Westinghouse Fuel Performance Update Meeting (Slide Presentations for February 15-16, 2011) and Associated Material (Non-Proprietary)

Westinghouse Fuel Performance Update Meeting

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Westinghouse Fuel Performance Update Meeting
Agenda
February 15, 2011

Tuesday, February 15, 2011 (Westinghouse, NRC, and Customers)

8:20 a.m. – 8:30 a.m.	Introductions and Welcome
8:30 a.m. – 9:30 a.m.	PWR Fuel Performance Update
9:30 a.m. – 9:45 a.m.	PWR Hydrogen Model
9:45 a.m. – 10:00 a.m.	BEACON™ Addendum 4
10:00 a.m. – 10:10 a.m.	Break
10:10 a.m. – 11:10 a.m.	Material Updates
11:10 a.m. – 11:40 a.m.	High Burnup Licensing
11:40 a.m. - Noon	Spent Fuel Pool Criticality
Noon – 12:30 p.m.	AP1000® Licensing
12:30 p.m. – 1:30 p.m.	Lunch
1:30 p.m. – 2:30 p.m.	BWR Fuel Performance Update
2:30 p.m. - 3:00 p.m.	Optima3 Update
3:00 p.m. – 3:15 p.m.	BWR Hydrogen Model
3:15 p.m. – 3:30 p.m.	Break
3:30 p.m. – 4:00 p.m.	ABWR Licensing Update
4:00 p.m. – 4:30 p.m.	ODEN Update
4:30 p.m. – 5:00 p.m.	CASL Update
5:00 p.m.	Adjourn

Acronym List

3D	Three Dimensional
ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AMA	Advanced Modeling Applications
AOO	Anticipated Operational Occurrence
BA	Burnable Absorber
BEACON	Best Estimate Analyzer for Core Operations-Nuclear
BNFL	British Nuclear Fuels
BU	Burn Up
BWR	Boiling Water Reactor
CAD	Computer Aided Drafting
CAP	Corrective Action Process
CASL	Consortium for Advanced Simulation of Light Water Reactors
CE	Combustion Engineering
CE-NSSS	Combustion Engineering Nuclear Steam Supply System
CFD	Computational Fluid Dynamics
CFRA	Critical Fuel Reliability Attributes
CILC	Crud-Induced Localized Corrosion
CIPS	Crud-Induced Power Shift
COL	Combined Operating License
CPR	Critical Power Ratio
CR	Control Rod
CRR	Core Reference Report
CY	Calendar Year
DCD	Design Certification Document
DFBN	Debris Filter Bottom Nozzle
DNB	Departure from Nucleate Boiling
DNB	Departure from Nucleate Boiling
DQ	Data Qualification
ECR	Equivalent Clad Reacted
EPRI	Electric Power Research Institute
F/A	Fuel Amendment
FAD	Fuel Assembly Distortion
FMEA	Failure Modes and Effects Analysis
FP	Fuel Performance
FR	Fuel Rod
FRISC	Fuel Reliability Improvement Steering Committee
FSAR	Final Safety Analysis Report
FSI	Fluid Structural Interaction
FY	Fiscal Year
GTRF	Grid-to-Rod Fretting
HD	High Density
HiBU	High Burnup
HPU	Hydrogen Pick Up
HTC	Haut Taux de Combustion
HTRF	Heat Transfer Research Facility
HuP	Human Performance

Acronym List

HWC	Hydrogen Water Chemistry
I&C	Instrumentation and Control
IASCC	Irradiation Assisted Stress Corrosion Cracking
ISG	Interim Staff Guidance
IT	Instrument Tube
LAR	License Amendment Request
LIME	Light weight Integrating Multiphysics Environment
LK3	Low Corrosion Cladding
LOCA	Loss of Coolant Accident
LTA	Lead Test Assembly
LTR	Licensing Topical Report
LUA	Lead Use Assembly
MFDI	Modified Fuel Duty Index
MNM	Models and Numerical Methods
MOX	Mixed Oxide
MPO	Materials Performance and Optimization
NF	Nuclear Fuel
NMCA	Noble Metal Chemistry Application
NRC	Nuclear Regulatory Commission
NRO	Office of New Reactors
NRR	Office of Nuclear Reactor Regulation
NWC	Normal Water Chemistry
OD	Outer Diameter
OE	Operating Experience
OFA	Optimized Fuel Assembly
[	][a,c]
ORNL	Oak Ridge National Laboratory
PCI	Pellet-Cladding Interaction
PCMI	Pellet-Cladding Mechanical Interference
PCT	Peak Cladding Temperature
PE	Project Engineering
PIE	Post-Irradiation Examine
PQD	Post Quench Ductility
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCA	Root Cause Analysis
RCCA	Rod Cluster Control Assembly
RFA	Robust Fuel Assembly
RIA	Rod Insertion Accident
RIP	Rod Internal Pressure
RPG	Robust P-Grid
SBLOCA	Small Break Loss of Coolant Accident
SER	Safety Evaluation Report
SP	Swedish Standards Laboratory
SPP	Second Phase Precipitate
SRP	Standard Review Plan (NUREG-0800)
SSD	Scheduled Shipping Date
STD	Standard

Acronym List

T/C	Thermocouple
T/H	Thermal/Hydraulic
TVA	Tennessee Valley Authority
UPG	Upgrade
UQ	Uncertainty Quantification
VCS	Virgil C. Summer
VR	Virtual Reactor
VRI	Virtual Reactor Integration
VUQ	Validation and Uncertainty Quantification
Zry-2	Zircaloy-2
Zry-2 B-Q	Zircaloy-2 Beta-Quenched
Zry-4	Zircaloy-4

PWR Fuel Performance Update

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Agenda

- Historical Performance

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Percent of Plants Operating with Zero Leakers

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Historical Performance

- In 2002 Westinghouse Leak-Free Performance in the U.S. near 70%
  - Westinghouse Leaker Performance lagging industry average
    - Performance Issues Across Westinghouse Fuel Product Lines

17x17 V5H	Grid-to-Rod Fretting
15x15 OFA/V5H	Grid-to-Rod Fretting
All CE Products	Grid-to-Rod Fretting
17x17 OFA	Missing Pellet Surface
All (PWR/BWR)	Debris(FM)/Internal Contamination

- Flawless Fuel Program (2002)
  - Emphasis on prevention rather than reaction
  - FMEA methodology
  - More robust fuel designs developed and implemented
  - Systematic Root Cause Analysis of leakers. Corrective actions implemented

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Number of Rods per Failure Mechanism Confirmed During Outages (for all Westinghouse supplied fuel)

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Number of Rods per Product Line Confirmed During Outages (for all Westinghouse Supplied Fuel)

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Agenda

- Fuel Reliability Improvement Process

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Fuel Reliability Improvement Process Background

- Westinghouse Flawless Fuel Program introduced in 2002
  - Moved from "find and fix" to "anticipate and prevent"
  - Initiated use of Failure Modes & Effects Analysis methodology to identify potential weaknesses in design and fabrication processes
  - Significant achievements of the Flawless Fuel Program include:
    - GTRF resistant designs developed and implemented (17RFA, 15UPG, 16NGF)
    - Improved pellet quality (design, process, inspection)
    - Implemented foreign material exclusion (hydrogenous contaminants, debris)
    - Systematic use of root cause analysis of leakers to implement effective corrective actions

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Fuel Reliability Improvement Process Map

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Process Designed to Insure Continuous Improvement and Sustainability

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Fuel Reliability Trimester Assessments

-CFRA-Related CAP analysis performed by Quality-

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Period 9/2009-3/2010:

Example

-FPT Recommendations Resulting from CFRA Analysis-

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Fuel Reliability Improvement Steering Committee (FRISC)

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Fuel Reliability Improvement Process

-FRI Projects to be Completed FY10-

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Process Sustainability:

- Formal Fuel Reliability Improvement Process -

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Agenda

- Current Performance Issues
  - U.S. leakers discharged in 2010
  - U.S. plants currently leaking
  - Healthy fuel performance

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U.S. Leakers Discharged in 2010

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Current Leaking Plants (January 2011)

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Healthy Fuel Inspections

Summary

- A total of twenty six plants fueled by Westinghouse have completed baseline inspections in the U.S. and Europe
- Twenty one PWRs and five BWRs
- Sixteen in the U.S. (12 Westinghouse-NSSS, 3 CE-NSSS, 1 BWR)
  - Priority 1: Twelve plants
  - Priority 3: Four plants

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Westinghouse Fuel Surveillance Inspection Status in the U.S.

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Inspection Results

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- Status of Current Challenges
  - Robust P-Grid (RPG) Update
  - Instrument Tube (IT) Wear

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Robust P-Grid Update

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Final Robust P-Grid Strap Design

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P-Grid/Bottom Grid ("Grid Combo") Intro

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Grid Laser Welding

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Instrument Tube Wear

- Instrument Tube (IT) Wear discovered at recent [ ]a,c outages
- In one case, the wear was through the IT wall and the assembly was not re-inserted. Evidence of interaction with P-grid was seen.
- A root cause analysis was performed. The extent of condition indicated two discharged assemblies with IT wear had been observed in the mid-1990s. They were considered isolated cases.
- Recently, ten discharged assemblies were inspected for IT wear at [ ]a,c. Inspection results showed zero to moderate IT wear.

