Neutronics Codes Currently Used in Japan for Fast and Thermal Reactor Applications

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  Under construction 3

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  Monju

  JOYO

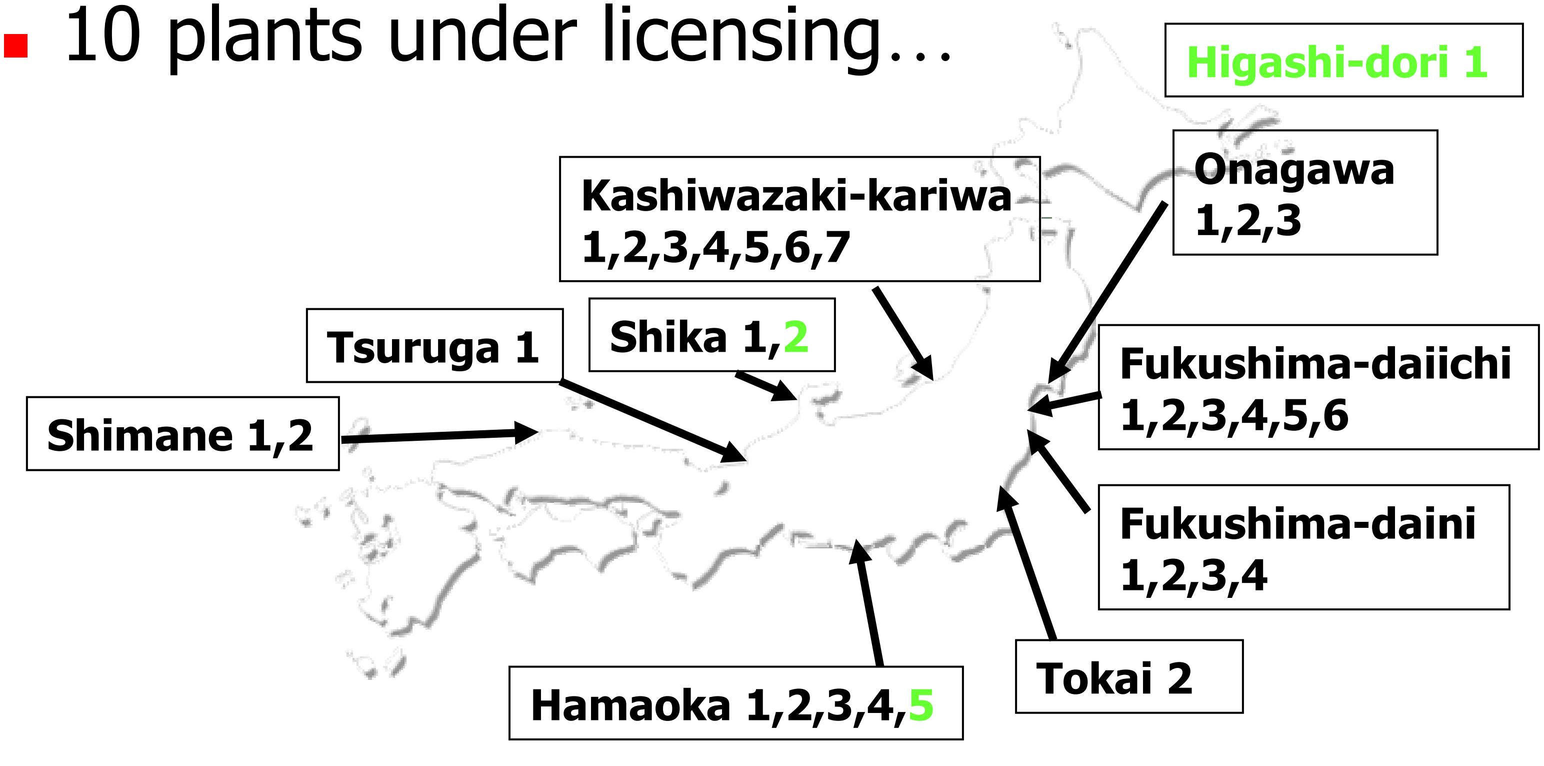
- Experimental Reactor

  JEARI

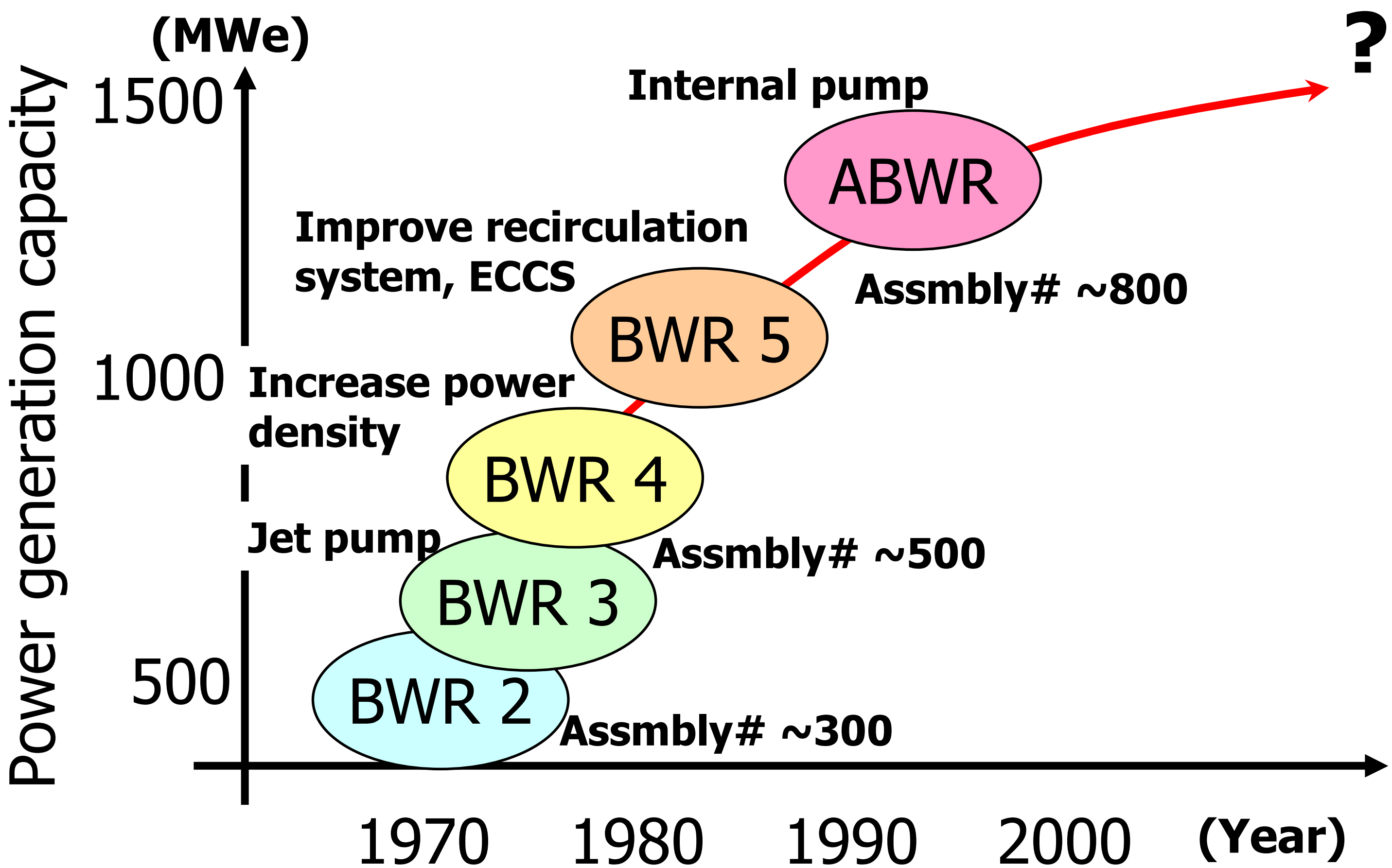
  University

2. BWRs in Japan

