November 5, 2002

Bob,

As requested, here is a redacted version of the presentation materials that we used in our meeting with NRC on May 23, 2002. As noted in Mike Tuckman's letter of August 15, 2002, the proprietary material is contained in Tabs 5, 6, 7, and 8. This Tuckman letter also contains the affidavit with supporting justification that supports withholding the information.

If you have any questions about the material, please give me a call.

Skip

Skip Copp

Duke Power – Nuclear Regulatory Commission Meeting

Topical Report DPC-NE-1005P

Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX

Rockville, MD

May 23, 2002

Agenda
Duke – NRC Meeting
Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX
Thursday, May 23, 2002

Topic	Discussion Lead
Introductions	
Mixed Oxide (MOX) Fuel Project Status and Plans	Duke (Nesbit)
Duke Nuclear Analysis Methodologies	Duke (Nesbit)
CMS – Core Management System (Analytical Models)	Studsvik (Smith)
Lunch	All
Qualification of Nuclear Analysis Methodologies	Duke (Eller)
Power Reactor Benchmark Analyses	Duke (Eller)
Fuel Pin Power Distribution Benchmark Analyses	Duke (Naugle)
Statistically Combined Power Distribution Uncertainty Factors	Duke (Eller)
Dynamic Rod Worth Measurement	Duke (Thomas)
NRC Questions	NRC
Adjourn	

1

Mixed Oxide (MOX) Fuel Project Status and Plans

Meeting with Nuclear Regulatory Commission

- DPC-NE-1005P Review -

S. P. Nesbit
Duke Power
May 23, 2002

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Plutonium Disposition Program

- **Goal: To dispose of surplus weapons plutonium**
  - **January 2000 Department of Energy (DOE) Record of Decision**
  - **September 2000 U.S.-Russian Federation Plutonium Disposition Agreement**
- **Initial Approaches**
  - **Fabrication into mixed oxide (MOX) fuel and use in existing light water reactors**
  - **Immobilization in vitrified high-level radioactive waste**

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MOX Fuel Project

- MOX Fuel Fabrication
  - 12/00: DCS submitted Environmental Report to NRC
  - 2/01: DCS submitted Construction Authorization Request to NRC
- MOX Fuel Qualification
- MOX Fuel Irradiation
- Fresh MOX Fuel Transportation and Packaging
- Project Management

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Plutonium Disposition Program Changes

- Recently announced changes from the Department of Energy
  - Termination of immobilization portion of program
  - Design changes to the MOX Fuel Fabrication Facility to accommodate a wider variation of feed material
  - One year delay in the provision of batch quantities of MOX fuel from the MOX Fuel Fabrication Facility

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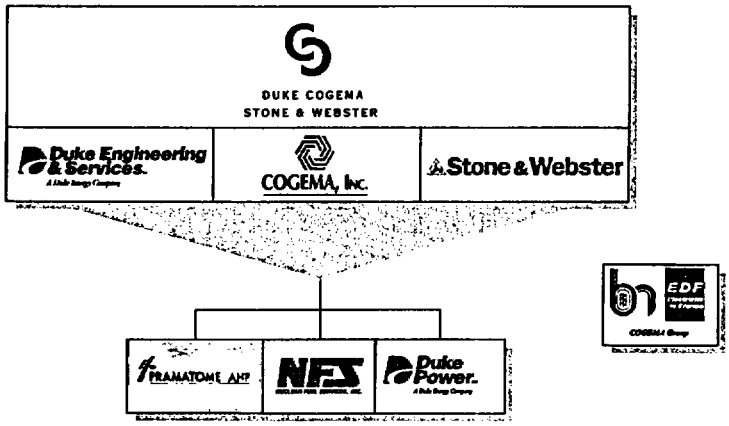
Duke Cogema Stone & Webster (DCS) Team



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MOX Fuel Qualification and Irradiation

- Maximize use of European experience base
  - Research programs
  - Established manufacturing process
  - Reactor irradiation experience
- Proven fuel assembly design
- Confirmatory lead assembly program
- NRC reactor operating license amendments in accordance with 10 CFR 50.90

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MOX Fuel-Related Submittals

- July 2000: DCS Fuel Qualification Plan provided to NRC for information
- August 2000: Framatome COPERNIC Topical Report (MOX applications)
- April 2001: DCS MOX Fuel Qualification Plan revised and provided to NRC for information
- August 2001: Duke Power Nuclear Analysis Topical Report (MOX and LEU applications)

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MOX Fuel-Related Submittals (cont.)

- September 2001: Duke Power Thermal-Hydraulic Statistical Core Design Topical Report, Appendix E (advanced Mk-BW fuel assembly design, to be used for MOX fuel)
- April 2002: Framatome Advanced Mark-BW Fuel Assembly Design Topical Report
- April 2002: Framatome MOX Fuel Design Topical Report

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MOX Fuel Lead Assembly Program

- Original approach - fabricate two MOX fuel lead assemblies at Los Alamos National Laboratory (LANL) and begin use in McGuire Nuclear Station in fall 2003
- LANL fabrication activities terminated May 2000
- Alternatives under consideration
  - Fabrication at existing European MOX fuel fabrication facilities
    - Start irradiation ~2004
  - Fabrication at Savannah River MOX Fuel Fabrication Facility, when constructed and licensed
    - Start irradiation ~2008

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MOX Fuel Qualification and Irradiation Plans

- 2002?: Submit MOX Fuel Lead Assembly License Amendment Request (Duke Power)
- 2003?: Submit Updated Fuel Qualification Plan (DCS)
- 2003: Submit MOX Fuel Safety Analysis Topical Report (Duke Power)

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MOX Fuel Qualification and Irradiation Plans (cont.)

- December 2003: Submit License Amendment Requests for Batch Utilization of MOX Fuel at McGuire and Catawba (Duke Power)
- 2004: Submit MOX Fuel LOCA Topical Report (Framatome)
- 2004?: Begin MOX fuel lead assembly irradiation

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2

Duke Nuclear Analysis Methodologies

Meeting with Nuclear Regulatory Commission

- DPC-NE-1005P Review -

S. P. Nesbit
Duke Power
May 23, 2002

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Duke Power Fuel Management

- Purchasing uranium, conversion, enrichment, and fabrication
- Core design and analysis
- Fuel mechanical design and analysis
- Fuel thermal-hydraulic analysis
- Safety analysis
- Criticality analysis
- Spent fuel management

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Duke Power Reload Analyses

- Full-scope reload analyses (except loss of coolant accident analyses) for the seven Oconee, McGuire, and Catawba units
- 1982: Startup of first Oconee core with Duke loading pattern and safety analysis
- 1991: Startup of first McGuire/Catawba core with Duke loading pattern and safety analysis

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Selected Duke Topical Reports

- 1981: NFS-1001 approved
  - Oconee
  - Steady-state nuclear analyses
  - EPRI-CELL, PDQ07, and EPRI-NODE
- 1985: DPC-NF-2010 approved
  - McGuire/Catawba
  - Steady-state nuclear analyses
  - EPRI-CELL, CASMO-2, PDQ07, and EPRI-NODE

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Selected Duke Topical Reports (cont.)

- 1992: DPC-NE-1004 approved
  - Oconee, McGuire, and Catawba
  - Steady-state nuclear analyses
  - CASMO-3 and SIMULATE-3P
- 2000: DPC-NE-2012 approved
  - McGuire/Catawba
  - Dynamic Rod Worth Measurement applications
  - CASMO-3, SIMULATE-3P, S3K

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Impetus for DPC-NE-1005P

- Implementation of CASMO4 lattice code
  - Improved methodology
  - General benefits for all fuel types
  - Consistency with Oconee (topical report submittal planned for 2002)
  - Methods transition planned for late 2002
    - Analyses supporting 2004 reloads at McGuire and Catawba
- Demonstration of MOX fuel analysis capability
  - Lead assembly cores (2004?)
  - Batch cores (2008?)

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DPC-NE-1005P - Overall Approach

- Compare CASMO4 and SIMULATE-3 MOX calculations to applicable plant and experimental data
  - Power reactor benchmarks
  - Critical experiment benchmarks
- Quantify uncertainty factors for MOX and LEU fuel applications at McGuire and Catawba
- Same fundamental approach as used in previously approved Duke nuclear analysis topical reports

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Overview of Presentations

- Analytical Models (Topical Report Section 2)
- Nuclear Analysis Methodology Qualification
- Power Reactor Benchmark Analyses (TR Section 3)
- Fuel Pin Power Distribution Benchmark Analyses (TR Section 4)
- Statistically Combined Power Distribution Uncertainty Factors (TR Section 5)
- Dynamic Rod Worth Measurement (TR Section 6)

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3

CMS - Core Management System
Studsvik Scandpower, Inc.