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Instrument Tube Wear

- Lower Core Plate/Bottom Nozzle Schematic -

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Instrument Tube Wear

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Instrument Tube Wear

Conclusion

- An evaluation of [ ]a,c IT wear data has concluded that the wear on the instrument tube does not create a safety or operational hazard to the core
- Neither the instrument tube nor the P-Grid provides any structural support to the assembly
- Interaction of these components with the flux thimble provides no adverse performance to the assembly or the flux thimble

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Cladding Hydrogen - Testing Status & Experimental Parameters Introducing Conservatism in Proposed NRC Criteria for LOCA Embrittlement & RIA Enthalpy

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Testing to Support Cladding Hydrogen Based Accident Criteria

- Demonstration of equivalence of ZIRLO®, Optimized ZIRLO™ and Zircaloy-4 cladding in term of embrittlement as function of ECR
- Demonstration of breakaway performance of ZIRLO cladding
- ZIRLO and Optimized ZIRLO cladding corrosion and hydrogen pickup fraction models
- Testing to establish post quench ductility (PQD) limits as a function of hydrogen and peak clad temperature (PCT)

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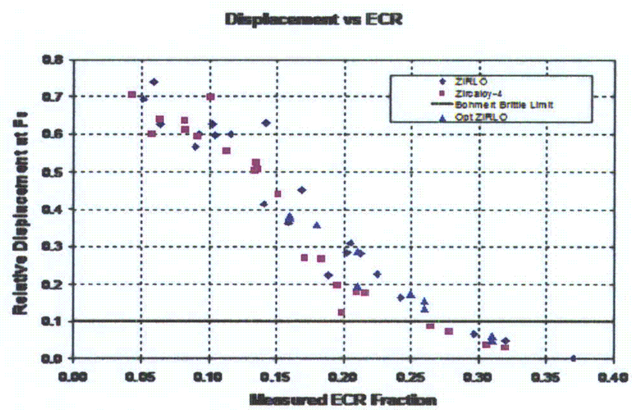
Demonstration of equivalence of ZIRLO, Optimized ZIRLO and Zircaloy-4 Cladding



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Demonstration of breakaway performance of ZIRLO cladding.

- Westinghouse has been corresponding with the NRC on the subject of breakaway times for ZIRLO cladding and differences between values reported by Westinghouse and those reported by Argonne National Laboratory (ANL)
- Subject has been covered in five letters
- Testing has established main difference between ANL and Westinghouse testing protocols is heatup rate

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Demonstration of breakaway performance of ZIRLO cladding.

- Based on test results ZIRLO cladding has a breakaway times of > 5,400 seconds when subjected to temperatures around 1,000 °C with heatup rates that are representative of SBLOCA events
- The reason for the difference between shorter ANL breakaway times and longer Westinghouse determined times is the difference in heatup rates
- Westinghouse breakaway heatup times use time temperature profiles based on SBLOCA events

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Cladding corrosion and hydrogen pickup fraction models.

- Corrosion and HPU models submitted to NRC in topical WCAP-12610-P-A & CENPD-404-P-A, Addendum 2-P, Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO, November 2008

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Testing to establish PQD limits as a function of hydrogen and PCT

- PQD testing at various PCTs is underway. Initial evaluation shows increase in ECR where embrittlement occurs at lower PCTs
- EPRI submits Zirc-4 cladding PQD test results to NRC
- Westinghouse submits ZIRLO and Optimized ZIRLO Cladding PQD test results to NRC

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Milestones to Impliment New Accident Criteria

- Owners groups submit performance margin assessment report with respect to proposed rule
- Owners groups and NRC meeting on performance margin assessment report
- ACRS on Proposed Rule
- Complete and submit ZIRLO and Optimized ZIRLO cladding PQD Testing at RTU
- Proposed rule published in Federal Register
- Work with industry to respond to Federal Register posting
- NRC Issues final rule

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Non-PWR Typical Experimental Parameters Used in NRC LOCA Data

- Faster Heat-up Rate than Typical PWR SBLOCA
  - Under-Prediction of Breakaway Oxidation Time
- Higher Peak Clad Temperature compared to higher burnup fuel in either LOCA or SBLOCA
  - Excessive Oxygen Embrittlement due to greater Oxygen Penetration in Clad Wall Interior
- Higher Pre-quench Cooling Rate than PWR Clad
  - Higher Oxygen Embrittlement due to Less Oxygen Partitioning in the Clad Microstructure
- Higher Quench Temperature
  - Higher H Embrittlement due to less H partitioning

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Conservative Factors in RIA Data used to Derive Proposed NRC Enthalpy Limits

- Simultaneous Application of Following Factors
  - Lower Coolant Temperature (NSRR) than PWR Coolant
  - Shorter Pulse width (NSRR) than Typical PWR Transient
  - Inclusion of MOX data (Cabri) to Derive Limits for UO_2 Fuel

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Conclusions on Hydrogen based Accident Criteria

- Based on test results, for ZIRLO and Optimized ZIRLO:
  - The breakaway time under SBLOCA conditions for ZIRLO and Optimized ZIRLO is > 5,400 sec
- PCT dependent ECR limits as a function of hydrogen based on testing will demonstrate margin to proposed limits
- RIA criteria is conservative with respect to PWR conditions

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BEACON™ Core Monitoring System Addendum 4

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Topics

- Updated Thermocouple Uncertainty Process
- Design Model Methodology
- Fixed Incore Detector Uncertainty Fit

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Updated Thermocouple Uncertainty Process

- Currently licensed method
  - Use thermocouple data saved during flux map processing from previous cycle
- Why improve
  - Thermocouples are maintained during cycle outage
  - Limited number of flux maps available
  - Almost all flux maps are at full power conditions

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Updated Thermocouple Uncertainty Process

- Changed to
  - Use of the current plant/cycle power dependent thermocouple data collected during initial power escalation
- Benefit
  - Significant amount of data
  - Data reflects current cycle thermocouple behavior
  - Provides power dependent thermocouple variability

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Updated Thermocouple Uncertainty Process

- WCAP-12472-P-A uncertainties remain the same with the exception of the functional form of the thermocouple mixing factor variability
- Variability accounts for two effects
  - Increased variability due to the decrease in temperature
  - Changes in cross flow patterns as the power is reduced

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Updated Thermocouple Uncertainty Process

Eq. 4-4 [ ] a, c

Addendum 4 [ ] a, c

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Updated Thermocouple Uncertainty Process

- Compliance:
  - Solve standard deviation equation at full power conditions to get $\sigma^{o}_{T/C}$
  - Use in the approved standard deviation equation

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Design Model Methodology

- The BEACON system will continue to use USNRC approved model methodology
  - PHOENIX-P/ANC
  - PARAGON/ANC
  - NEXUS/ANC
- Future design model updates will not generate a separate BEACON addendum

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Fixed Incore Detector Uncertainty Fit

- Monitoring uncertainties are a function of detector measurement variability and fraction of inoperable detectors
- Addendum 1 / 3:

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- The Addendum 3 uncertainties will be generically used for all fixed in-core detector applications.
- The equations can have more terms depending on the shape and data range needed to bound the limiting uncertainties.

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Material Updates

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AGENDA

- Current and future LTA/LUA programs.
- Vogtle Creep and Growth Program Updates.
- AXIOM™ Alloys

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LTA/LUA Programs

Plant	Alloys Used	Number FA/FR and type	Start	Burnup MWd/kgU	Status Fall 2009

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LTA/LUA Programs

Plant	Alloys Used	Number FA/FR and type	Start	Burnup MWd/kgU	Status Fall 2009

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Vogtle Creep & Growth Test

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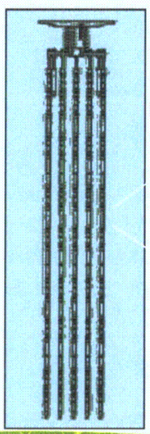
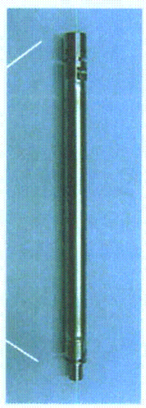
Vogtle Creep & Growth Test - Setup





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Vogtle Creep Capsule Key Results

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Vogtle Creep Capsule Key Results (Cont'd)

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Milestones

- Completed inspections and evaluation of results for cycles 1 to 3.
- Initiated PIE of Four Cycle Test Assemblies A6 and A4
- Initial results for Optimized ZIRLO:
  - Irradiation creep and growth is consistent with previous material.

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AXIOM Alloys

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Introduction

- AXIOM cladding is an advanced zirconium-based cladding material developed by Westinghouse to meet challenging fuel management schemes.
- Currently 4 major variants (X1, X2, X4, and X5, with X5A as another variant of X5, to evaluate effects of manufacturing variations on irradiation performance.)
- New generation of robust alloys
  - Improved corrosion resistance
  - Low hydrogen pickup
  - Lower creep and growth
- An extensive out-reactor characterization program has been conducted on AXIOM alloys.
- AXIOM cladding tubes have been irradiated in multiple commercial reactors and test reactors, with burnups exceeding 70 GWd/MTU.

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AXIOM and ZIRLO nominal chemical compositions in weight % and microstructure from final heat treatment

Alloy	Nb	Sn	Fe	Cr	Cu	V	Ni	Zr
X1	0.7-1	0.3	0.05		0.12	0.2		Bal.
X2	1		0.06					Bal.
X4	1		0.06	0.25	0.08			Bal.
X5	0.7	0.3	0.35	0.25			0.05	Bal.
X5A	0.3	0.45	0.35	0.25				Bal.
ZIRLO	1	1	0.1					Bal.

