPRE-01***
- **Two Operating Conditions**
  - ✓ Nominal
    - Power – 425 MW(th)
    - Mass Flow Rate – 3200 kg/sec
    - Inlet Temperature – 570.9 K
    - Outlet Temperature – 593.5 K
    - Outlet Pressure - 13.2 MPa
  - ✓ Limiting Case
    - Power – 476 MWth
    - Mass Flow Rate – 3200 kg/sec
    - Inlet Temperature – 573.1 K
    - Outlet Temperature – 598.0 K
    - Outlet Pressure – 13.0 MPa
- **91 Burnup Steps**



Bundle Numbering

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Thermal-Hydraulic Analysis Results

[CCI per Affidavit 4(a)-(d)]

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Fuel Centerline Temperature – Limiting Time, Limiting Channel

[CCI per Affidavit 4(a)-(d)]

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DNBR – Limiting Time, Limiting Channel

[CCI per Affidavit 4(a)-(d)]

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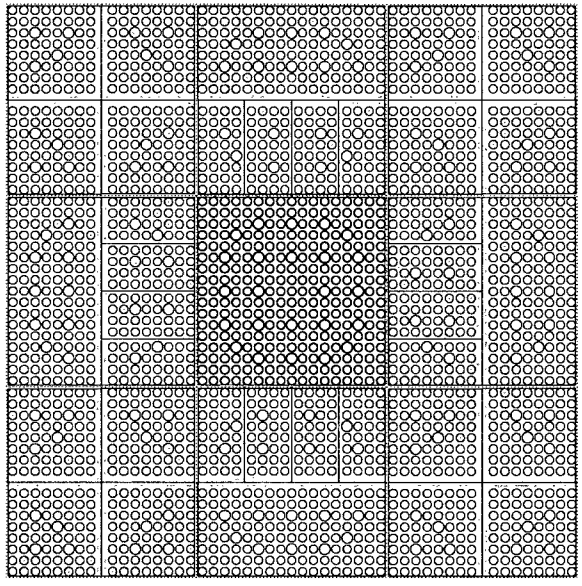
Subchannel Thermal-Hydraulic Analysis

- **Detailed 3-D Thermal-Hydraulic Model**
  - ✓ Single Bundle Modeled at the Subchannel Level With Surrounding Bundles Modeled As Lumped Channels
  - ✓ Each of the 13 Unique Bundles in the Core Will Be Modeled Individually
  - ✓ Results From the Core Model Will Be Used to Ensure That the Subchannel Model Boundary Conditions Are Correct
- **The Same Operating Conditions and Burnup Steps Used in the Core Model Will Be Used**
- **Results From the Two Models Will Be Checked Against Each Other For Consistency**
- **Model Is Currently Under Development, Preliminary Results Compare Well With the Core Model**



Subchannel Model

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Core Thermal-Hydraulic Design Summary

- **Models**
  - ✓ Models Will Use ***VIPRE-01***
  - ✓ Overall Core Thermal-Hydraulics Model Is Complete
  - ✓ Detailed Subchannel Thermal-Hydraulics Model Is Under Development
- **Critical Heat Flux Correlation**
  - ✓ EPRI Correlation Is Currently Being Used For Design
  - ✓ A New Correlation Based On A Subset Of The EPRI CHF Database And New Data From Stern Labs Is Planned

[CCI per Affidavit 4(a)-(d)]

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Conclusion

- **Mechanical Design**
  - ✓ Unique, Simple Fuel Assembly Design
  - ✓ Mechanical Tests To Qualify The Design Planned
  - ✓ Fuel Performance Code V&V Report In 2012
- **Nuclear Design**
  - ✓ Sufficient Excess Reactivity For Extended Cycle (4-Year) Core Operation
  - ✓ Conservative Nuclear Design With Respect To Peaking Factors, Shutdown Margin, And Other Nuclear Parameters
  - ✓ Nuclear Design Methodology Topical To Be Submitted In September 2010
- **Thermal-Hydraulic Design**
  - ✓ Large Thermal Margins Resulting From Low Linear Heat Rate
  - ✓ Conservative DNBR
  - ✓ Critical Heat Flux Test Program In Progress
  - ✓ Thermal-Hydraulic Design Methodology Topical To Be Submitted in 2011

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[CCI per Affidavit 4(a)-(d)]

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B&W mPower™ Design Evolution