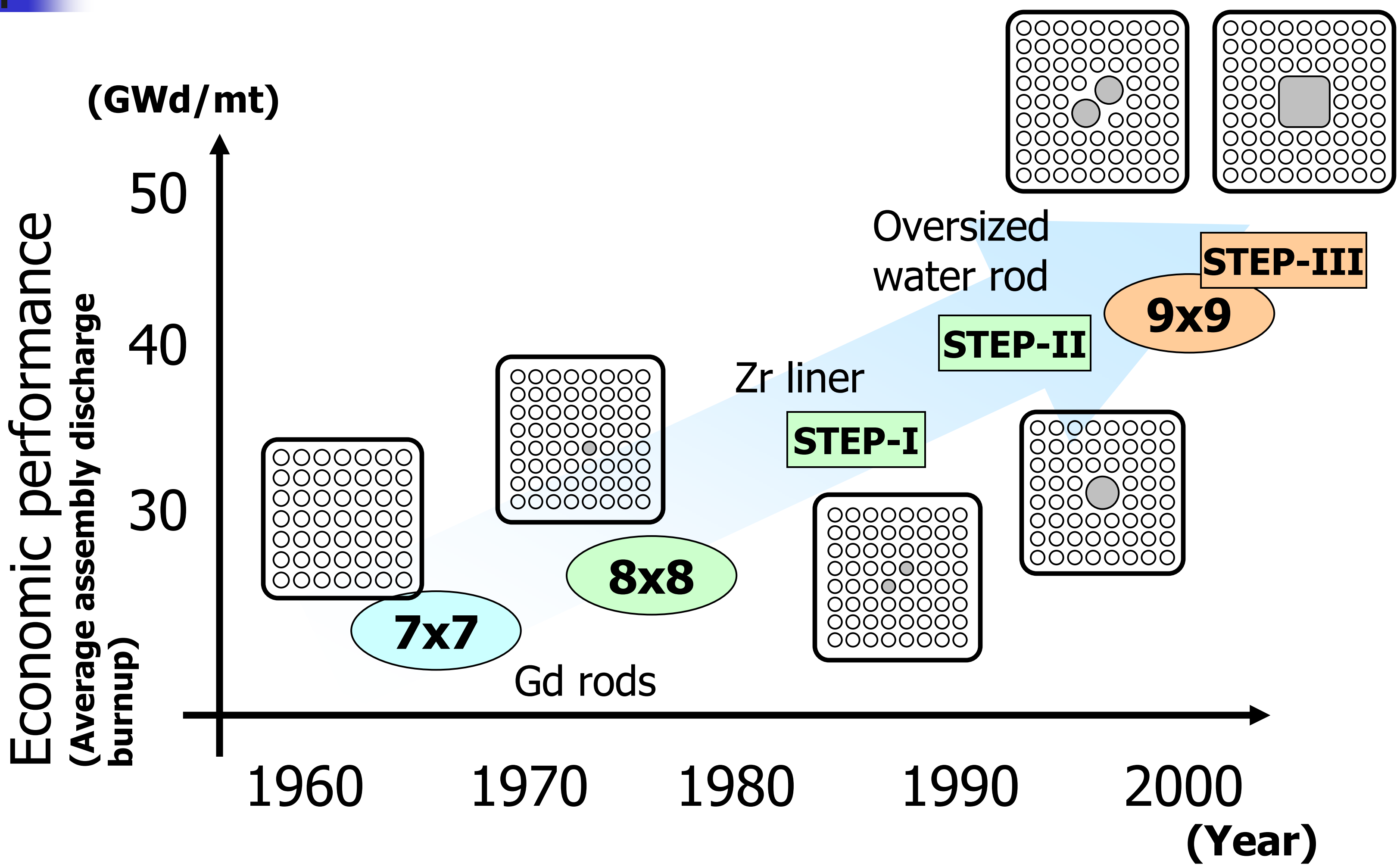
- Status of Japanese BWRs
  - 29 plants operating (BWR-2,3,4,5,ABWR)
  - 3 plants under construction (BWR-5,ABWR)
  - 10 plants under licensing…