X1: Low Sn and Cu against corrosion,V for reduced HPU and strength.
X2: Similar to E110 and M5
X4: Similar to X2 but with Cr and Cu for lower corrosion, more robust against abnormal chemistry corrosion and improved strength and weldability.
X5: Modified X5A with lower Sn and higher Nb.
X5A: Previously called Alloy A; a variant of X5 and will be used to support X5 and to evaluate effects of manufacturing variations on irradiation performance.

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Extensive Out-reactor Characterization Program

- Many tests were conducted on the AXIOM alloys before being irradiated in reactors
  - Autoclave corrosion tests, hydrogen pickup
  - Short-term mechanical property evaluation (e.g., tensile properties, elastic properties, weld strength, burst strength),
  - Time-dependent mechanical property determination (creep)
  - Microstructure evaluation
  - Evaluation of physical properties including density, specific heat, thermal expansion, phase transition temperatures and thermal conductivity.

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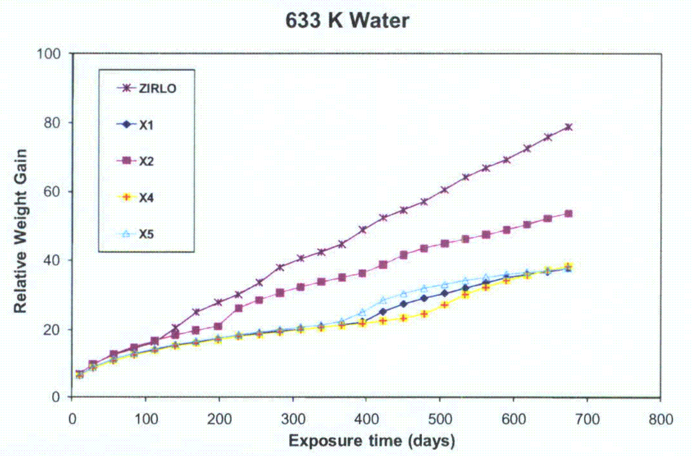
Autoclave Corrosion Testing



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AXIOM Irradiation Programs

Reactor	Alloys	Number of Fuel Assemblies/ Rods	Burnup (GWd/MTU) (October 2010)	Status as of October 2010
B	X1, X2, X4, X5, X5A	Non-fueled samples in thimble tube locations	To ~85	Discharged after 4th cycle, PIE on going
A	X1, X2, X4, X5	2/32	65 to 72	Discharged after 3rd cycle, pool side PIE completed and hot cell PIE planned for 2011
D	X1, X2, X4, X5	4/64	45 to 54	Completed 2 cycles. All-AXIOM LTA in 1st cycle; includes 16 high burn-up AXIOM rods
I	X1, X2, X4, X5	4/72	50 to 56	In 3rd cycle; Recon completed
O	X1, X2, X4, X5	2/64	18	End of 2nd cycle
J	X5A	4/160	58 to 69	Discharged after 3 cycles
H	X5A	4/64	56 to 65	Discharged after 4 cycles

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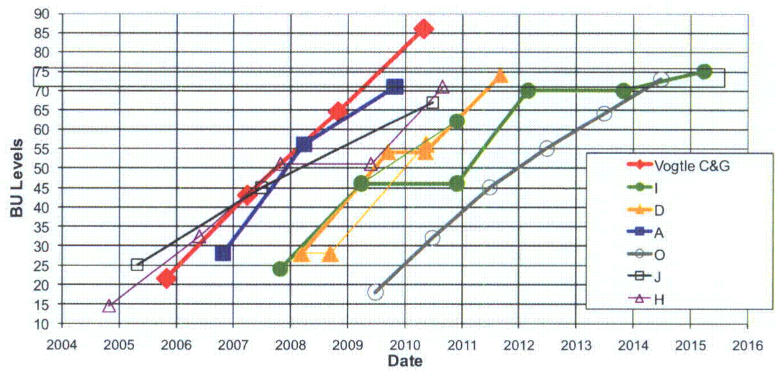
Burnup Plan



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AXIOM - Data Collection

- Oxide (Corrosion) to 70+ GWd/MTU
  - Confirmed low to 70+ GWd/MTU (maximum < 40 μm)
- Hydrogen pick-up up to 70 MWd/kgU
  - Awaiting data to 70+ MWd/kgU from hot-cell PIE
- Rod growth to 70+ MWd/kgU
  - Confirmed less than ZIRLO up to 70+ MWd/kgU
- Irradiation Creep to 70+ MWd/kgU
  - Confirmed less than ZIRLO ~20 and ~40 MWd/kgU
  - Awaiting data to 70+ MWd/kgU.

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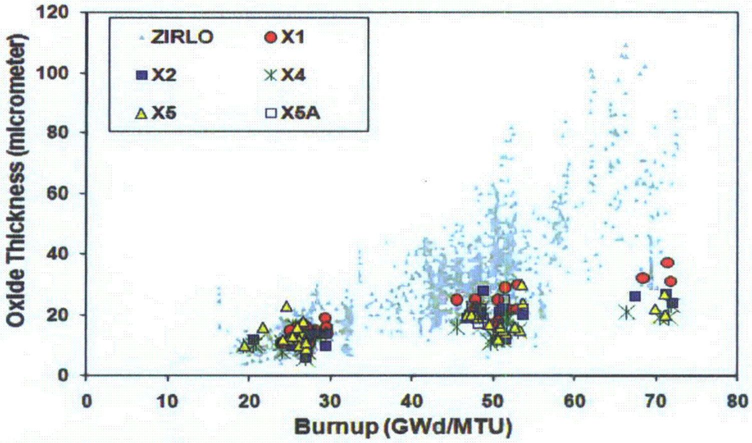
Corrosion Performance of AXIOM Alloys: Oxide Vs. Burnup



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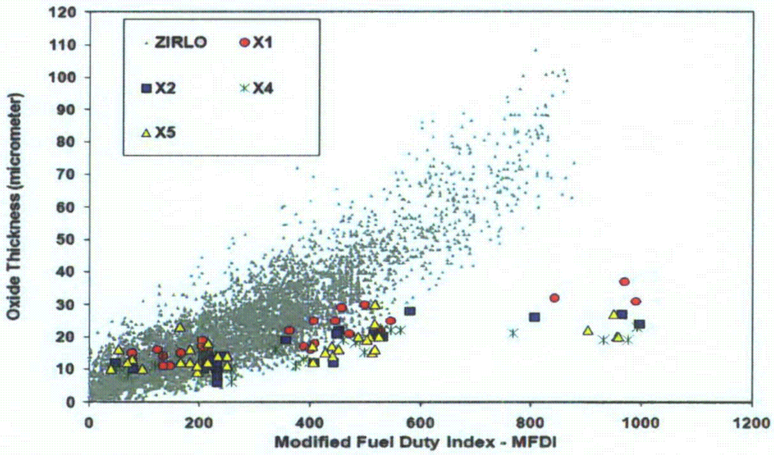
Corrosion Performance of AXIOM Alloys: Oxide Vs. MFDI



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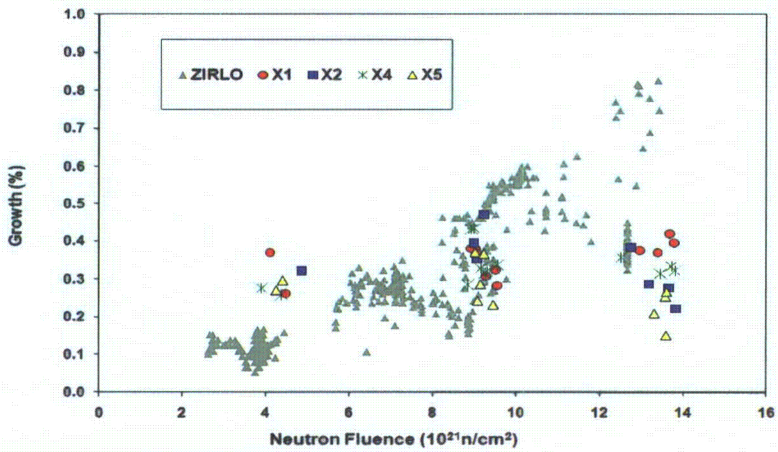
AXIOM Rod Growth



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Relative Comparisons of AXIOM Creep and Growth to ZIRLO alloy after One Cycle of Irradiation of Non-fueled Samples

Alloy	Diameter Change	Free Axial Growth
X1	0.80	0.58
X2	0.65	0.57
X4	0.80	0.51
X5	0.78	0.69
X5A	0.84	0.69
ZIRLO	1.00	1.00

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Summary

- AXIOM alloys have achieved significant in-reactor irradiation with lead rods in multiple commercial PWR plants and test reactor programs
  - High burnup data over 70 GWd/MTU
  - Better in-reactor corrosion performance compared to ZIRLO alloy, especially in high duty operating environments, with oxide thickness less than 40 µm for MFDI up to 1000 and burnups of about 70 GWd/MTU
  - Excellent in-reactor dimensional stability. Less irradiation growth and creep than ZIRLO alloy
- Westinghouse has four major variants that show promise for potential future applications. Testing will continue to identify the most promising candidate for future implementation.