Overview of the B&W mPower Reactor Design

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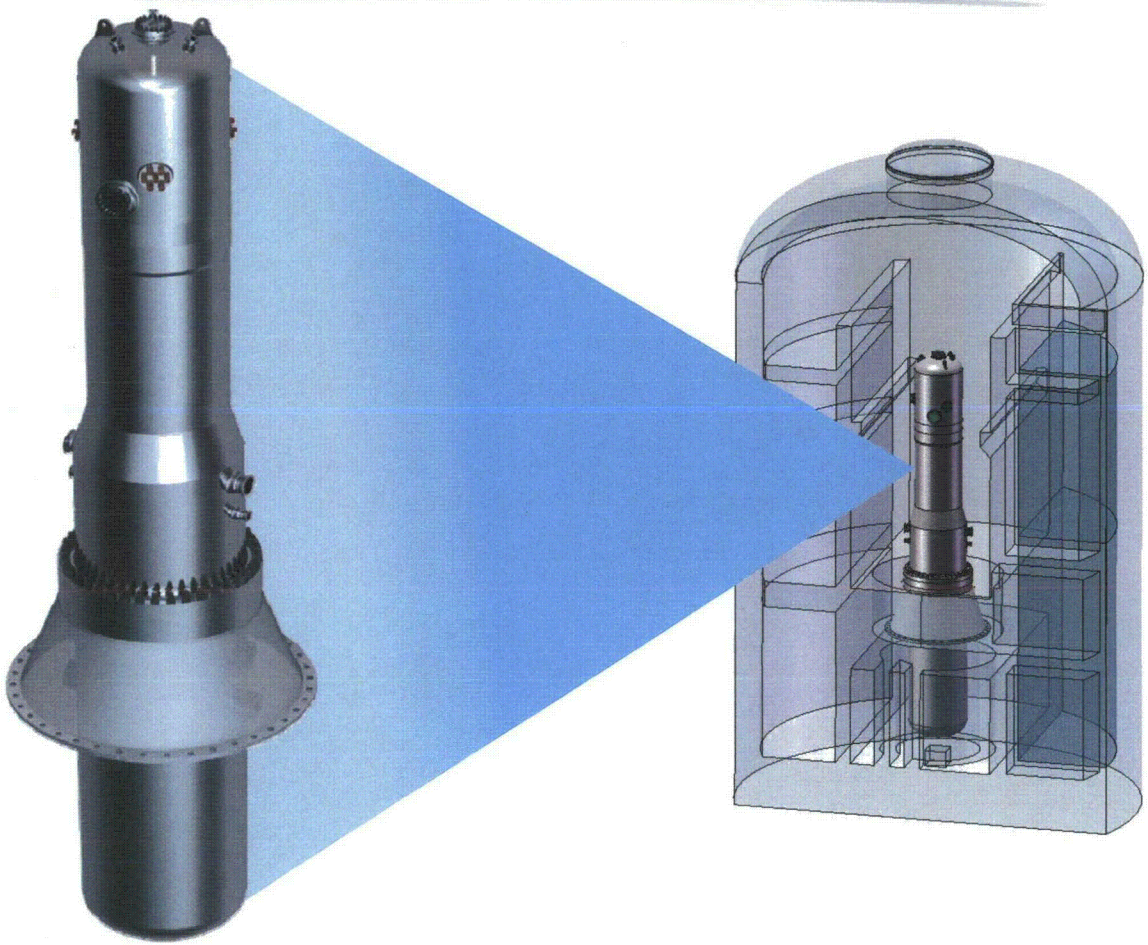

Overview of the B&W *mPower*™ Reactor Design

Briefing for the Nuclear Regulatory Commission

July 29, 2010

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Integral Nuclear Steam Supply System

- Integrates core, steam generator, and pressurizer into a single vessel
- Control rod drive mechanisms (CRDMs) and primary coolant pumps inside vessel
- Reactor coolant pressure boundary penetration size and location minimize coolant loss during LOCA
- Rail shippable
- Core remains covered throughout design basis LOCA

Integral design reduces overall plant complexity and enhances safety

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B&W *mPower* NSSS Design

[CCI per Affidavit4(a)-(d)]

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NSSS Design Characteristics

[CCI per Affidavit 4(a)-(d)]

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Lower Reactor Assembly

[CCI per Affidavit 4(a)-(d)]