Kord S. Smith
Vice-President of Technical Development
Studsvik Scandpower, Inc.
504 Shoup Ave., Suite 201
Idaho Falls, ID 83402
(208) 522-1060
kord@west.soa.com

Studsvik Scandpower Organization

Products: Computer codes for nuclear power plant in-core fuel management.
Staff: More than 40 nuclear engineers, with offices worldwide.

- Studsvik Scandpower AB, Sweden
- Studsvik Scandpower, USA
- Studsvik Scandpower, Norway
- Studsvik Scandpower, Germany
- Studsvik Scandpower, Switzerland
- Studsvik Japan, Japan

Studsvik Scandpower CMS

CASMO-4
Lattice Physics Code

CMS-LINK
Linking Code

XIMAGE
Fuel management

SIMULATE-3
Steady-State Analysis

GARDEL
On-line Monitoring

SIMULATE-3R
Real Time Simulator

SIMULATE-3K
Transient Analysis

CMS Has Wide Range of Applications

- 11 Countries
- 55 Companies
  - Nuclear Utilities
  - Nuclear Fuel Vendors
  - Regulatory Agencies
  - National Laboratories
  - Universities
- Applied to more than 70 BWRs and 90 PWRs
- Applied to more than 2000 BWR and PWR Cycles

CASMO-4 Lattice Physics Code

- Purpose:
  - Analyze the detailed behavior of a fuel bundle over its lifetime
  - Treat fuel, burnable absorbers, control rods, instruments, water gaps
  - Provide bundle data for downstream core analysis codes
- Neutronic Data Library:
  - Basic data library is NJOY-generated (70 groups from 0 -10 MeV) using mostly ENDF/B-IV, with some JEF-1, and JEF-2.1 data.
  - Numerous temperatures and background cross sections used to treat resonance self-shielding (Doppler broadening)
  - Contains more than 100 materials commonly used in LWRs

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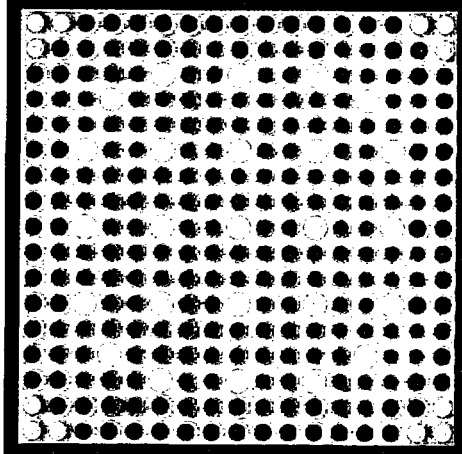
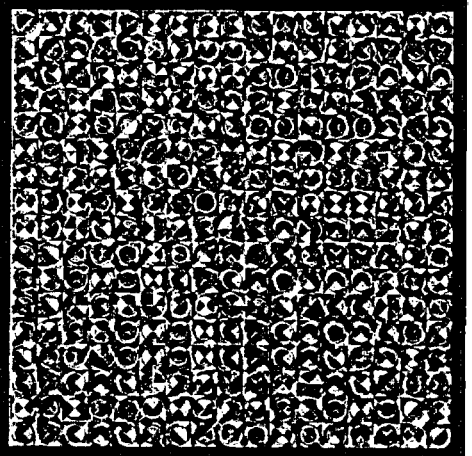
2-D Heterogeneous Transport Model



Physical Geometry



"Flat Source" Representation

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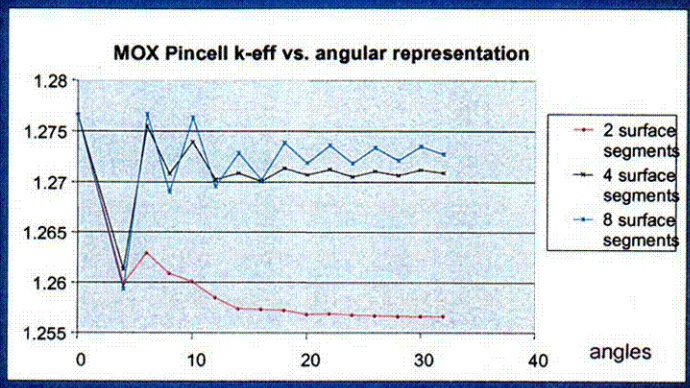
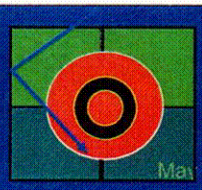
CASMO-4 Overcomes Limitations Inherent In Current-Coupling-Collision-Probability Methods





Surface segment 1

Surface segment 2

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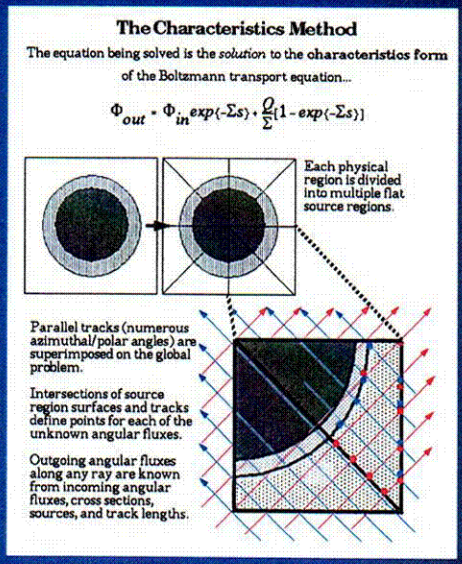
Method of Characteristics (MOC)

- Much more angular detail:
  - Azimuthal: 32-128 angles
  - Polar: 2-5 angles
  - Ray spacing: 0.1 cm



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C-01

CASMO-4 Benchmarking of k-eff vs. Criticals

	L-library (ENDF/B-IV)	
B&W	(LEU PWR, cold)	1.00050
KRITZ-3	(LEU PWR, cold and hot)	0.99699
KRITZ-4	(LEU BWR, cold and hot)	0.99900
KRITZ-3	(MOX, cold and hot)	0.99912
VIP	(MOX, cold)	0.99973
	ENDF/B-VI library	
B&W	(LEU PWR, cold)	1.00301
KRITZ-3	(LEU PWR, cold and hot)	0.99701
KRITZ-4	(LEU BWR, cold and hot)	0.99990
KRITZ-3	(MOX, cold and hot)	0.99803
VIP	(MOX, cold)	0.99530
	JEF-2.2 Library	
B&W	(LEU PWR, cold)	1.00227
KRITZ-3	(LEU PWR, cold and hot)	0.99779
KRITZ-4	(LEU BWR, cold and hot)	0.99945
KRITZ-3	(MOX, cold and hot)	0.99785
VIP	(MOX, cold)	0.99863

CASMO-4 Library Status

- L-library is currently used by all customers for production analysis
- L-library has little bias in reactivity between LEU and MOX
- ENDF/B-VI Library has -400 pcm bias between LEU and MOX
- JEF-2.2 library has about -150 pcm bias between LEU and MOX
- Pin power distribution accuracy is insensitive to library
- At present, Studsvik Scandpower recommends customers to continue to use the L-library
- When new libraries are fully tested and ready, JEF-2.2 will probably move into production sooner than ENDF/B-VI (at Studsvik Scandpower)

CASMO-4 Case Matrix For Each Unique Lattice

- Depletion Cases (History Cases):
  - Several coolant temperatures and/or void
  - Several boron concentrations
  - Several fuel temperatures
- Branch Cases:
  - Many coolant temperatures
  - Many boron concentrations
  - Many fuel temperatures
  - All control rod types
- Reflector Cases:
  - Radial baffle/reflector
  - Top and bottom axial reflectors

CASMO-4 Data Produced

- Macroscopic cross sections (two groups)
- Microscopic cross sections and yields for fission products (Xe, Sm)
- Discontinuity factors (treats bundle heterogeneities)
- In-core detector constants
- Pin-by-pin power distributions (two groups)
- Bundle-averaged isotopics vs. depletion
- Kinetics data:
  - delayed neutron yields and decay constants in 6 groups
  - Neutron velocities