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Optimized ZIRLO™ Status

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Agenda

- Customer Value
- Optimized ZIRLO Performance
- Availability
- Implementation Process
- SER Conditions & Limitations
- LAR Submittals
- Rollout Schedules

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Customer Value of Optimized ZIRLO Fuel Cladding

- Provides improved fuel margins
  - Clad corrosion
  - Rod internal pressure
  - Post-LOCA clad oxidation
  - Rod insertion accidents (RIA)
- By using Optimized ZIRLO fuel cladding, the customer may
  - Increase peaking factors and reduce the number of feed assemblies
  - Enable plant uprating
  - Extend cycle length
  - Operate to higher rod burnup levels

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Optimized ZIRLO Cladding Experience

- Used in 16 units, worldwide
- More than 105,000 fuel rods
- High burn-up experience to above 70 MWd/kgU
- Significant improvement to ZIRLO® cladding experience with regard to corrosion
- Licensed in the USA
- First full reloads in 2008

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Optimized ZIRLO Performance

- Reached burnups of ~74 GWd/mtU
- Reached Fuel Duty (MFDI) > 1000
- Optimized ZIRLO cladding has:
  - 40% lower fuel rod corrosion.
  - Similar fuel rod cladding creep.
  - Similar PCI failure threshold.
  - Similar high temperature oxidation kinetics and post quench ductility (PDQ) behavior.

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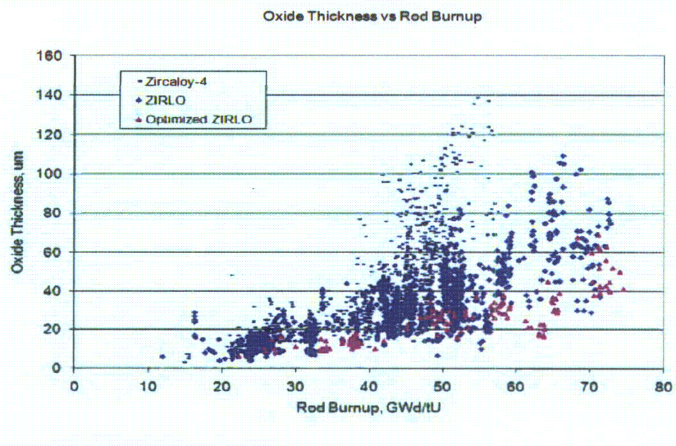
Zircaloy-4, ZIRLO and Optimized ZIRLO Cladding Corrosion Trends



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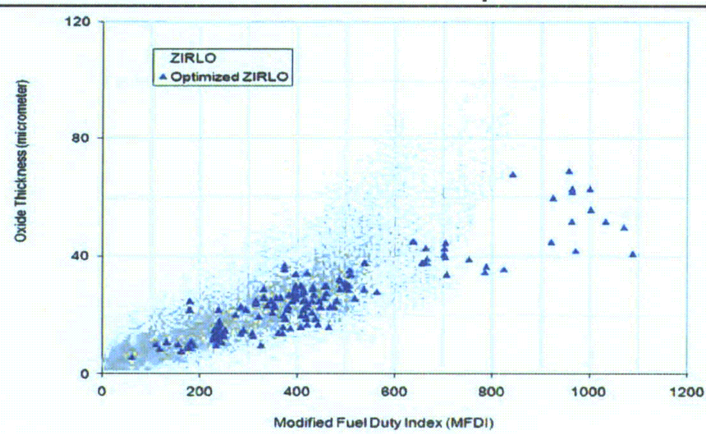
Corrosion Trend with Respect to MFDI



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Major Milestones

- Plant A Optimized ZIRLO clad fuel rods reached a peak rod burn-up level of ~74 MWd/kgU
  - Shipped rods to hot cell
  - Hot Cell Exam Report: March 2012.
- April 2011: Approval of WCAP-12610-P-A & CENPD-404-P-A, Addendum 2-P, ZIRLO and Optimized ZIRLO Corrosion Model.
- April 2011: Provide updated Vogtle data and 63 MWD/kgU Calvert Cliffs-1 data in final Optimized ZIRLO data package to NRC.
- June 2011: Submit ZIRLO and Optimized ZIRLO PQD data to NRC.

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Availability of Optimized ZIRLO Fuel Cladding

- Substantial data base exists for Optimized ZIRLO fuel cladding
  - Robust and ongoing irradiation test program
  - Three cycles of successful LUA operation at three U.S. plants
  - Multi-region operation at two CE-NSSS plants
- Westinghouse is offering Optimized ZIRLO fuel cladding for all of our fuel product lines
  - W14, W15, W16, W17, CE16

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Optimized ZIRLO Implementation Process

- Provide customer with licensing requirements
  - WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A, July 2006
  - White paper on the implications of the ten Conditions & Limitations (C&Ls) in the SER
  - Westinghouse Letters demonstrating SER compliance in response to C&Ls 6 and 7
- Prior to start of reload design, conduct Challenge Board Review
  - Customer focused review of implementation plan
  - Board consists of customer and Westinghouse members
  - Reload design team presents plans for implementation
  - Addresses engineering, licensing and manufacturing schedules

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SER Conditions & Limitations

- Exemption request required for current 10CFR50.46.
  - Submitted by Licensee
  - Westinghouse provides LAR template, modified to customer needs
- Additional data submittal requirements
  - Westinghouse provides to NRC annually and notifies customers. Final letter to NRC will be issued this year satisfying Conditions 6 and 7. Westinghouse would like the NRC to formally acknowledge that Conditions 6 and 7 have been satisfied.

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SER Conditions & Limitations (Cont'd)

- Meet existing standards and requirements
  - 62 GWd/MTU lead rod burnup for Westinghouse fuel designs and 60 GWd/MTU for CE fuel designs
  - All standards & methodologies applicable to ZIRLO cladding
- Meet modified standards and requirements
  - Maximum fuel waterside corrosion
  - Decreased Yield and Ultimate Tensile Strength
  - If using LOCBART or STRIKIN-II, may need to re-evaluate
  - Reduced PCT limit during the Locked Rotor Event

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Estimated Optimized ZIRLO Clad Usage by Year

- U.S. Plants
  - Spring 2011 - Four plants
  - Fall 2011 – Two plants
  - 2012 – Ten plants
  - 2013 – Eight plants
  - 2014 – Four plants
- European Plants
  - 2012 – Three plants
  - 2013 - Two plants

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High Burnup Program

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Agenda

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High Burnup Program - Objective

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High Burnup Program – Overall Strategy

[ ]a,c

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High Burnup – Key Aspects

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Near Term PWR Program Definition

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[ ]a,c High Burnup Project

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High Burnup Program - Status

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[ ]a,c High Burnup Project - Milestones

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Westinghouse Licensing Schedule

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Summary

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Spent Fuel Pool Criticality Analysis

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Agenda

- Draft Interim Staff Guidance (ISG) discussion
- Methodology Update
  - Computer Code Validation
  - Fuel Geometry Effects
  - Uncertainty Treatment
- Update on current NRC work

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Draft ISG – Overall impressions

- New issues were identified with no further guidance (control rod insertion, justification of NUREG/CR-6801, efficiency of neutron absorber)
- Overly conservative requirements (interface analysis, application of depletion uncertainty)
- Vague wording leaves issues open to intrepretation in the future (control rod insertion licensee controls, validation of fission products)

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Draft ISG – Section 2. Depletion Uncertainty

- Current wording (2.a.ii) requires reactivity decrement to be calculated based on difference between fresh fuel assembly and fuel assembly at burnup of interest "*either with or without residual integral burnable neutron absorber, whichever results in the larger reactivity decrement.*"
- This can be corrected by deleting "whichever results in the larger reactivity decrement"

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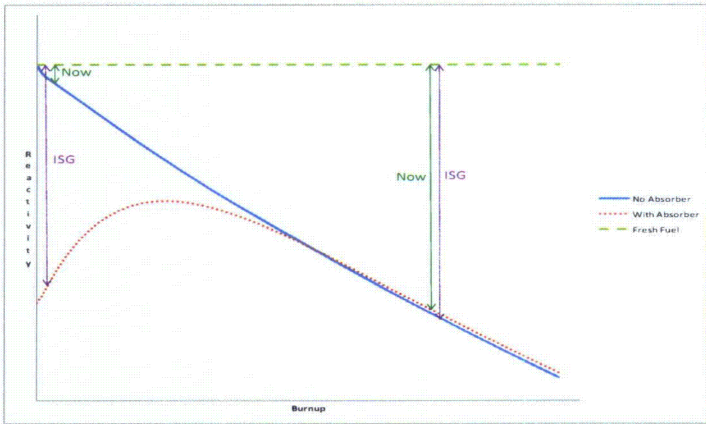
Draft ISG – Depletion Uncertainty



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Draft ISG – Neutron Absorber Efficiency

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Draft ISG – Fission Product Treatment

- No clear guidance provided on how to address inclusion and validation of the fission products
- Fission product credit has been approved multiple times over the past ten years
- Initial Oak Ridge work confirms that the fission product uncertainty is small, consistent with previous industry presentations to NRC
- The depletion uncertainty inherently covers the fission product uncertainty

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Depletion Parameters

- Moderator Temperature

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- Fuel Temperature

[ ]a,c

- Soluble Boron Concentration
  - Maximum cycle average
- Relative Power
  - Assembly average
- Burnable Absorbers/Rodded Operation

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Methodology Update – Fuel Geometry Effects (Due to Depletion)

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Methodology Update – Validation

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Methodology Update – Uncertainties

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Methodology Update – Uncertainties (cont.)