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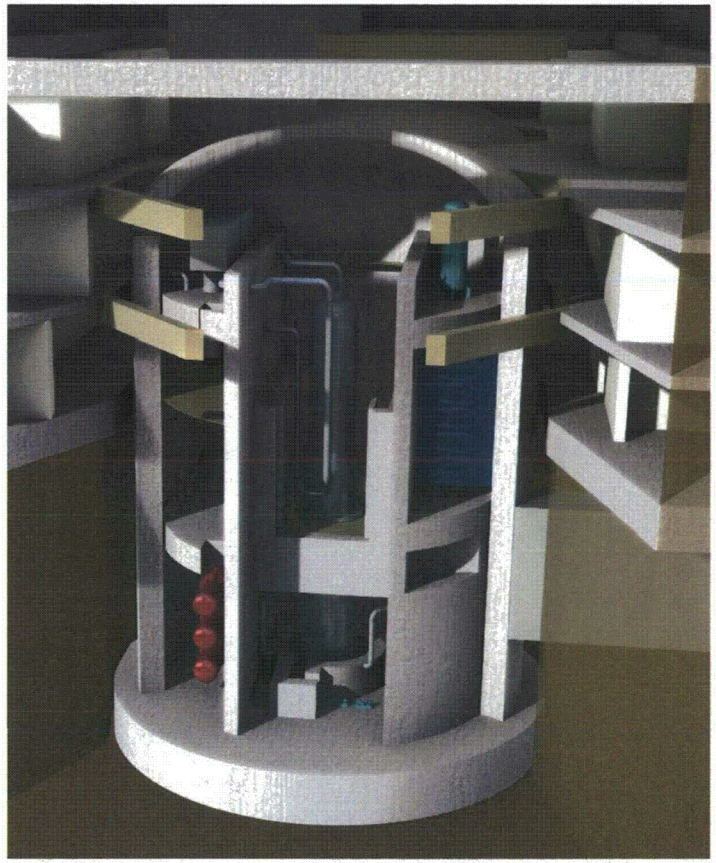
Containment Layout



Each NSSS is housed within a steel lined, reinforced concrete, dry containment

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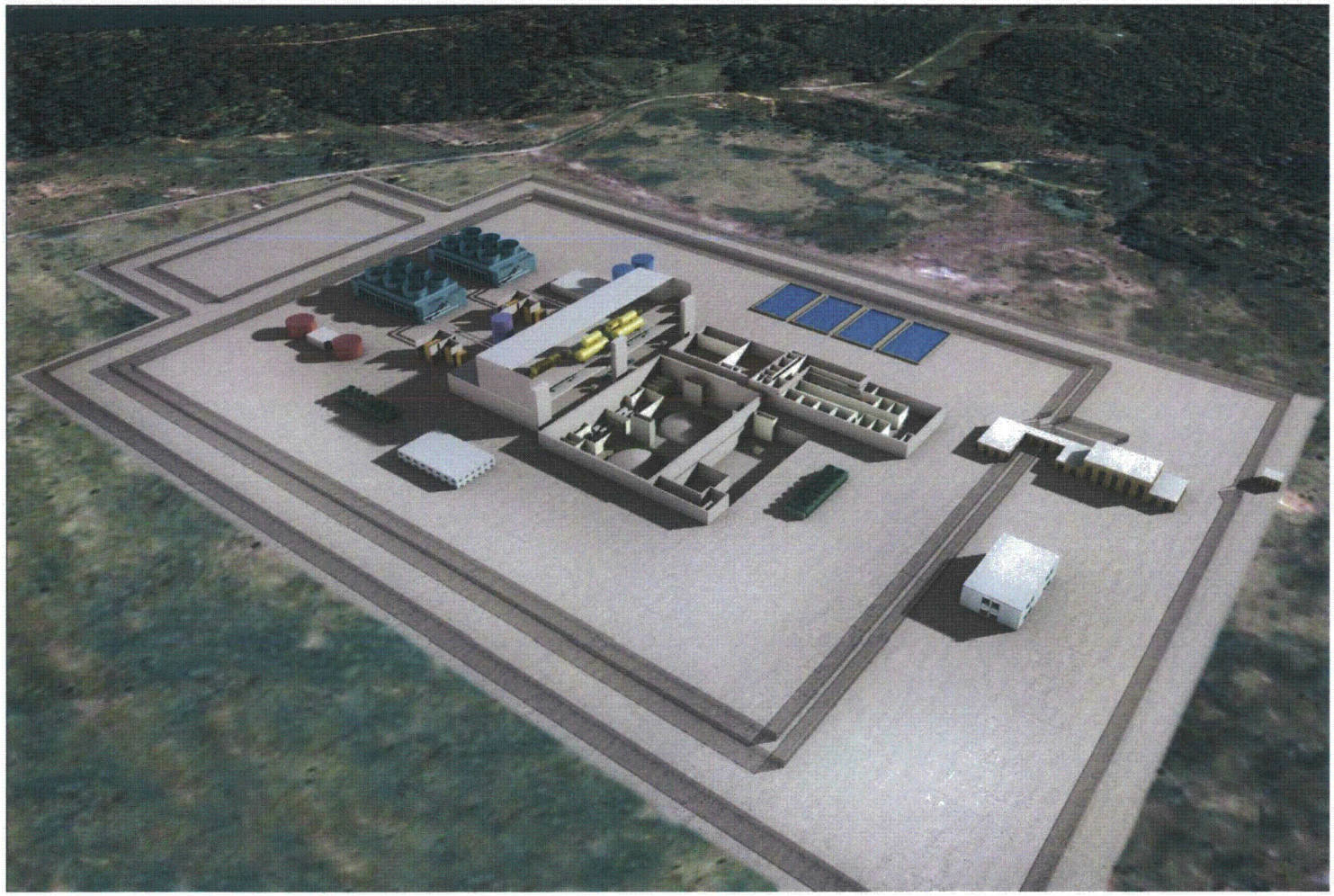
Two-Module Plant Conceptual Layout



