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DATE OF MEETING
5/23/02

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Docket Number(s)	50-413, 50-414, 50-369, and 50-370
Plant/Facility Name	Catawba & McGuire Nuclear Station
TAC Number(s) (if available)	MB2726, MB2729, MB2578, MB2579
Reference Meeting Notice	May 10, 2002
Purpose of Meeting (copy from meeting notice)	To discuss Topical Report DPC-NE-1005P, Nuclear Design methodology using CASMO-4/SIMULATE-3 MOX, Submitted by Duke for Catawba & McGuire Nuclear Station.

NAME OF PERSON WHO ISSUED MEETING NOTICE	TITLE
Chandu P. Patel	Project Manager

OFFICE
NRR

DIVISION
DLPM

BRANCH
PD-II

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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	

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

DUKE COGEMA
STONE & WEBSTER

Duke Engineering & Services. | COGEMA, Inc. | Stone & Webster

FRAMATOME | NFS | Duke Power.

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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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Duke Nuclear Analysis Methodologies

Meeting with Nuclear Regulatory Commission

- DPC-NE-1005P Review -

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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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CMS - Core Management System
Studsvik Scandpower, Inc.

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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

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USA

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Norway

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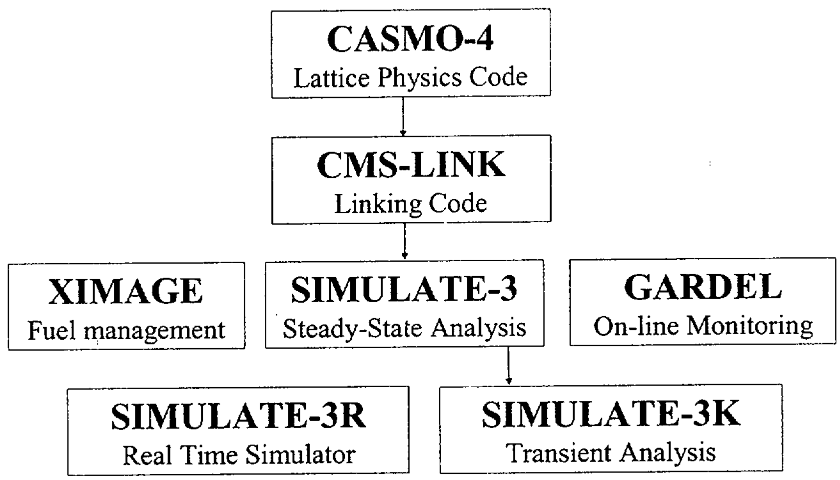
Germany

Studsvik Scandpower
Switzerland

Studsvik Japan
Japan

Studsvik Scandpower CMS



CMS Has Wide Range of Applications

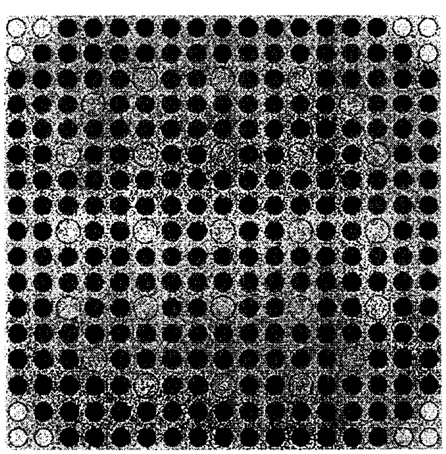
- 11 Countries
- 55 Companies
  - Nuclear Utilities
  - Nuclear Fuel Vendors
  - Regulatory Agencies
  - National Laboratories
  - Universities
- Applied to more than 70 BWRs and 90 PWRs
- Applied to more than 2000 BWR and PWR Cycles

CASMO-4 Lattice Physics Code

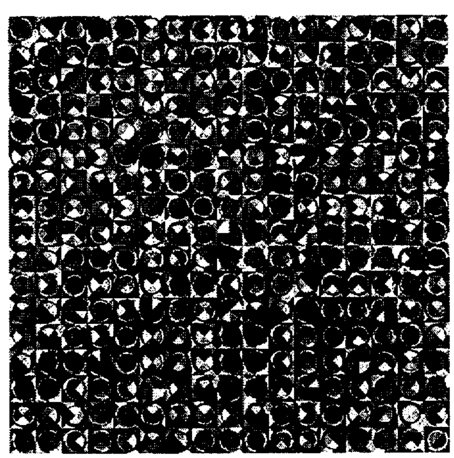
- Purpose:
  - Analyze the detailed behavior of a fuel bundle over its lifetime
  - Treat fuel, burnable absorbers, control rods, instruments, water gaps
  - Provide bundle data for downstream core analysis codes