CMS-LINK Linking Code

- Reads all relevant CASMO-4 output data for each fuel type
- Collects fuel descriptors and geometrical data
- Computes 1-D, 2-D, and 3-D tabular data tables for each variable
  - Cross sections
  - Detector data
  - Pin-by-pin data
  - Etc.
- Creates separate tables for history and instantaneous parameters
  - Coolant density or void
  - Fuel temperature
  - Control rod
  - Etc.
- Spline fits data to guarantee accuracy of downstream linear interpolation
- Creates binary data library for SIMULATE-3 and SIMULATE-3K

SIMULATE-3 MOX

- SIMULATE-3 core simulator first introduced in 1985
- Used for steady-state core analysis: reload core design, safety parameter generation, RPS limit generation, and operational plant support
- Full two-group advanced nodal code:
  - 1 or 4 nodes per assembly
  - Explicit reflectors (no albedos)
  - Explicit tracking of I, Xe, Pm, Sm
  - Discontinuity factors to treat bundle heterogeneities
  - Quartic polynomial spatial representation of intra-nodal flux distributions
  - Quadratic transverse leakage treatment
  - Quadratic intra-nodal burnup gradient modeling
  - Spectral history treatment of bundle interface spectrum interactions
  - Pin power reconstruction

SIMULATE-3 MOX Enhancements

- First MOX analysis applications in 1989
- Goal: achieve MOX/LEU mixed-core accuracy comparable to that of LEU cores
- Continuous development throughout the 1990's
- Currently used in Germany, Switzerland, UK, Japan, and the U.S.

- Enhancements relative to original SIMULATE-3:
  - Analytic (sinh, cosh) intra-nodal thermal fluxes replaced quartic polynomials
  - Corner point flux interpolation model improved
  - Spatial re-homogenization of cross sections to treat global flux gradients
  - Two-group pin power form functions replaced total power form functions
  - Instantaneous spectral effects on 2-group cross sections modeled at interfaces
  - P-3 transport effects modeled at bundle interfaces

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Verification/Validation of SIMULATE-3 MOX vs. CASMO-4

- **Direct ¼-core calculations with CASMO-4**
- **Same detail as lattice physics computation**
- **Explicit isotopic depletion for all nuclides**
- **Reference cases for SIMULATE-3 MOX - permit testing of all SIMULATE-3 MOX modeling approximations:**
  - **Verify/Improve Nodal Approximations**
  - **Investigate errors in Pin Power Predictions**
  - **Investigate Complicated Depletion Effects**
  - **Investigate Detector Modeling Approximations**
  - **Study 2-D Baffle/Reflector Effects**
  - **Perform Moveable/Detector Analysis**

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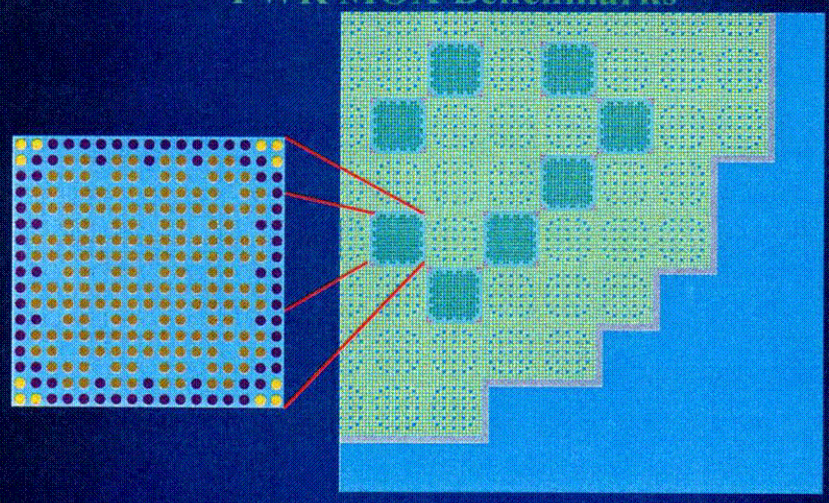
¼ Core CASMO-4
PWR MOX Benchmarks



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Assembly Power Comparisons

1.336 1.321 -.015	1.070 1.064 -.006	0.866 0.876 +.010	1.070 1.070 +.000	0.921 0.934 +.013	1.125 1.121 -.004	1.191 1.175 -.016	0.423 0.421 -.002
	1.234 1.233 -.001	1.191 1.195 +.004	1.294 1.298 +.004	1.305 1.314 +.009	1.348 1.350 +.002	1.171 1.155 -.016	0.396 0.394 -.002
		0.908 0.920 +.012	1.169 1.171 +.002	1.384 1.399 +.015	1.224 1.234 +.010	0.938 0.929 -.009	0.279 0.274 -.005
			1.423 1.432 +.009	1.345 1.357 +.012	1.163 1.166 +.003	0.494 0.489 -.005	
				1.304 1.305 +.001	0.999 0.982 -.017	0.299 0.291 -.008	
					0.381 0.372 -.009		RMS diff .009

20 b.p. | MOX

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C-02

SIMULATE-3K MOX Features

- Transient version of SIMULATE-3 MOX
- Permits time-dependent boundary conditions for:
  - Boron concentration
  - Core inlet coolant temperature/flow
  - Control rods positions
  - System pressure
- Features:
  - Spatial neutronics model is identical to steady-state SIMULATE-3 MOX
  - All neutronic data taken from standard CMS-LINK library
  - Fully-implicit temporal differencing of frequency-transformed diffusion equation
  - Analytic solution of delayed neutron precursor equations (6 groups)
  - Spontaneous fission/alpha-n neutron sources modeled
  - User-specified or automatic time-step selection

SIMULATE-3K/SIMULATE-3 Differences

- Fuel temperatures are computed using an explicit fuel pin conduction model
- Coolant densities are computed using an explicit channel hydraulic model
- At HZP, pin conduction and channel hydraulic differences have zero effect on computations
- All MOX enhancements are identical in S3 and S3K

SIMULATE-3K Applications

- PWRs:
  - Reactivity excursions (ejected bank/rod)
  - Dynamic rod worth measurements (DRWM)
  - Dropped rod transients
  - Steam line break analysis
  - Boron dilution accidents
- BWRs:
  - Stability analysis
  - SCRAM reactivity curve generation
  - Reactivity excursions (dropped/stuck rod)
  - Operational transients (e.g., pump trips, turbine trips)

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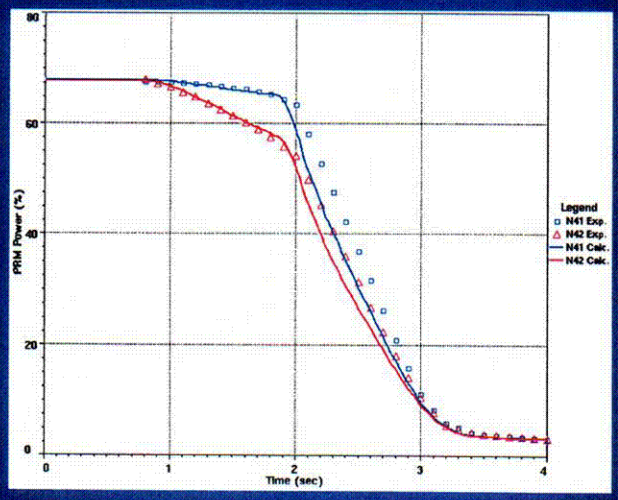
Ringhals PWR Dropped Rod Analysis



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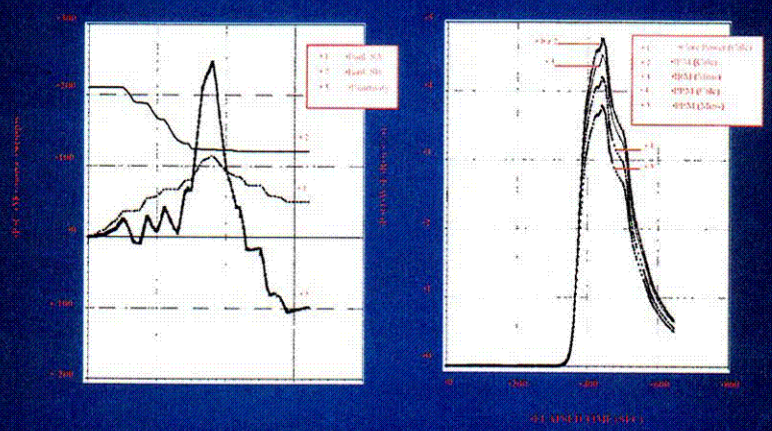
C-03