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Summary

- NSSS utilizes an integral PWR design
  - Uses a single integral economizer once through steam generator to produce superheated steam
  - Internal reactor coolant pumps and control rod drive mechanisms
  - Internal pressurizer
- Long operating cycle without soluble boron
- Store spent fuel outside of containment (below ground)
- Reactor plants for multiple module designs will not share engineered safety features

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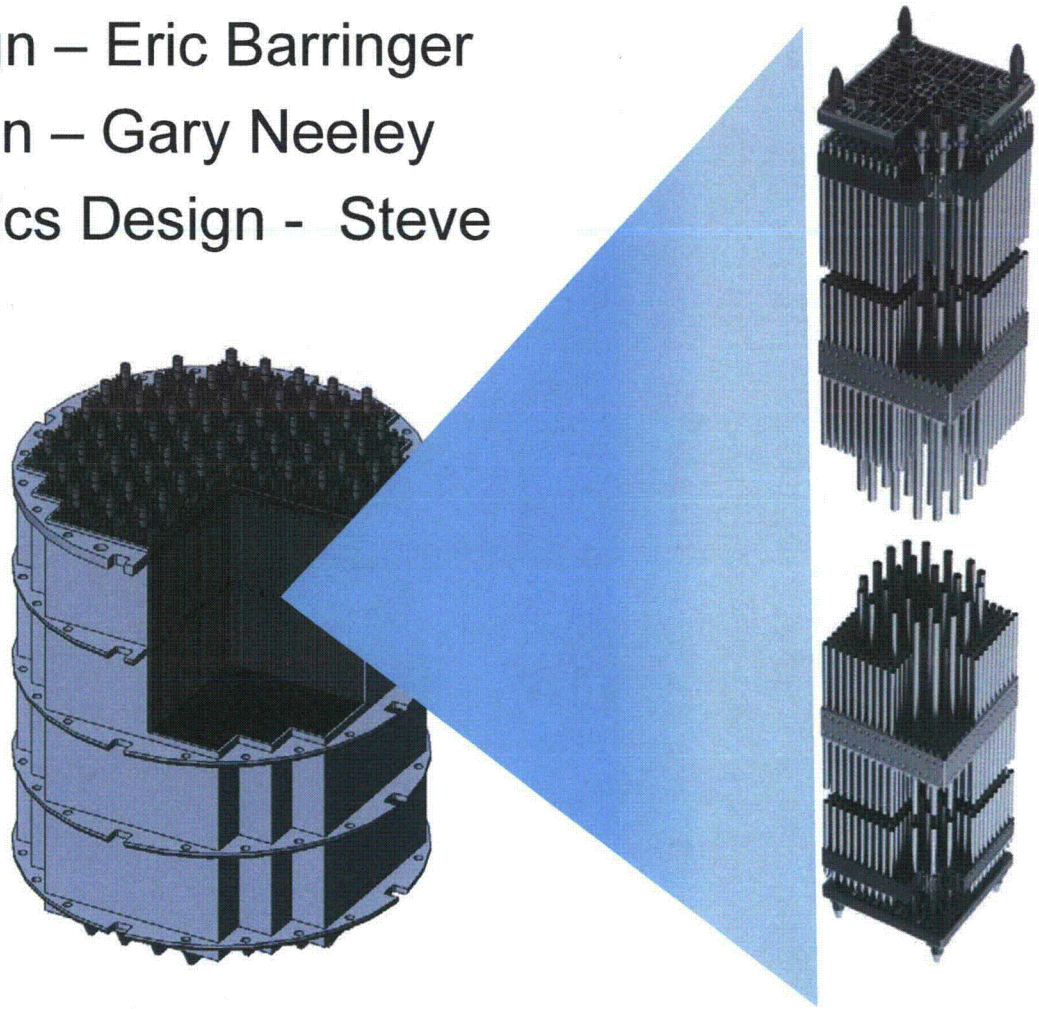
Core and Fuel Design Topics

- Fuel Mechanical Design – Eric Barringer
- Core Neutronics Design – Gary Neeley
- Core Thermal-Hydraulics Design - Steve Scoles
- Topical Reports



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