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Ongoing NRC/ORNL Work

- Industry should have an opportunity to provide comment on Oak Ridge work prior to issuance as a NUREG.
- Schedule?

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References

1. A. H. Wells, et al., "Criticality Effect of Neutron Streaming Between Boron Carbide Granules in Boral for a Spent Fuel Shipping Cask," Trans. Am. Nucl. Soc., 54, 205 (1987)
2. S. E. Turner, "Reactivity Effects of Streaming Between Discrete Boron Carbide Particles in Neutron Absorber Panels for Storage or Transport of Spent Nuclear Fuel," Nucl. Sci. Eng., 151, 344 (2005).
3. S. E. Turner, T. G. Haynes III, "Analytical and Experimental Investigation of Recent Challenges to Neutron Attenuation and the Relationship with Criticality Analyses," Rad. Prot., 169, 195 (2010).
4. M. D. DeHart, *Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages, ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National* Laboratory, May 1996.

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AP1000® Core Reference Report Update – Event Categorization

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Overview

- Core Reference Report Schedule
- Updates to the SRP

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Core Reference Report Proposed Time Line

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Core Reference Report Contents

- Fuel and Core Design evolutions that will be addressed in the Core Reference Report:

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Core Reference Report Format

1. Introduction and Summary
2. Fuel Mechanical Design Features
3. Nuclear Design
4. Thermal and Hydraulic Design
5. LOCA / Non LOCA Transient Analyses

Appendices

- A. Summary and Justification for Technical Specification Changes
- B. Markups to DCD leading to FSAR Chapters 4, 15 and 16
- C. Completed Sections without markups
- D. Sample COLR

Will follow format of previously submitted fuel update reports (e.g., Power Uprates, Reload Transition Safety Reports (RTSRs), etc.)

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Background

- During presentation for the Planning Meeting for the Core Reference Report on September 8th , the NRC asked Westinghouse if Chapter 15 would be updated to the latest Standard Review Plan (SRP) Revision (Rev. 3).
- The original plan for the Core Reference Report was to perform Chapter 15 to the current DCD version of the analyses (Rev. 2 of the SRP).
- Westinghouse was asked to review the latest SRP revision with respect to Chapter 15 and determine whether it was appropriate to change or provide information as to why there was no need to change to the current revision.

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Review of SRP Revisions

- Review of the Rev. 3 SRP indicates the major difference as:
  - Change from:
    - ANSI 18.2 classification of plant conditions
      - Condition I: Normal operation and operational transients
      - Condition II: Faults of Moderate Frequency
      - Condition III: Infrequent Faults
      - Condition IV: Limiting Faults
  - To:
    - Anticipated Operational Occurrences (AOO)
    - Postulated Accidents

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Review of SRP Revisions

- The new categorization is similar to the ANSI 18.2 for the following:
  - AOOs have same criteria as Condition II Events.
  - Postulated Accidents have the same criteria as the Condition IV events.
- The difference is related to the Condition III Events and how their specific criteria compares against the current DCD/FSAR criteria.

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Review of SRP Revisions

- The Condition III Events from the DCD are:
  - Minor steam line break
  - Complete loss of flow
  - Single RCCA withdrawal
  - Inadvertent loading and operation of a fuel assembly in an improper position
  - Inadvertent opening of the automatic depressurization system (ADS)
  - Small Break loss of coolant accidents (SBLOCAs)
  - Radiological release events in Chapter 15.7 (not Fuel Handling)

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Evaluation

- Minor steam line break & complete loss of flow
  - Rev. 3 of SRP classifies as AOOs
  - DCD criteria for steam line break and all loss of flow events are the same as those for Condition II events
  - Therefore meet the conditions for AOO also

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Evaluation

- Single RCCA withdrawal
  - SRP Rev. 3 classifies single rod withdrawal as AOO for the events that are result of a SINGLE failure (electrical or mechanical)
  - DCD 15.4.3 states **"No single electrical or mechanical failure in the system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation."**
  - Therefore the 5% failed fuel limit/radiological consequences is applicable to a postulated accident

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Evaluation

- Inadvertent loading and operation of a fuel assembly in an improper position
  - SRP Rev. 3 acceptance criteria is radiological
  - DCD states: **"In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects are either readily detected by the online core monitoring system or cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes."**
  - Therefore no radiological consequences and meets intent of SRP

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Evaluation

- Inadvertent opening of the automatic depressurization system (ADS)
  - This event is similar to inadvertent opening of the pressurizer safety valve
  - Although event not specifically called out in SRP – assumed it would be considered an AOO (for initial portion of the event)
  - Analysis in DCD 15.6.1 was performed to meet the Condition II criteria
  - Therefore meet the conditions for AOO also

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Evaluation

- Small Break loss of coolant accidents (SBLOCAs)
  - SRP criteria is to meet 10CFR50.46 criteria
  - DCD analysis meets 10CFR50.46 criteria
  - Therefore meet the conditions for the revised SRP

- Radiological release events in Chapter 15.7 (not Fuel Handling)
  - As stated in DCD, Sections 15.7.1 and 15.7.2 no longer in the SRP
  - SRP not updated for remainder of the events
  - Therefore no change to this section

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Conclusions

- Chapter 15 acceptance criteria are consistent with respect to Rev. 3 of the Standard Review Plan
- An explanation of why the ANSI 18.2 Conditions can remain in Chapter 15 will be added to the Core Reference Report

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15

BWR Fuel Performance Update

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Outline

- Fuel Technology Program
  - Cladding Performance
  - Channel Performance
  - Fuel Reliability
  - Water Chemistry Experience
  - Optima3 [ ]^{a,c} Fuel Inspection Results
  - CR 99 Control Rod Blade Follow Up Program
- Conclusions

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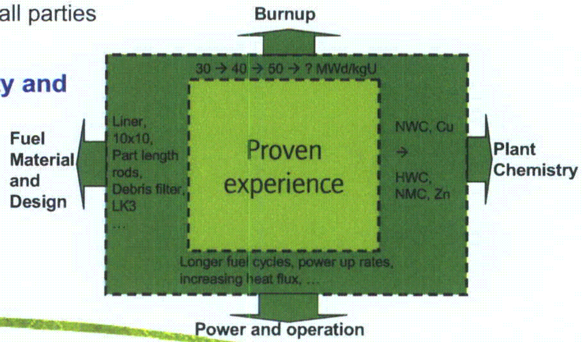
Fuel Technology Program

- Experience exchange between utilities and Westinghouse to verify fuel performance
  - Expand the box of proven experience
  - Sharing data which results in large benefits for all parties

→ **Improved availability and reliability!**



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10x10 BWR Fuel Experience

a,c

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Outline

- Fuel Technology Program
  - Cladding Performance
  - Channel Performance
  - Fuel Reliability
  - Water Chemistry Experience
  - Optima3 [ ]a,c Fuel Inspection Results
  - CR 99 Control Rod Blade Follow Up Program
- Conclusions

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Cladding Performance

LK3 – 2010 Measurements

[ ]a,c

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Cladding Development

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Cladding Development

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End Plug Material Change

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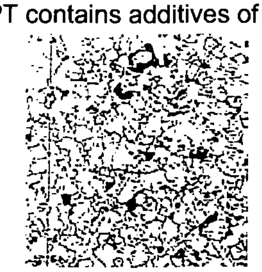
ADOPT™ Pellet Overview

- High density UO_2 pellets with enlarged grain size.
- ADOPT contains additives of Cr- and Al-oxide.



Standard pellet

ADOPT pellet

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ADOPT – In Pile Behavior

– Reduced fission gas release

[ ]a,c

– PCI behavior, positive indications from [ ]a,c

– Secondary degradation behavior, positive indications from [ ]a,c

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ADOPT Experience

[ ]a,c

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ADOPT Verification

[ ]a,c

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Outline

- Fuel Technology Program
  - Cladding Performance
  - Channel Performance
    - [ ]a,c ZIRLO Performance
  - Fuel Reliability
  - Water Chemistry Experience
  - Optima3 [ ]a,c Fuel Inspection Results
  - CR 99 Control Rod Blade Follow Up Program
- Conclusions

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Channel Performance

Westinghouse Channel Growth

[ ] a,c

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Channel Performance

Westinghouse Channel Bow

[ ] a,c

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Channel Performance

Westinghouse Channel Corrosion

[ ]a,c

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[ ]a,c ZIRLO Channel LTA Program

[ ]a,c

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Future [ ]a,c ZIRLO Deliveries

[ ]a,c

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[ ]a,c ZIRLO Channel Inspection Plans

[ ]a,c

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Outline

- Fuel Technology Program
  - Cladding Performance
  - Channel Performance
  - Fuel Reliability
  - Water Chemistry Experience
  - Optima3 [ ]a,c Fuel Inspection Results
  - CR 99 Control Rod Blade Follow Up Program
- Conclusions

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Leakers for Modern Westinghouse BWR Fuel (Liner)

[ ]a,c

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BWR Fuel Performance

[ ]a,c

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Outline

- Fuel Technology Program
  -
  -
  -
  - Water Chemistry Experience
  - Optima3 [ ] Fuel Inspection Results
  - CR 99 Control Rod Blade Follow Up Program
- Conclusions