- 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

2-D Heterogeneous Transport Model

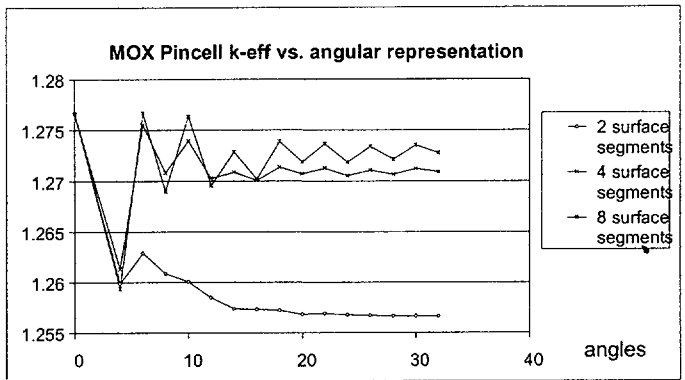


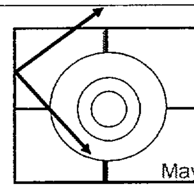
Physical Geometry



"Flat Source" Representation

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





Surface segment 1

Surface segment 2

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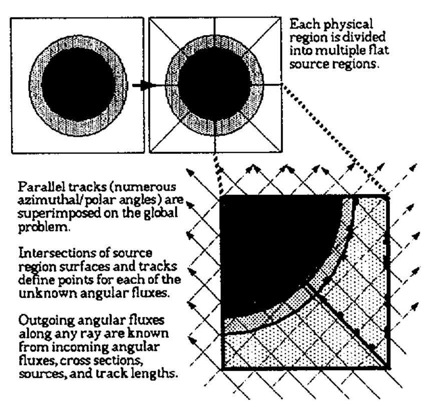
Method of Characteristics (MOC)

The Characteristics Method

The equation being solved is the *solution* to the characteristics form of the Boltzmann transport equation...

$$\Phi_{out} = \Phi_{in} exp\{-\Sigma s\} + \frac{Q}{\Sigma}[1 - exp\{-\Sigma s\}]$$

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



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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

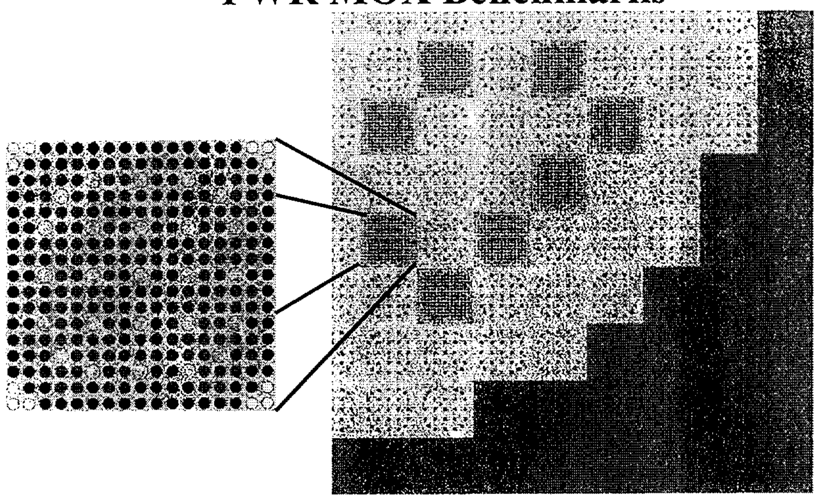
- First MOX analysis applications in 1989
- Goal: achieve MOX/LEU mixed-core accuracy comparable to that of LEU cores
- Continuous development throughout the 1990's
- Currently used in Germany, Switzerland, UK, Japan, and the U.S.