PWR HZP Inadvertent Bank Withdrawal



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CMS MOX Usage

- BNFL:
  - Reload core design for MOX fuel sales support
  - Analysis of MOX loaded cores for fuel contracts
  - Analysis of spent/recycled fuel for reprocessing facility support
- Japan
  - Analysis of European MOX-loaded cores for licensing activities
  - Core design/support for MOX fuel introduction at TEPCO (BWR)
  - Core design/support for MOX fuel introduction at Kansai (PWR)
- Germany:
  - Core design and support for MOX-loaded Gundremingen (BWR)
  - Licensing authority (TUV) verification analysis
- Switzerland:
  - Core design/support for Beznau Units I and II (PWR)
  - On-line core monitoring (GARDEL) for Beznau Unit I
- U.S.:
  - Duke Power

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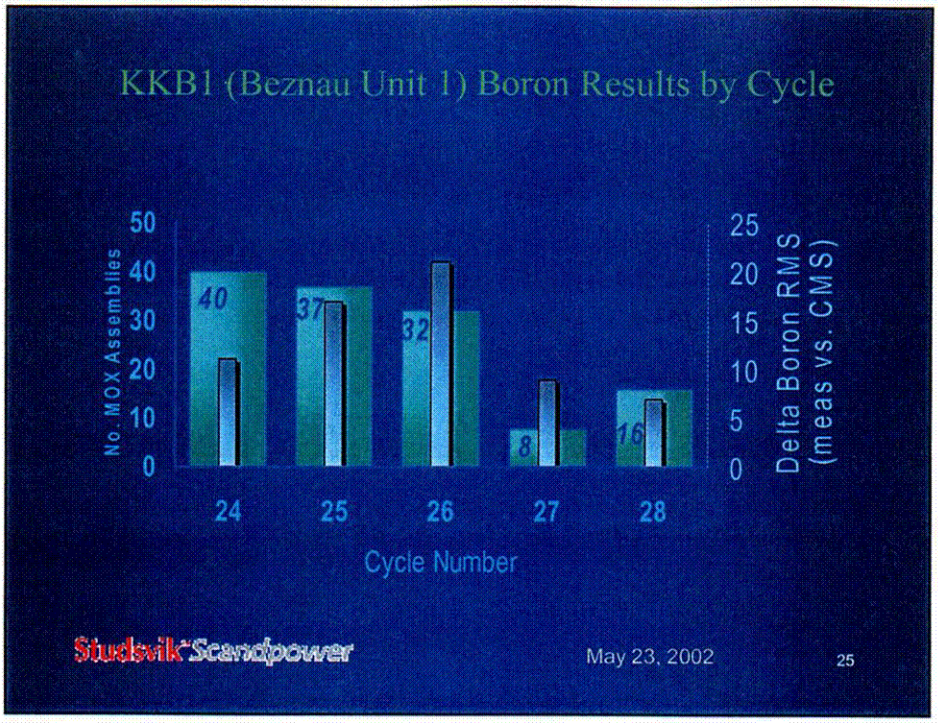
KKB1 (Beznau Unit 1) Boron Results by Cycle
No. MOX Assemblies
50
40
30
20
10
0
40
37
32
8
16
24
25
26
27
28
Cycle Number
25
20
15
10
5
0
Delta Boron RMS
(meas vs. CMS)
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May 23, 2002
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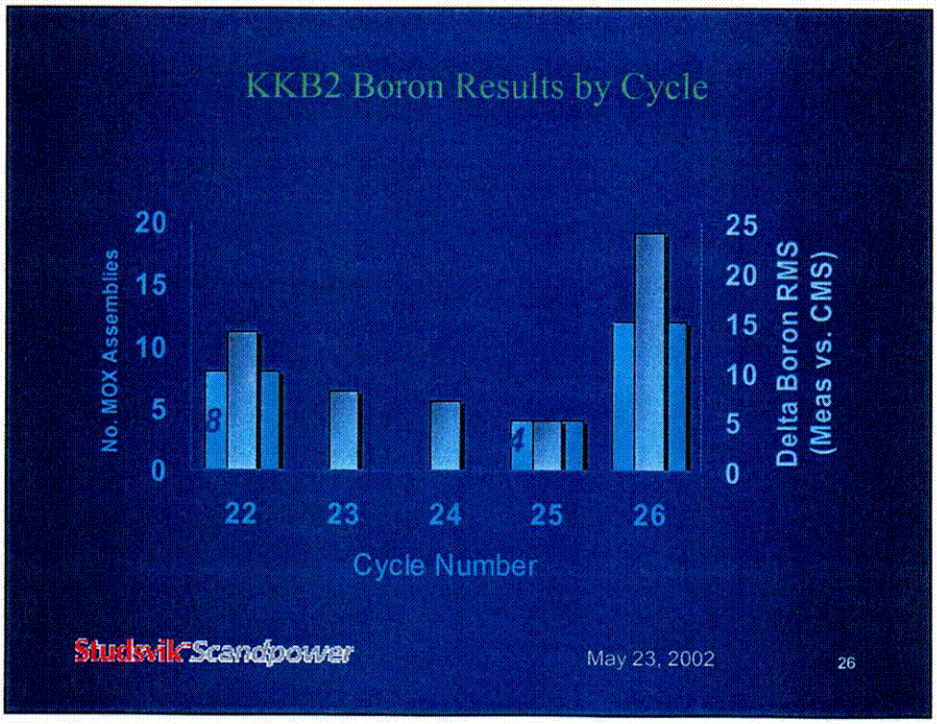
KKB2 Boron Results by Cycle
No. MOX Assemblies
20
15
10
5
0
8
4
22
23
24
25
26
Cycle Number
25
20
15
10
5
0
Delta Boron RMS
(Meas vs. CMS)
Studsvik Scandpower
May 23, 2002
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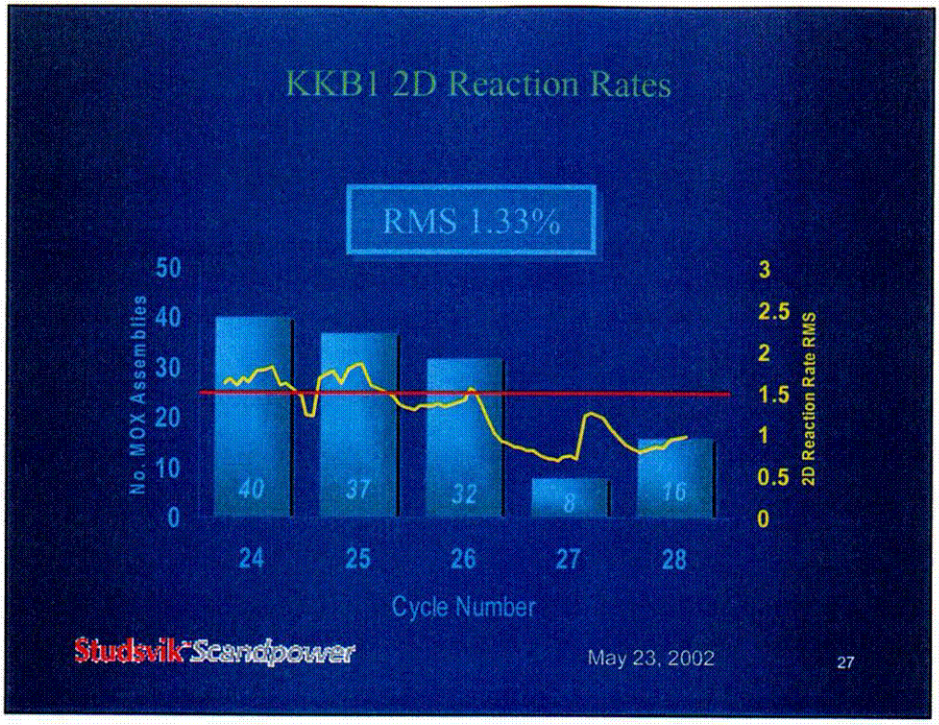
C-04