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Water Chemistry Experience

Diagnostics – Plant Status

[ ]a,c

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Outline

- Fuel Technology Program
  - Cladding Performance
  - Channel Performance
  - Fuel Reliability
  - Water Chemistry Experience
  - Optima3 [ ]a,c Fuel Inspection Results
  - CR 99 Control Rod Blade Follow Up Program
- Conclusions

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SVEA-96 Optima3
Previous Experience for the Fuel Design

[ ]a,c

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SVEA-96 Optima3
Previous Experience for the Fuel Design (cont'd)

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Previous Experience for the [ ]a,c ZIRLO Channel Material

[ ]a,c

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SVEA-96 Optima3 Fuel Inspection in [ ]a,c 2010

[ ]a,c

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SVEA-96 Optima3 Fuel Inspection in [ ]a,c 2010

- SVEA-96 Optima3 fuel design

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Fuel Inspection in [ ]a,c 2010 Visual Inspection

[ ]a,c

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[ ]a,c ZIRLO Channel Experience with Instrumentation Tube Presence

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Fuel Inspection in [ ]a,c 2010 Visual Inspection

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Fuel Inspection in [ ]a,c 2010
Visual Inspection

[ ]a,c

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Fuel Inspection in [ ]a,c 2010
Rod Growth Measurements

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Fuel Inspection in [ ]a,c 2010
Rod Growth Measurements

[ ]a,c

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Fuel Inspection in [ ]a,c 2010
Channel Growth Measurements

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Fuel Inspection in [ ]a,c 2010
Channel Bow Measurements

[ ]a,c

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Future Inspection Plans

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Outline

- Fuel Technology Program
  - Cladding Performance
  - Channel Performance
  - Fuel Reliability
  - Water Chemistry Experience
  - Summary] Fuel Inspection Results
  - CR 99 Control Rod Blade Follow Up Program
- Conclusions

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BWR Control Rod Blade Products

[ ]a,c

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CR 99 Absorber Material Outline

[ ]a,c

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Evolutionary CR 99 Development

[ ]a,c

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CR 99 3rd Generation

- 3rd generation CR 99 – Launced 2007
  - Performance objective: IASCC threshold > [ ][a,c]

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Conclusions

- BWR Fuel Performance
  - [ ][a,c] ZIRLO channel performance is within expectations
  - Debris fretting remains the only fuel failure mechanism for 10x10 liner fuel
- SVEA-96 Optima3 LTA in [ ][a,c] is at leading burnups
  - Future fuel surveillance will continue to verify its performance
  - Fuel performance within expectations
  - Spacers show expected corrosion resistance
  - Rod and channel growth as well as channel bow within experience
- CR 99 High Performance Control Rod Blade
  - Evolutionary development step by step
  - Design up to [ ][a,c] supported by follow up program

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Optima3 Update

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Outline

- Optima3 Update
  - Optima2 Overview
  - Optima3 in Comparison to Optima2
  - TripleWave+
  - D5 Correlation
  - Optima3 Verification
  - Optima3 Implementation Plan
- Summary

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2

SVEA-96 Optima2 – Key Features

[ ] a,c

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SVEA-96 Optima3

[ ] a,c

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SVEA-96 Optima3

- Modifications of SVEA-96 Optima2 for

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SVEA-96 Optima3 Sub-Bundle Bottom End

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SVEA-96 Optima3 Sub-Bundle Top End

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SVEA-96 Optima3 – Space Gain

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SVEA-96 Optima3 Spacer with Mixing Vanes

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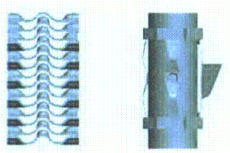
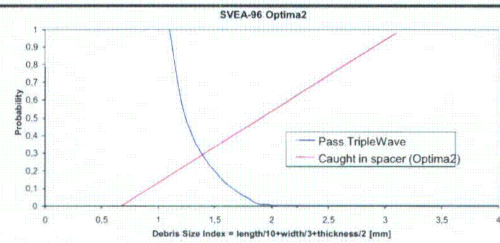
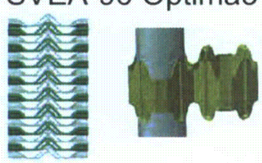
Debris Fretting Mitigation

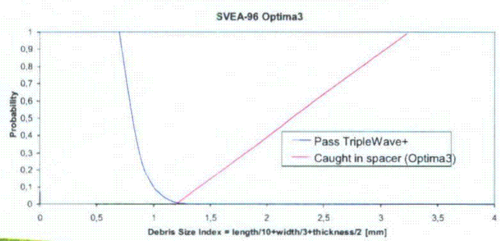
SVEA-96 Optima2





SVEA-96 Optima3





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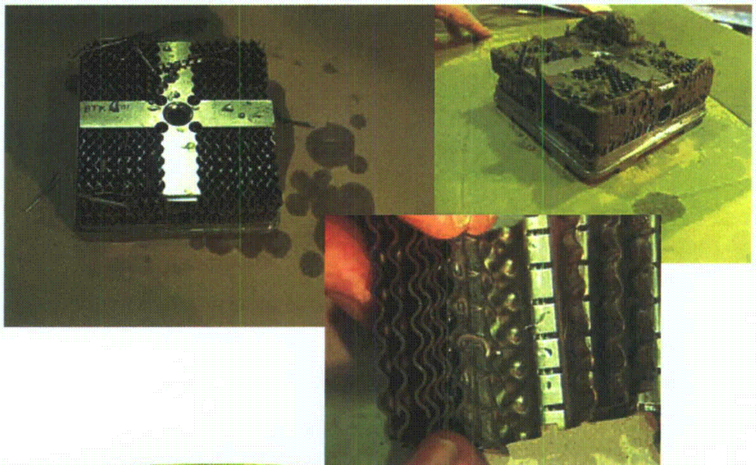
TripleWave+ Testing



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Optima3 Experimental Database

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Measured dryout performances at FRIGG

Optima2 versus Optima3 optimized critical power

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Measured dryout performances at FRIGG

Optima2 versus Optima3 optimized critical power

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D5 dryout correlation form (sub-bundle)

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SVEA-96 Optima3 - Verification

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Implementation of Optima3 in the U.S.

[ ] a,c

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Conclusions

- Westinghouse BWR Operational Experience
  - Optima2 has demonstrated excellent fuel reliability
  - Optima3 is an evolutionary design building upon the already proven performance of Optima2 to further improve fuel reliability and in particular debris protection

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BWR Cladding Hydrogen Model

Westinghouse/NRC Fuel Update Meeting
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February 15, 2011

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Westinghouse Non-Proprietary Class 3

Contents

- Background
  - Current Licensed Hydrogen Model
  - Requirements on RIA and LOCA

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2

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Current Licensed Hydrogen Model

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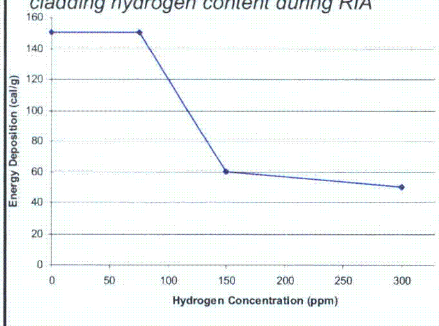
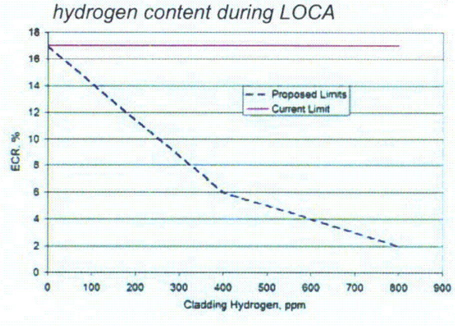
Requirements on RIA and LOCA

Max allowed delta enthalpy vs. pre-transient cladding hydrogen content during RIA



Max allowed ECR vs. pre-transient cladding hydrogen content during LOCA



New requirements call for a need to model hydrogen content continuously during the life time of the cladding, e.g., as a function of burnup

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Contents

- Westinghouse BWR Cladding
  - Evolutionary development
  - Cladding performance in general
  - Hydrogen data

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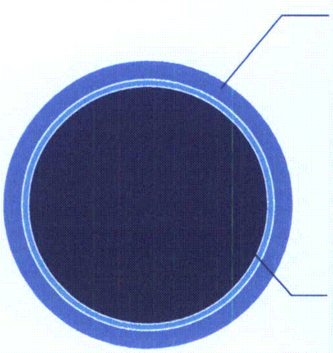
Westinghouse BWR Liner Cladding



Outer Component:
- **Zircaloy-2**
- **Current standard cladding LK3 is optimized for:**
  - **Corrosion resistance**
  - **Low hydrogen pick-up**
  - **High burnup performance**

Liner Component
- **ZrSn alloy with tailored Fe**
- **10% of wall thickness**
- **Metallurgically bonded to outer component**

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Evolutionary Development

LK = Low Corrosion (Låg Korrosion in Swedish)

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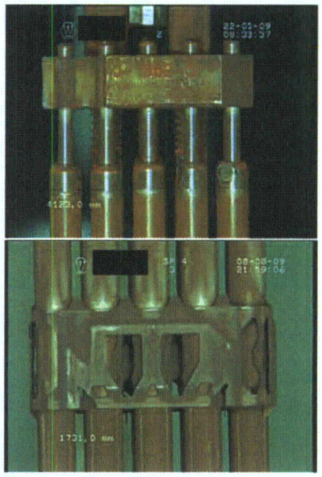
Cladding Performance in General