- 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

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**

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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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

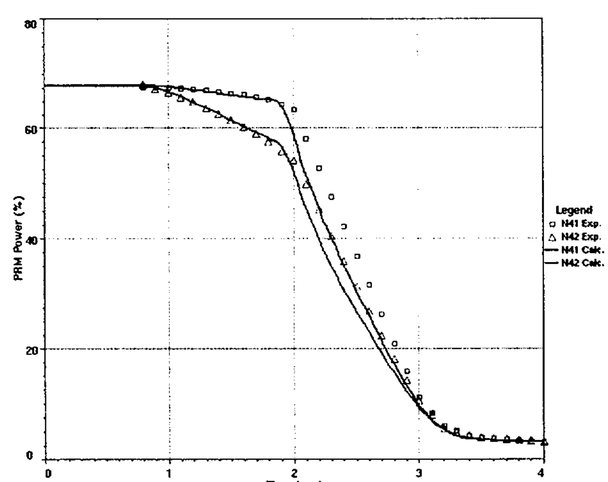
- Fuel temperatures are computed using an explicit fuel pin conduction model
- Coolant densities are computed using an explicit channel hydraulic model
- At HZP, pin conduction and channel hydraulic differences have zero effect on computations
- All MOX enhancements are identical in S3 and S3K

SIMULATE-3K Applications

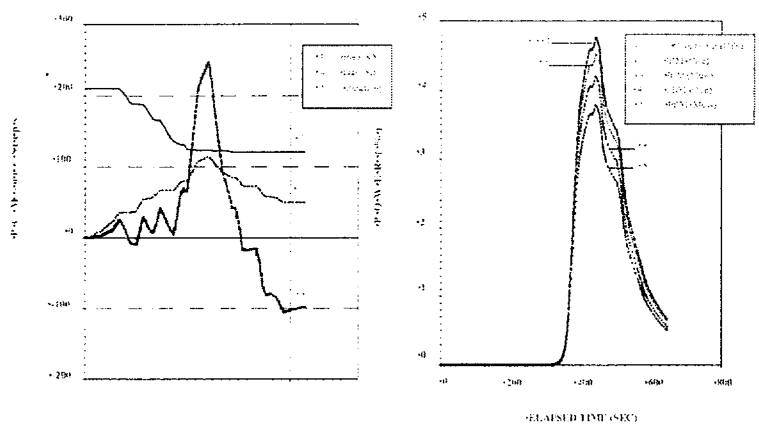
- PWRs:
  - Reactivity excursions (ejected bank/rod)
  - Dynamic rod worth measurements (DRWM)
  - Dropped rod transients
  - Steam line break analysis
  - Boron dilution accidents

- BWRs:
  - Stability analysis
  - SCRAM reactivity curve generation
  - Reactivity excursions (dropped/stuck rod)
  - Operational transients (e.g., pump trips, turbine trips)

Ringhals PWR Dropped Rod Analysis



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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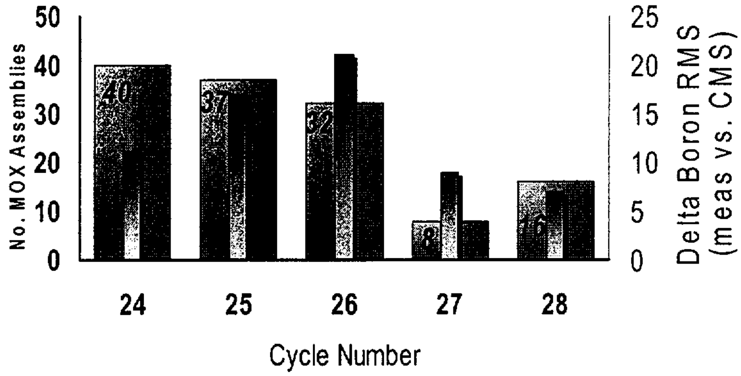
KKB1 (Beznau Unit 1) Boron Results by Cycle



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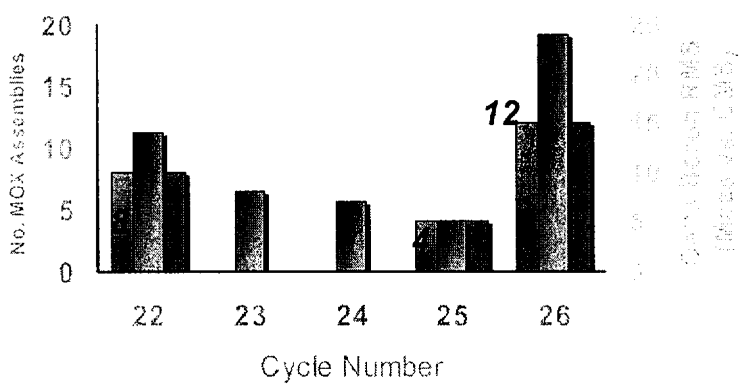
KKB2 Boron Results by Cycle