History of Japanese BWRs



History of BWR Fuels

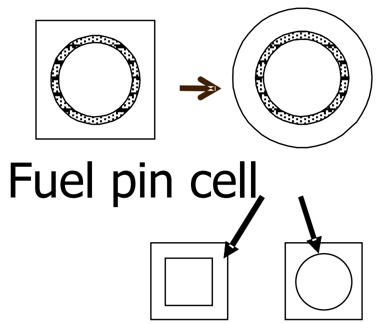


Core simulators

- BWR core simulators and users

	Code share in reload design
TGBLA/LOGOS, TGBLA/PANACEA	~48%
GNF-J, GIS	
CASMO/SIMULATE	~45%
TEPSYS,TOiNX	
NEUPHYS/COS3D	~7%
NFI,CTI	

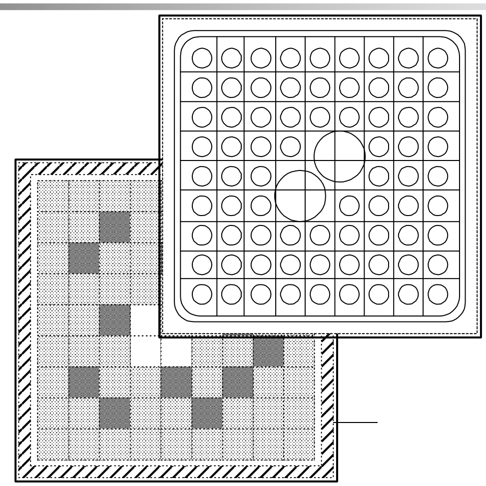
BWR core calculation



Effective macroscopic cross sections

<u>Pin cell calculation</u>

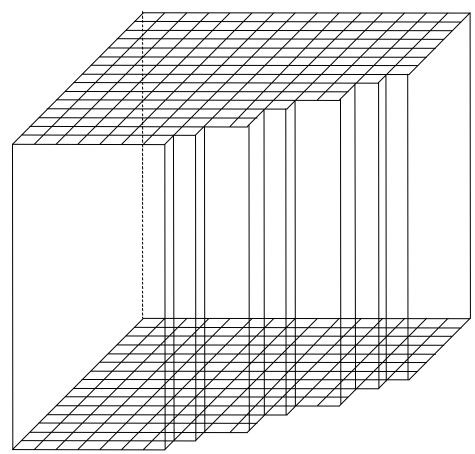
- **Cylindrical transport calculation**
- **Fine energy groups**
- **Resonance calculation**



<u>Assembly calculation</u>

- **2D calculation**
- **Few/several energy groups**
- **Diffusion/transport theory**
- **Homo/heterogeneous geometry**

Assembly constants



<u>Core calculation</u>

- **3D calculation**
- **few energy groups**
- **diffusion theory**
- **FDM/ANEM**
- **Void feedback**
- **Doppler feedback**

TEPSYS CASMO/SIMULATE system

- Developed by Studsvik Scandpower (SSP)
- Vendor-independent code system
- Standard core simulator of TEPSYS
  - BWR reload core design
  - Fuel bid evaluation
  - On-line core monitoring system (adaptive model)

A subsidiary of Tokyo Electric Power Company

Experiences

- Applied more than 70 reload cycles to 13 TEPCO BWR plants from 1988
- Introduced CASMO/SIMULATE core monitoring system to 2 TEPCO BWR plants in 1999 and 2002