<u>On-site verification</u>

- Visual inspection
- Cladding oxide measurement
- Rod growth measurement

Extensive on-site database and data collection programs to verify continued stable fuel performance



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On-site Oxide Measurements

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On-site Rod Length Measurements

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LK3/Liner Cladding Hydrogen Database

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Contents

- Modeling
  - Hydrogen pick-up mechanistics
  - FRAPCON 3.4 hydrogen model

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Hydrogen pick-up mechanistics

- Two phases of hydrogen pick-up:
  - *Prohibited pickup*
  - *Accelerated pickup*

    SPP dissolution → corrosion and hydrogen pick-up acceleration

Control of SPP size and frequency is key to extending Zry-2 materials' life time in reactor

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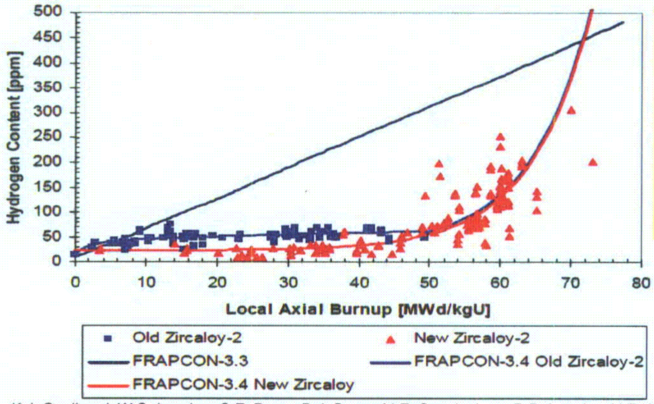
FRAPCON 3.4 Hydrogen Model



New-Zircaloy-2 includes data from Westinghouse LK3/L cladding as well as from other vendors

K.J. Geelhood, W.G. Luscher, C.E. Beyer, D.J. Senor, M.E. Cunningham, D.D. Lanning, H.E. Adkins
Predictive Bias and Sensitivity in NRC Fuel Performance Codes
NUREG/CR-7001, October 2009

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LK3/Liner cladding applicability to FRAPCON 3.4 hydrogen model

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Conclusions

- LK3/L cladding has a stable, well monitored in-reactor performance

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ABWR Fuel Licensing Update

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Objectives

- The currently U.S.-licensed methodology for fuel and fuel safety analysis is for application to reload analyses and fuel vendor transition

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Changes Introduced by ABWR and Full-scope Fuel Related FSAR Applications

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ABWR Fuel Licensing Submittals

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ABWR Fuel Licensing Status

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ABWR Fuel Licensing Strategy

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ODEN DNB Test Facility

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Outline

- Background
- ODEN overview
- ODEN vs. HTRF capabilities
- Qualification test program
- Summary

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Background
Closure of Previous Test Facility - HTRF

- Columbia University **H**eat **T**ransfer **R**esearch **F**acility (HTRF)
  - –Opened 1951
  - –Closed December 2003
  - –Total 30,000⁺ DNB points, 513 tests
    - – 42% from Westinghouse, CE, BNFL
  - –Source of all Westinghouse DNB data for licensed DNB correlations
- Westinghouse secured critical HTRF components

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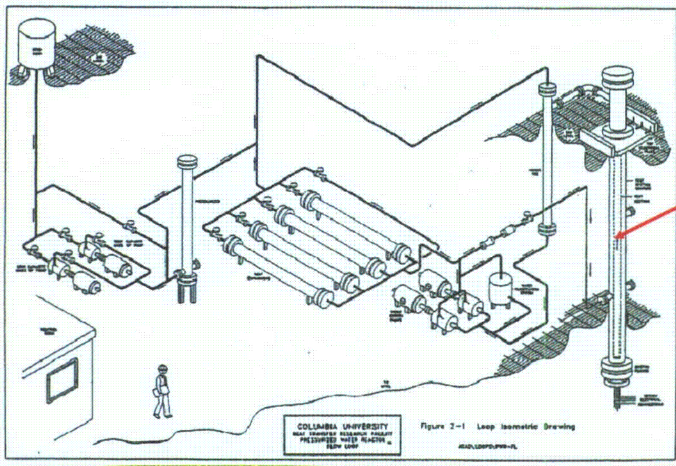
HTRF Loop - Schematic



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ODEN Loop - Overview

- Objectives
  - –Replace HTRF with in-house capability for DNB testing PWR conditions
- Location
  - –Co-located with "FRIGG" (existing BWR loop)
  - –Westinghouse T/H Test Facility (Västerås, Sweden)

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- Based on HTRF

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ODEN Loop - Evolution

Milestone	**Date**
Program Start	[ ] a,c
Facility Design	
Loop Installation	
First DNB points	
Shakedown Testing Complete	
Qualification Testing Complete	
Data Analysis / Documentation	

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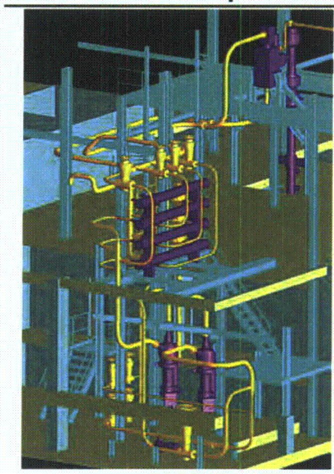
ODEN Loop - Characteristics



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ODEN Loop - Schematic

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ODEN vs. HTRF – Loop Capability

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ODEN vs. HTRF – Test Capability

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ODEN – Primary Measurement Systems

- Test Section Measurements (safety-related)
  - Tin, Tout
  - Pout
  - Flow
  - Bus-Bus Voltage, Heated Length Voltage, Current → Power
- Characteristics
  - Primary and redundant (backup) measurements per parameter
  - Calibration traceable to Swedish standards laboratory (SP)
  - Uncertainties consistent with HTRF

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ODEN - Data Acquisition System (DAS)

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ODEN – Quality Assurance Program

- ODEN QA program conforms to Westinghouse Quality Management System (QMS), which is compliant with
  - 10 CFR 50, Appendix B
  - ASME NQA-1 (1994)
  - ISO 9001

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ODEN – Qualification Program

Objectives:

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ODEN - Qualification Program

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ODEN – Qualification Test Geometry

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ODEN – Qualification Test Geometry

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ODEN – Qualification Test Conditions

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ODEN vs. HTRF – Preliminary Results

- DNB data
  - agrees well with HTRF
  - good ODEN-ODEN repeatability
- Heat balance -- consistent with HTRF

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ODEN - Summary

- Replaces HTRF as Westinghouse DNB test facility
- Integrated with the Westinghouse Västerås T/H Test Facility in Sweden
- Similar capacity and capability for DNB testing as HTRF
- Qualification testing completed
- Preliminary data analysis shows good agreement to HTRF
- Verification of data and documentation is in progress

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CASL: The Consortium for Advanced Simulation of Light Water Reactors

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Outline

- Background and Purpose
- CASL Program Overview
- CASL Objective to Develop Advanced Simulation Tools
- Proposed Vision of Virtual Reactor (VR)
- Summary

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CASL
Consortium for Advanced Simulation of LWRs

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Background and Purpose

- A requirement of the DOE solicitation for the Nuclear Hub was to engage NRC in review of the "Virtual Reactor" tool.
- Early interfacing will be done primarily through the existing MOU between EPRI and RES
- An introductory meeting was held in October with RES personnel

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Background and Purpose

- At this meeting, provide brief summary of current CASL Advanced Tool Development
- Plans for first 5 years are to primarily engage with RES for review and advice on the tool development, while keeping other NRC branches informed
- Plans for second 5 years are to primarily engage with NRR for Westinghouse licensing of selected CASL development tool applications

The CASL Team:
A unique lab-university-industry partnership

Core partners

Oak Ridge National Laboratory
Electric Power Research Institute
Idaho National Laboratory
Los Alamos National Laboratory
North Carolina State University
Sandia National Laboratories
Tennessee Valley Authority
University of Michigan
Massachusetts Institute of Technology
Westinghouse Electric Company

Building on longstanding, productive relationships and collaborations to forge a close, cohesive, and interdependent team that is fully committed to a well-defined plan of action

Individual contributors

ASCOMP GmbH
CD-adapco, Inc.
City University of New York
Florida State University
Imperial College London
Rensselaer Polytechnic Institute
Southern States Energy Board
Texas A&M University
University of Florida
University of Tennessee
University of Wisconsin
Worcester Polytechnic Institute

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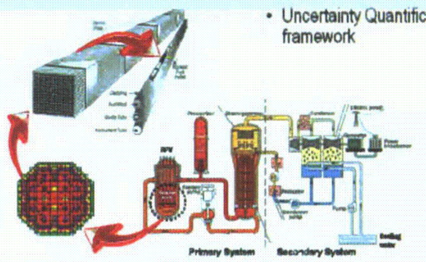
CASL Vision:
Create a Virtual Reactor for predictive simulation of LWRs

Leverage
- Current state-of-the-art neutronics, thermal-fluid, structural, and fuel performance applications
- Existing systems and safety analysis simulation tools

Develop
- New requirements-driven physical models
- Efficient, tightly-coupled multi-scale/multi-physics algorithms and software with quantifiable accuracy
- Improved systems and safety analysis tools
- Uncertainty Quantification framework