KKB1 2D Reaction Rates
RMS 1.33%
No. MOX Assemblies
50
40
30
20
10
0
40
37
32
8
16
24
25
26
27
28
Cycle Number
3
2.5
2
1.5
1
0.5
0
2D Reaction Rate RMS
Studsvik Scandpower
May 23, 2002
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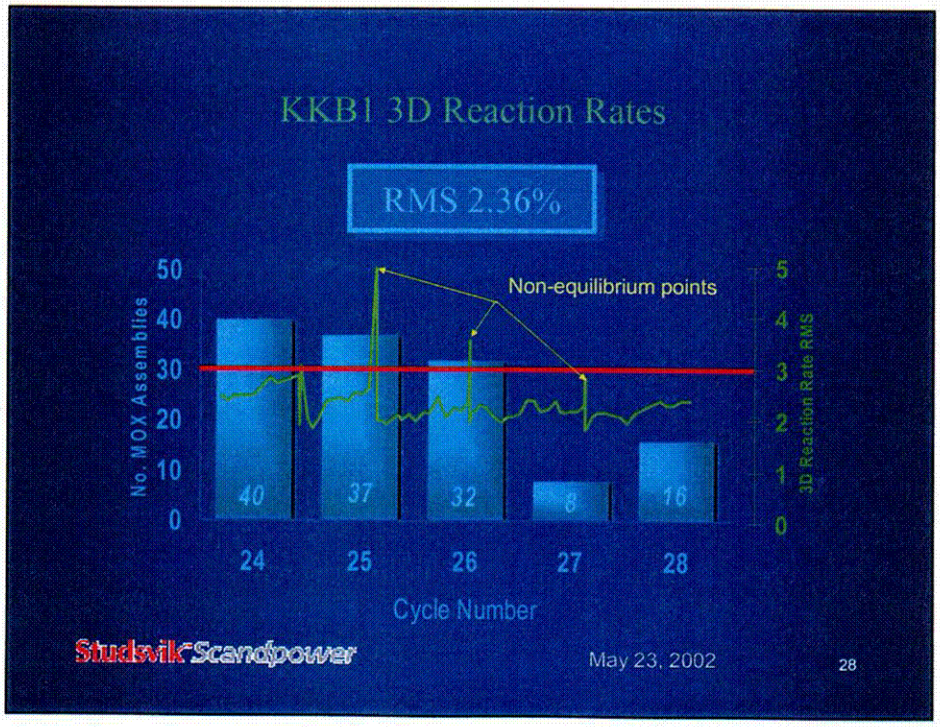
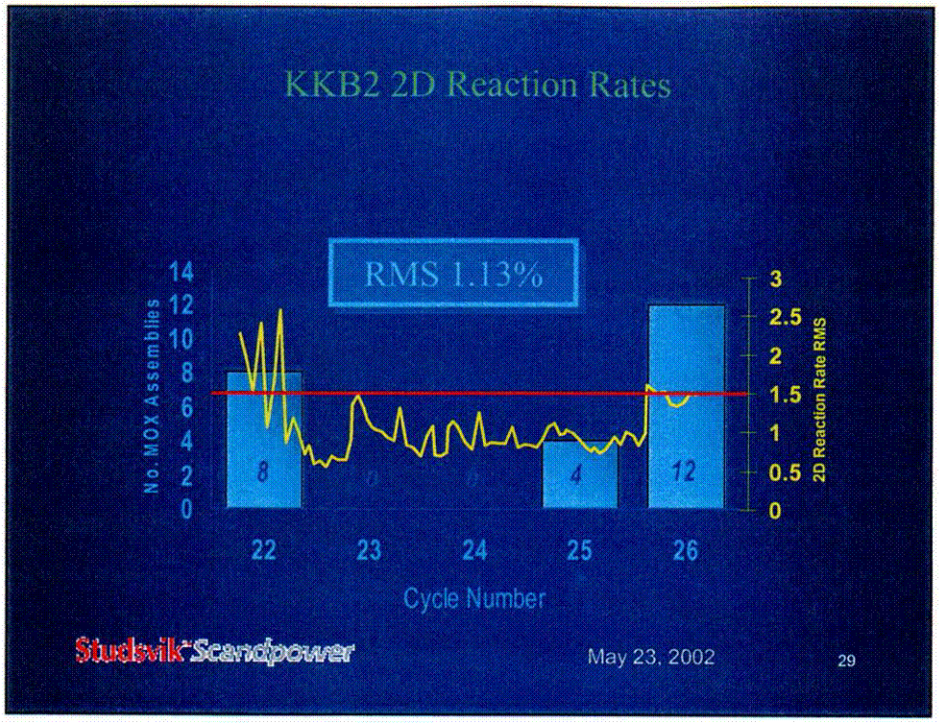

KKB1 3D Reaction Rates
RMS 2.36%
Non-equilibrium points
No. MOX Assemblies
50
40
30
20
10
0
40
37
32
8
16
24
25
26
27
28
Cycle Number
5
4
3
2
1
0
3D Reaction Rate RMS
Studsvik Scandpower
May 23, 2002
28

C-05


KKB2 2D Reaction Rates
RMS 1.13%
No. MOX Assemblies
14
12
10
8
6
4
2
0
8
4
12
22
23
24
25
26
Cycle Number
3
2.5
2
1.5
1
0.5
0
2D Reaction Rate RMS
Studsvik Scandpower
May 23, 2002
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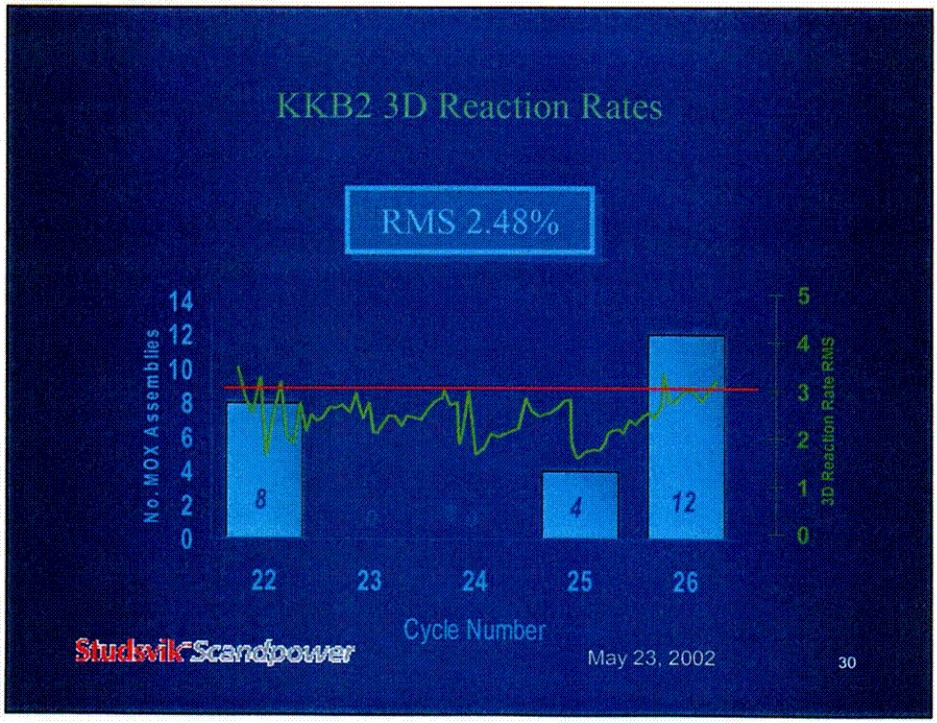
KKB2 3D Reaction Rates
RMS 2.48%
No. MOX Assemblies
14
12
10
8
6
4
2
0
8
4
12
22
23
24
25
26
Cycle Number
5
4
3
2
1
0
3D Reaction Rate RMS
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May 23, 2002
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C-06

Summary

- CASMO-4 and SIMULATE-3 MOX have already been used for core design and analysis in MOX-fueled PWRs and BWRs.
- Accuracy in MOX-fueled cores is comparable to that obtained in LEU-fueled cores.
- CASMO-4/SIMULATE-3 MOX can be applied with confidence for Duke Power's upcoming MOX applications.

Studsvik Scandpower remains committed to continued development of models and codes for applications in LEU- and MOX-fueled LWRs.