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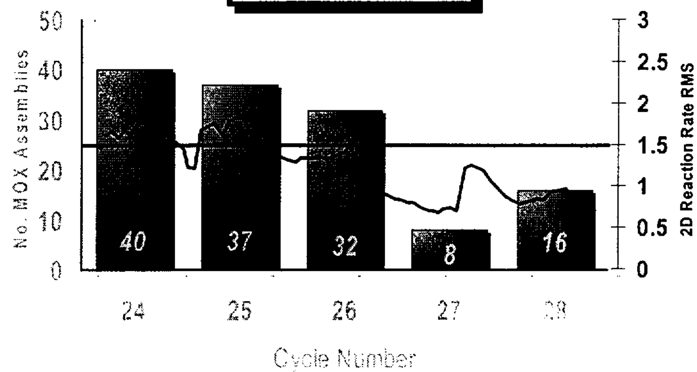
KKB1 2D Reaction Rates

RMS 1.33%



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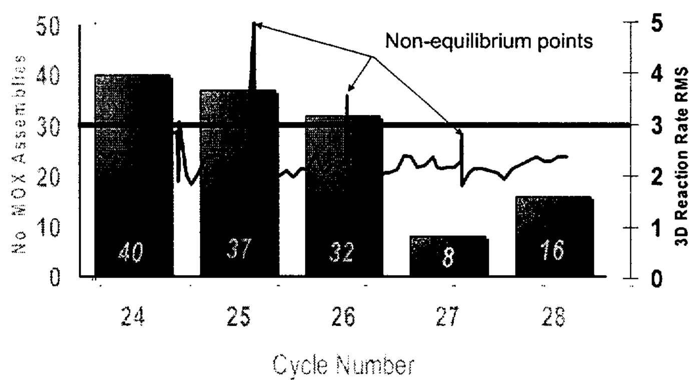
KKB1 3D Reaction Rates

RMS 2.36%



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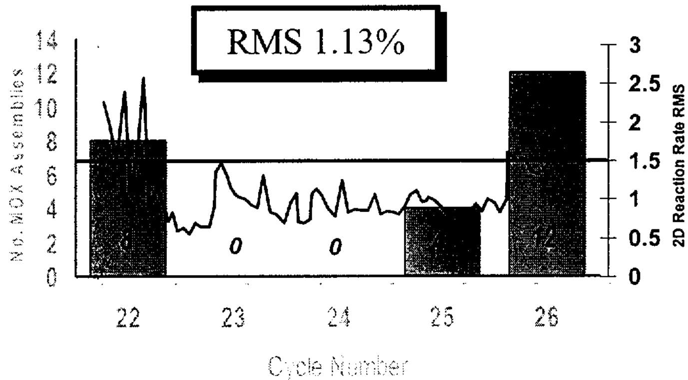
KKB2 2D Reaction Rates



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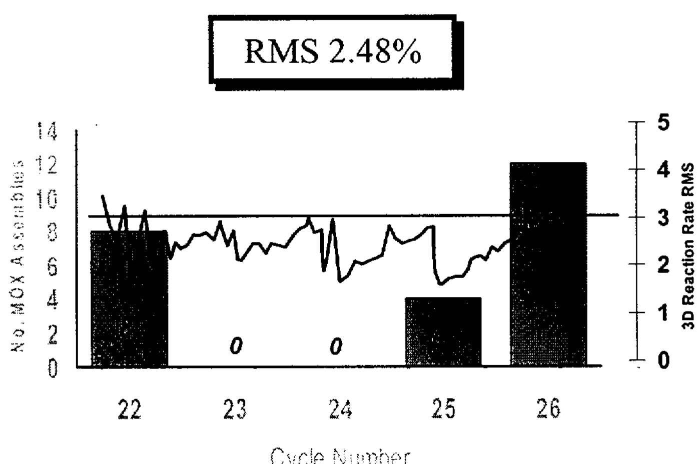
KKB2 3D Reaction Rates



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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.

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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 <u>assembly</u> power

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Core Design Methodology

- Requires a conservative verification of multiple fuel <u>pin</u> 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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