Deliver
- An unprecedented predictive simulation tool for simulation of physical reactors
- Architected for platform portability ranging from desktops to DOE's leadership-class and advanced architecture systems (large user base)
- Validation basis against 60% of existing U.S. reactor fleet (PWRs), using data from TVA reactors
- Base M&S LWR capability



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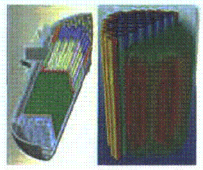
CASL mission:

Develop and apply the VR to address 3 critical performance goals for nuclear power

1. **Reduce capital and operating costs** per unit energy by:
   - Power uprates
   - Lifetime extension
2. **Reduce nuclear waste** volume generated by enabling higher fuel burnups
3. **Enhance nuclear safety** by enabling high-fidelity predictive capability for component and system performance from beginning of life through failure



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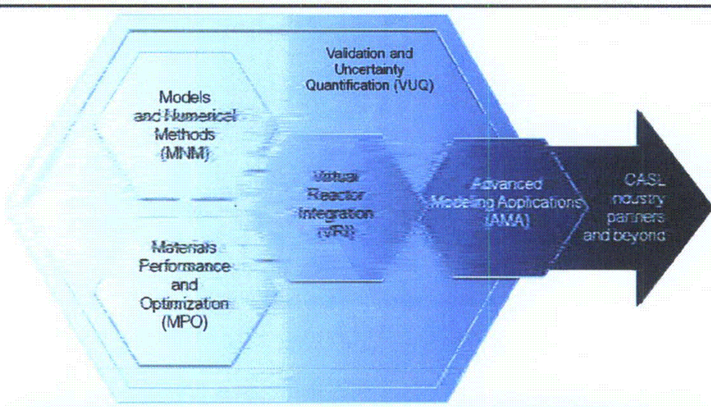
CASL Focus Areas



AMA will drive the Virtual Reactor development

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CASL Challenge Problems:

Based on real-world industry experience and goals

- Industry goals will challenge fuel and plant performance:
  - Power uprates
  - Higher burnup
  - Life extension
- We evaluated detailed safety, operating, and design criteria to determine key phenomena that limit reactor performance
- Challenge problems tackle these key phenomena

Core CASL objective:
Develop advanced M&S methods and investigate new fuel designs to address challenge problems

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Key Challenge Problems Limiting Reactor Performance

	Power uprate	High burnup	Life extension
Operational			
CRUD-induced power shift (CIPS)	×	×	
CRUD-induced localized corrosion (CILC)	×	×	
Grid-to-rod fretting failure (GTRF)		×	
Pellet-clad interaction (PCI)	×	×	
Fuel assembly distortion (FAD)	×	×	
Safety			
Departure from nucleate boiling (DNB)	×		
Cladding integrity during loss of coolant accidents (LOCA)	×	×	
Cladding integrity during reactivity insertion accidents (RIA)	×	×	
Reactor vessel integrity	×		×
Reactor internals integrity	×		×

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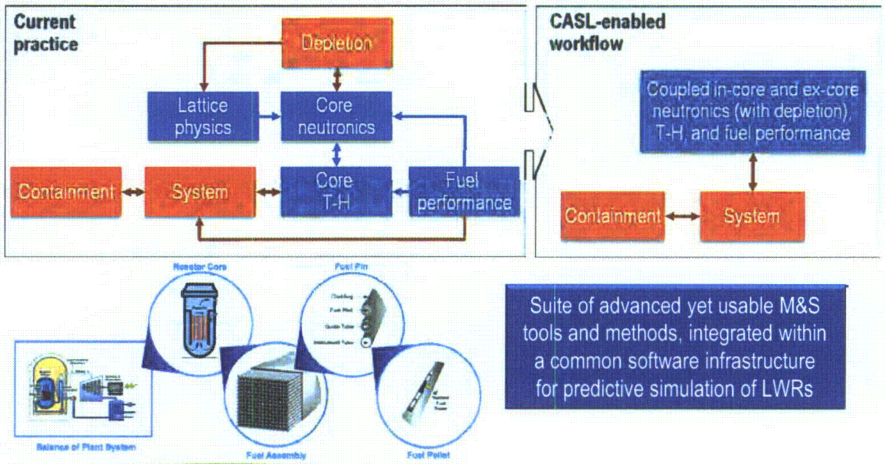
CASL VR is at the heart of the plan and is the science and technology integrator



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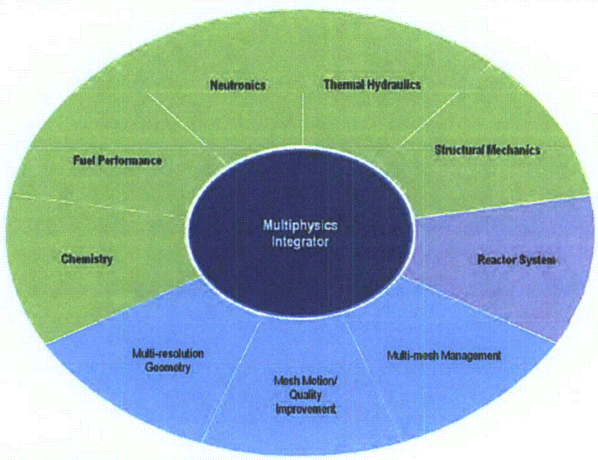
Use Light weight Integrating Multiphysics Environment (LIME) Tool to Couple Codes



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Coupling Existing Tools to Help Develop Advanced Tools

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Proposed 3x3 Modeling to Help Develop New Model and Advanced Tools

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3x3 Pin Bench CAD and CFD Models

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3x3 Neutronics and CFD Coupling

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3x3 Advanced Pin Modeling

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17x17 Neutronic/CFD Modeling

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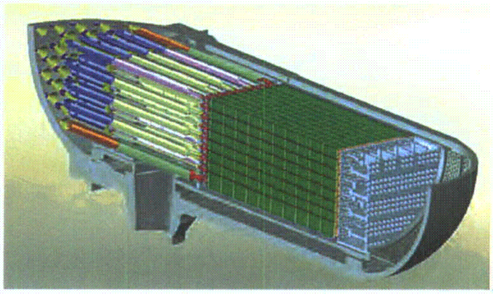
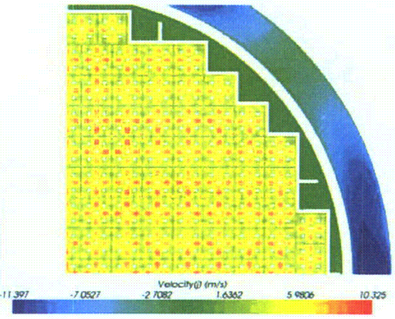
Expand from 3x3 to Vessel CFD Modeling



3D CAD Models for Vessel



Perform detailed pin CFD modeling and two-phase flow

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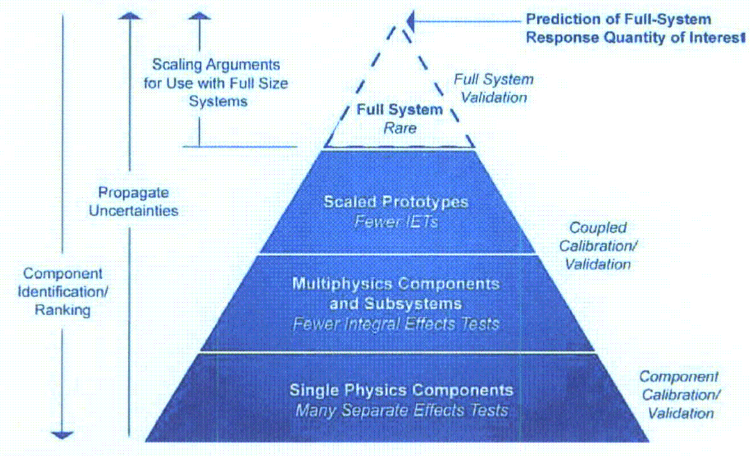
Validation Hierarchy



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Summary

- Advanced Modeling and Simulation will help us address Key Challenge Problems for Power Uprates, High Burnup & Life Extension
- Coupling Existing Tools will guide us in developing the advanced ones
- We have an opportunity to develop and validate one multi-physics tool starting with the 3 x 3 Pin Model
- The 3 x 3 Pin model then can be expanded to the entire vessel to create the Virtual Reactor with validation
- Will keep NRC up to date on the progress of CASL development

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CASL Consortium for Advanced Simulation of LWRs 21

Westinghouse Fuel Performance Update Meeting

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 15, 2011

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Westinghouse Fuel Performance Update Meeting
Agenda
February 16, 2011

Wednesday, February 16, 2011 (Westinghouse and NRC)

8:00 a.m. Westinghouse Current Organization

Topical Report Update

Issues the NRC would like to raise

General Licensing Topics
- Concurrent Reviews
- Scheduling Issues
- GSI-191
- NRO Audit in Sweden
- NRR/NRO Review of Documents
- 50.46 Rule Making and Implications
- EPU/Spent Fuel Pool Criticality Relationship

Noon Adjourn

Westinghouse/NRC Interfaces

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 16, 2011

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Westinghouse/NRC Organizational Interfaces

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Topical Report Schedules

Westinghouse/NRC Fuel Update Meeting
Rockville, MD
February 16, 2011

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PWR Topical Report Schedule

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AP1000® Topical Report Schedule

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BWR and ABWR Topical Report Schedule

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BWR and ABWR Topical Report Schedule

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