4

Qualification of Nuclear Analysis Methodologies

Meeting with Nuclear Regulatory Commission

DPC-NE-1005P Review

Jim Eller

Duke Power

May 23, 2002

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Goal

Define a modeling technique which has :

- **Acceptable accuracy**
- **Reliable performance**
- **Direct and understandable approach**
- **Builds on existing experience base**
- **Effective use of human and computer resources**

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Benchmarking Approach

Dictated by the type of measured data that is available

or

Dictated by the type of measured data that is NOT available

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PWR Measurements

- BOC startup tests at HZP
  - Critical soluble boron concentration
  - Control rod bank worth
  - Isothermal temperature coefficient
- At power critical soluble boron letdown
- At power core power distribution measurement

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Measurement of Core Power Distribution

- Moveable incore fission chamber
- Travels up central instrument tube of fuel assembly
- Approximately $^1/_3$ of all fuel assemblies instrumented
- Measured electrical signal is proportional to flux level in center of fuel assembly
- Flux level measured in radial center of fuel assembly is related to average assembly power

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Core Design Methodology

- Requires a conservative verification of multiple fuel pin performance criteria
- Precision of core model pin by pin power distribution prediction must be known
- Measured pin by pin power distribution data is not available from power reactor operation

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Laboratory Experiments

- Some experiments measure power distribution in critical arrays of fuel pins
- Useful experiments utilize materials and lattice arrangements that are similar to PWR fuel
- Analytic models of experimental geometries allow comparison of predicted and measured pin power distributions

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Summary

- DPC-NE-1005 seeks to extend and improve currently licensed reload core design methodology
- Goal is to define a core modeling technique that is accurate, consistent, and efficient
- Benchmark approach is dictated by available measurements

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Power Reactor Benchmark Analyses

Meeting with Nuclear Regulatory Commission

DPC-NE-1005P Review

Jim Eller
Duke Power
May 23, 2002

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Reactors Modeled

- **McGuire Nuclear Station Units 1 and 2**
  - **three fuel cycles for each unit**
- **Catawba Nuclear Station Units 1 and 2**
  - **three fuel cycles for each unit**
- **Saint Laurent B1**
  - **six fuel cycles**

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McGuire & Catawba

- 3411 MW thermal power level
- 193 fuel assemblies in core
- 17x17 fuel assembly lattice
- Base loaded 18 month fuel cycles

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Saint Laurent B1

- 2775 MW thermal power level
- 157 assemblies in the core
- 17x17 fuel assembly lattice
- Cycles 5 – 10 are base loaded 12 month fuel cycles

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PWR Measurements

- BOC startup tests at HZP
  - Critical soluble boron concentration
  - Control rod bank worth
  - Isothermal temperature coefficient
- At power critical soluble boron letdown
- At power core power distribution measurement

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McGuire
Deviation in Critical Boron Concentration

[ ]

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Catawba
Deviation in Critical Boron Concentration

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Saint Laurent B1
Deviation in Critical Boron Concentration

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McGuire, Catawba, Saint Laurent B1 Deviation in Control Rod Bank Worth

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McGuire, Catawba, Saint Laurent B1 Deviation in Control Rod Bank Worth

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McGuire, Catawba, Saint Laurent B1
Deviation in Isothermal Temperature Coefficient

[ ]

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Comparison of Power Reactor Benchmark Results

	Average Deviation	Standard Deviation
McGuire / Catawba		
BOC HZP Soluble Boron (PPMB)	[	]
HFP Soluble Boron (PPMB)		
BOC HZP Control Rod Worth (%)		
BOC HZP ITC (pcm / F)		
Saint Laurent B1		
BOC HZP Soluble Boron (PPMB)		
HFP Soluble Boron (PPMB)		
BOC HZP Control Rod Worth (%)		
BOC HZP ITC (pcm / F)		]

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Statistical Methodology

- One sided upper tolerance limit uncertainties
- Total uncertainty factor = 1 – bias + Kσ
- K factor ensures with a 95% confidence level that 95% of local power predictions are equal to or larger than the measured value

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Comparison of Assembly Power Distribution Uncertainties

		Bias Calc - Meas	Statistical Deviation Kσ	Uncertainty Factor 1 - bias + Kσ
McGuire / Catawba LEU Fuel				
	FΔh			
	Fq			
	Fz			
Saint Laurent B1 LEU Fuel				
	FΔh			
	Fq			
	Fz			
Saint Laurent B1 MOX Fuel				
	FΔh			
	Fq			
	Fz			

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Summary

- McGuire, Catawba, and Saint Laurent B1 reactor cores are physically and neutronically similar
- A consistently applied core modeling technique demonstrates similar performance for all 3 reactors
- Models of mixed cores of LEU and MOX fuel demonstrate essentially the same fidelity as do the models of all LEU fuel cycles

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Fuel Pin Power Distribution Benchmark Analyses

Meeting with Nuclear Regulatory Commission

- DPC-NE-1005P Review -

Kenneth Naugle
Duke Power
May 23, 2002

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Overview

- Overall Approach
- Typical Critical Experiments
- Low enriched uranium fuel (B&W)
- MOX fuel (Saxton, EPICURE, and ERASME)
- Theoretical benchmarks
- Overall pin uncertainty

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Overall Approach

- Goal: Develop pin power distribution uncertainty factors
- Method: Compare computer code predictions to measured critical experiment power distributions
  - Power reactors provide relative assembly powers only
  - Critical experiments provide detailed pin-by-pin power distributions

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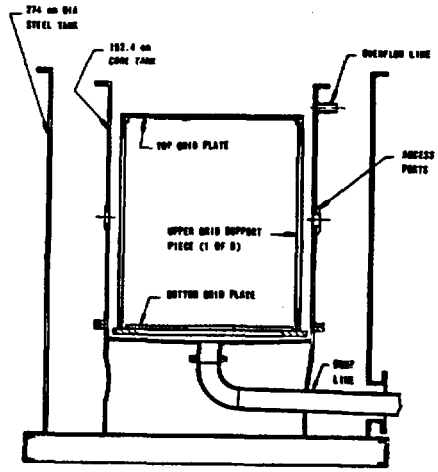
Typical Critical Experiment

- Pins are supported in their positions by the grid plates
- Criticality is achieved by control of water level &/or boron concentration
- Pin powers are inferred by gamma scans of the fuel pins after conclusion of the experiment



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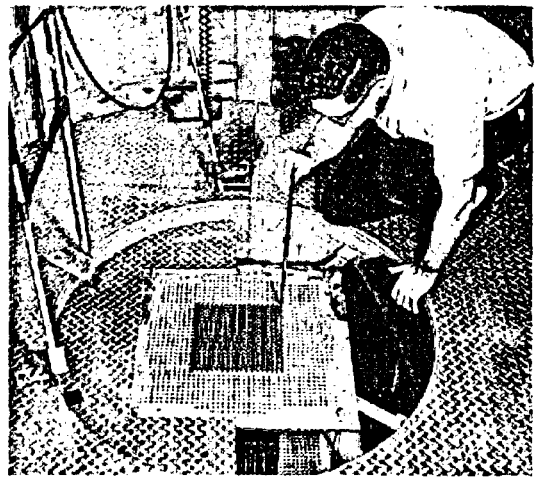
Typical Critical Experiment

- Pin configuration is established dry, then tank is filled with water
- Typical active fuel region height is ~80-100 cm



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Methodology

Two methods were used to derive the pin power uncertainty from critical experiment data:

- Method 1: Model experiment using CASMO-4
- Method 2: Create cross-section information using CASMO-4 and model experiment with SIMULATE-3 MOX

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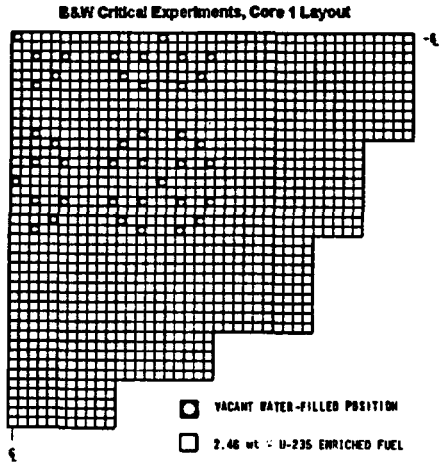
B&W Critical Experiments Layout

- Used previously by Duke Power
- Low enriched uranium fuel (2.5% and 4.0% U-235)
- Size ≅ 81 pin pitches
- Measured powers are in central 15x15 or 16x16 region



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LEU Pin Power Uncertainty

- Based on three B&W experiments
  - Same as previously-approved Duke Power topical report (DPC-NE-1004)
  - SIMULATE-3 MOX model using CASMO-4 based cross-section inputs
  - [ ] measured pin powers
- 95/95 uncertainty: [ ] (Method 2)
  - Comparison model with CASMO-4 (Method 1) produced a 95/95 uncertainty of [ ]

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Desirable Characteristics of MOX Critical Experiments

- Comparable plutonium isotopics (~93% Pu-239)
- Comparable fissile content (up to ~5%)
- Comparable 17x17 fuel pin lattice
- Lattice includes LEU pin regions
- Radially-zoned MOX concentrations in lattice
- Poison materials: silver-indium-cadmium [AIC] and boron carbide [B_4C]

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Saxton Critical Experiments

- Performed at Westinghouse Reactor Evaluation Center in 1965
- 19x19 single region MOX to 27x27 two region MOX/LEU (i.e. ≤27 pin pitches)
- Includes AIC pins, aluminum slab, and water gaps
- MOX fuel pins: 6.6% plutonium (90% Pu-239)
- LEU fuel pins: 5.7% U-235

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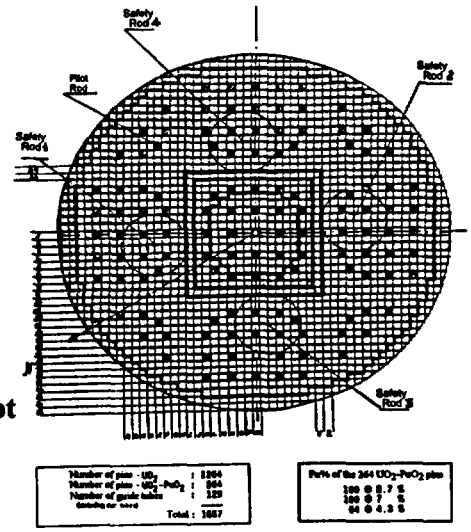
EPICURE Experiments - UMZONE

- Reactor grade MOX fuel (~70% Pu-239) in central 17x17 region surrounded by 3.7% LEU fuel
- Zoned MOX concentrations
- Configurations include some with AIC & B_4C control rods in 24 central guide tubes
- Size ≅ 45 pin pitches
- Pin OD and spacing is identical to Duke Power fuel
- Aluminum overclad to simulate hot condition fuel/moderator ratio



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EPICURE Experiments – UM & MH

- UM experiments consist of a 17x17 uniform concentration MOX region surrounded by 3.7% LEU fuel
  - 7% MOX region
  - 11% MOX region
  - Size ≅ 45 pin pitches
- MH-1.2/93 experiment has a cylindrical 7% MOX region surrounded by 3.7% LEU fuel
  - Size = 55 pin pitches

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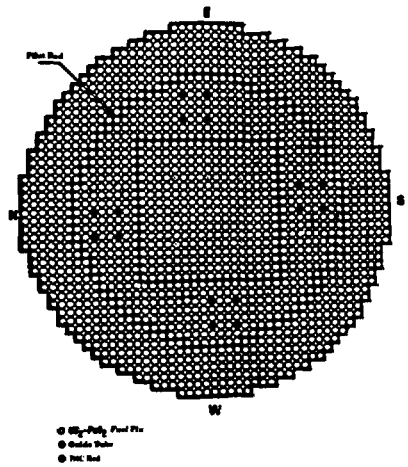
ERASME Experiments

- Pin spacing (1.19 cm) is slightly smaller than EPICURE (1.26 cm)
- Size ≅ 45 pin pitches
- All MOX pins (11% reactor grade Pu)
- B_4C rods in lattice



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MOX Critical Experiment Results

- CASMO-4 models (Method 1)
- 95/95 uncertainty
  - Calculated from the combined data set of Saxton, EPICURE, and ERASME experiments ([ ] pin powers)
  - Uncertainty = [ ]

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Theoretical Benchmarks

- The MOX fuel critical experiments are too small to model with SIMULATE-3 MOX using CASMO-4 based cross-section inputs (Method 2)
- Pin power uncertainty consists of:
  - CASMO-4 to MOX fuel critical experiments
  - SIMULATE-3 MOX to CASMO-4 predictions
- Theoretical Model Description
  - Infinite lattices with 2x2 MOX & LEU assemblies (colorsets)
  - Same problem modeled with CASMO-4 and SIMULATE-3 MOX
  - Provides for quantification of SIMULATE-3 MOX to CASMO-4 uncertainty

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Theoretical Model Configurations

Case 1: Checkerboard MOX/LEU feed with two MOX reinserts

4.37% MOX 24 BP rods pulled 15 GW/d/Mtm	4.37% MOX 24 BP rods FEED
4.0% LEU No BP rods FEED	4.37% MOX No BP rods 25 GW/d/Mtm

Case 2: Face adjacent MOX feed with two LEU reinserts

4.0% LEU No BP rods 20 GW/d/Mtm	4.37% MOX No BP rods FEED
4.0% LEU No BP rods 20 GW/d/Mtm	4.37% MOX 24 BP rods FEED

Case 3: One MOX feed with one MOX & two LEU reinserts

4.0% LEU No BP rods 20 GW/d/Mtm	4.37% MOX No BP rods FEED
4.0% LEU No BP 25 GW/d/Mtm	4.37% MOX No BP rods 20 GW/d/Mtm

Case 4: Face adjacent MOX/LEU feed, with two MOX reinserts

4.37% MOX No BP rods 15 GW/d/Mtm	4.37% MOX 24 BP rods FEED
4.37% MOX 24 BP rods pulled 25 GW/d/Mtm	4.0% LEU No BP rods FEED

Case 5: 2 MOX feed & 1 LEU feed (face adjacent) and 1 LEU reinsert

4.0% LEU No BP rods FEED	4.37% MOX 24 BP's rods FEED
4.37% MOX No BP rods FEED	4.0% LEU No BP rods 20 GWD/Mtm

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Theoretical Model Results

- Comparisons of CASMO-4 and SIMULATE-3 MOX calculations were performed at lattice burnups of 0, 10, & 20 GWd/MThm
- [ ] data points
- 95/95 uncertainty: [ ]

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MOX Pin Power Uncertainty

- CASMO-4 uncertainty from combined Saxton, EPICURE, & ERASME experiments: [ ]
- SIMULATE-3 MOX to CASMO-4 uncertainty from theoretical models: [ ]
- Overall MOX pin uncertainty

  Uncertainty = [ ] = [ ]

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Conclusions

- SIMULATE-3 MOX pin power uncertainty for LEU fuel is nearly identical to previous Duke results
- The pin power uncertainty for MOX fuel pins is slightly higher
  - Smaller, more challenging critical experiments
  - Conservative characterization of SIMULATE-3 MOX to CASMO-4 uncertainty

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Statistically Combined Power Distribution Uncertainty Factors

Meeting with Nuclear Regulatory Commission

DPC-NE-1005P Review

Jim Eller

Duke Power

May 23, 2002

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Statistical Methodology

- **One sided upper tolerance limit uncertainties**
- **Total statistically combined uncertainty factor is calculated as follows**

  $$SCUF = 1 - bias + sqrt\ [\ (K_a\sigma_a)^2 + (K_{p1}\sigma_{p1})^2 + (K_{p2}\sigma_{p2})^2\ ]$$

- **K factor insures with a 95% confidence level that 95% of local power predictions are equal to or larger than the measured value**

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Statistically Combined Uncertainty Factors

	Bias Calc - Meas	Assembly Statistical Deviation Kσ	Pin Statistical Deviation Kσ	C4 to SIM Statistical Deviation Kσ	Total Uncertainty Factor
Saint Laurent B1 LEU Fuel					
FΔh					
Fq					
Fz					
Saint Laurent B1 MOX Fuel					
FΔh					
Fq					
Fz					
McGuire / Catawba LEU Fuel					
FΔh					
Fq					
Fz					
McGuire / Catawba MOX Fuel					
FΔh					
Fq					
Fz					

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Summary

- **CASMO-4 / SIMULATE-MOX core models can predict power distributions in mixed cores of LEU and MOX fuel with a statistical accuracy which is similar to the accuracy for cores containing only LEU fuel**

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Dynamic Rod Worth Measurement

Meeting with Nuclear Regulatory Commission

DPC-NE-1005P Review

Scott B. Thomas
Duke Power
May 23, 2002

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Overall Objectives

- **Update current analytical methods for DRWM calculations**
- **Demonstrate acceptability of DRWM technique for mixed LEU-MOX cores**

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DRWM Background

- Efficient method to measure bank worth for physics testing
- Westinghouse DRWM Technique (WCAP-13360-P-A)
- Duke technology transfer documented in DPC-NE-2012A
- Duke has calculated DRWM analytical factors for 13 tests at Catawba and McGuire
- Measurement requires analytical factors to correct for flux redistribution and delayed neutron effects

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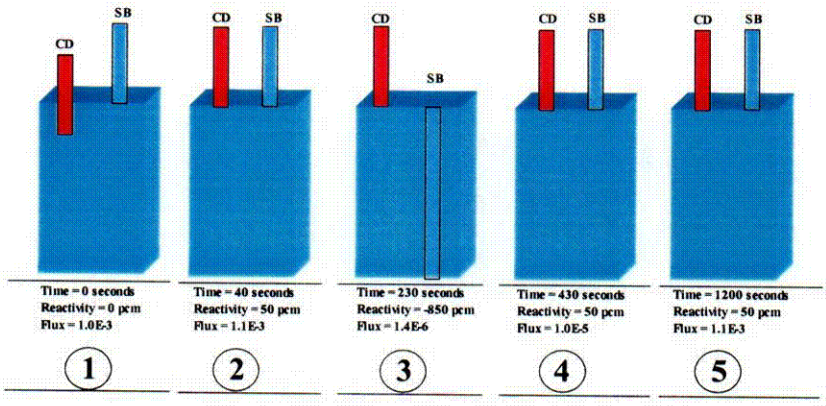
DRWM Measurement Process



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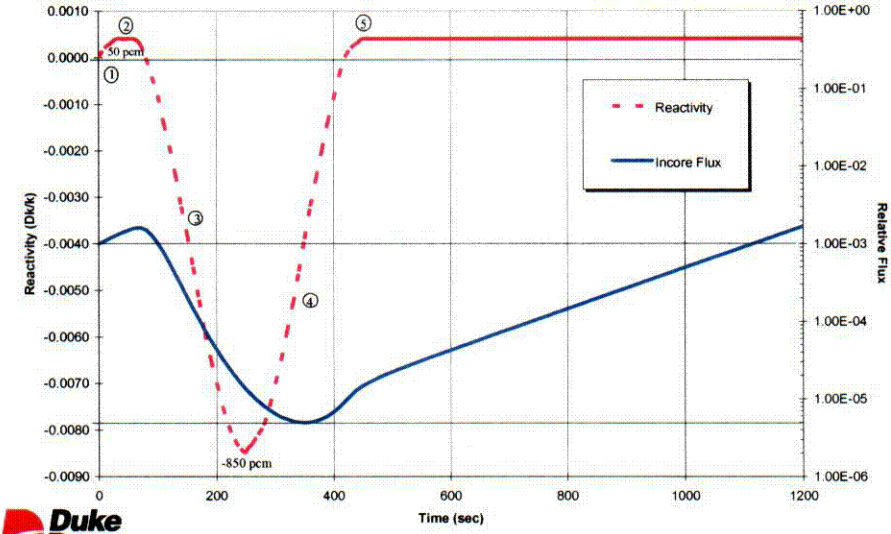
C-07

DRWM Measurement Process



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Technology Transfer Criteria

- **A set of five criteria was established for technology transfer**

  1) **Eligibility of Codes for DRWM Computations**
  2) **Application of Procedures to DRWM Computations**
  3) **Training and Qualification of Utility Personnel**
  4) **Comparison Calculations (Utility vs. Westinghouse)**
  5) **Quality Assurance and Change Control**

- **Responses provided in DPC-NE-2012A for criteria 1, 2, 3, and 5 are still applicable with new codes**

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C-08

Comparison Calculations

- Calculations contained in DPC-NE-2012A were repeated with new codes
- Extensive comparisons to Westinghouse results for 6 cycles
  - Catawba 1 Cycle 11, and Cycle 12
  - Catawba 2 Cycle 10
  - McGuire 1 Cycle 13
  - McGuire 2 Cycle 12, and Cycle 13
- Comparisons based on "Acceptable Deviation" criteria

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Individual Bank Worth Comparison

Predicted Bank Worth (W-D)

Acceptable Deviation - +/- 25 pcm
(W-D) = (Westinghouse – Duke

Measured Bank Worth (W-D)

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Total Bank Worth Comparison

Predicted Bank Worth %(W-D)/W

[ ]

Acceptable Deviation - +/- 2%
(W-D) = (Westinghouse – Duke)

Measured Bank Worth %(W-D)/W

[ ]

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Summary - Comparison Calculations

- **Measured bank worths are nearly identical**
- **Predicted differences are well within expected range for a comparison of two independent methodologies**
- **CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX codes are suitable replacements for Westinghouse codes**

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DRWM MOX Considerations

- Measured excore signal quality (signal/noise ratio)
  - Lower core average delayed neutron fraction
  - Higher fast to thermal neutron flux ratio
- Predicted excore signal
  - Importance of treating spatial fission energy spectra
- Deduced bank worth error sensitivity
  - Model error impact on measured worth

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MOX DRWM Simulations

- Evaluation performed by North Carolina State University
  - CASMO-3/NESTLE and DORT/TORT

"Evaluation of the Effects of Mixed LEU-MOX Core on Dynamic Rod Worth Measurement", I. Ariani, P. J. Turinsky, Electric Power Research Center, North Carolina State University

- Measurement process simulated using a reference core model and perturbed core models
  - Deduced measured bank worths evaluated for errors in control rod cross-sections, power distribution, and delayed neutron fraction
  - Both LEU and mixed LEU-MOX cores were modeled

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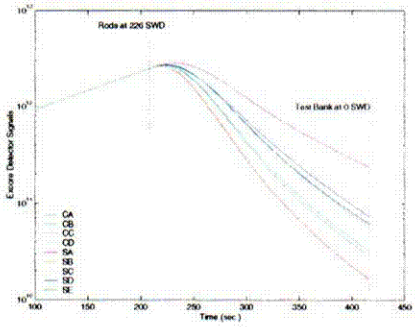
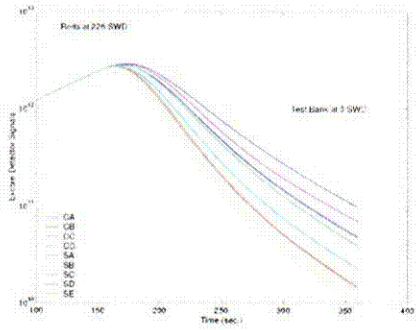
Simulate Excore Detector Signals

LEU Core



Mixed LEU-MOX Core



Relatively small change in minimum measured excore signal

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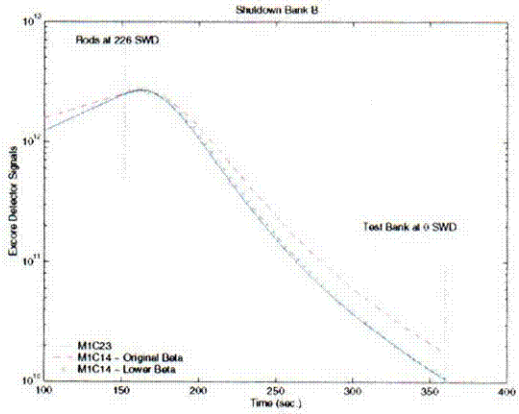
Impact of Lower Delayed Neutron Fraction



- **Change in excore signal due almost entirely to different delayed neutron fractions**

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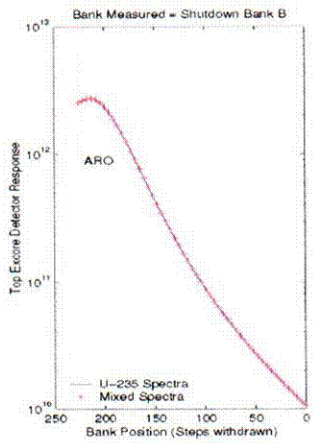
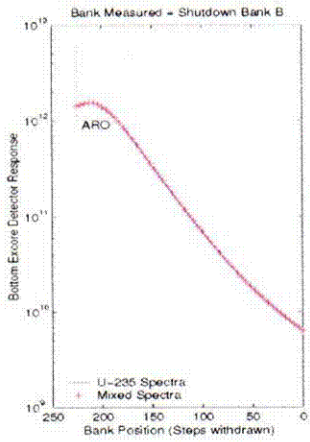
C-09

Energy Spectra Impact on Predicted Excore Response

Top Detector



Bottom Detector



- **Treatment of explicit energy spectra is not necessary**

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Summary –DRWM MOX Considerations

- **Measured excore signal quality essentially unchanged for mixed LEU-MOX core**
  - **Lower delayed neutron fraction produces a relatively small decrease in minimum measured signal**
- **Energy spectra differences shown to be insignificant**
  - **U-235 spectra compared to explicit mixed spectra**
  - **Normal excore weight factors are acceptable**

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C-10

Summary –DRWM MOX Considerations (continued)

- **Sensitivities of deduced bank worths on model errors are essentially the same for LEU and mixed LEU-MOX cores**
  - **Control rod absorption cross section**
  - **Fission neutron density distribution errors**
  - **Delayed neutron fraction**

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Conclusions

- **CASMO-4/SIMULATE-3 MOX/SIMULATE-3K MOX codes are suitable replacements for Westinghouse codes**
- **Existing DRWM methodology is applicable to mixed LEU-MOX Cores**

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