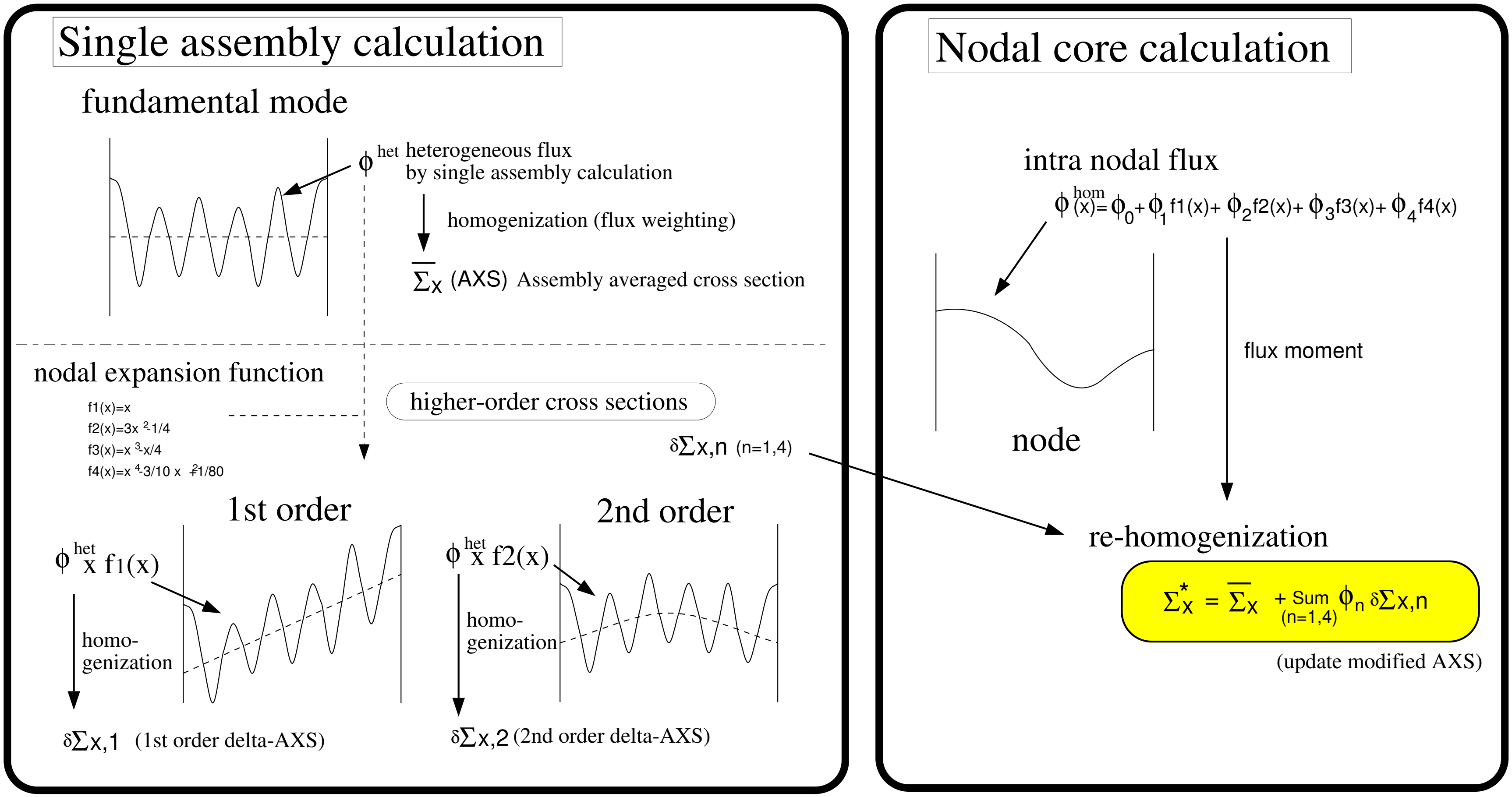
Computational methods used in CASMO/SIMULATE

- CASMO
  - Pin cell transport + Assembly transport
  - Explicit geometry analysis by Method of Characteristics (MOC)
  - Extended version (Capability of 2D whole core analysis) (2000~)
- SIMULATE
  - Advanced nodal expansion method (ANEM)
  - BWR enhancement models (2000~)

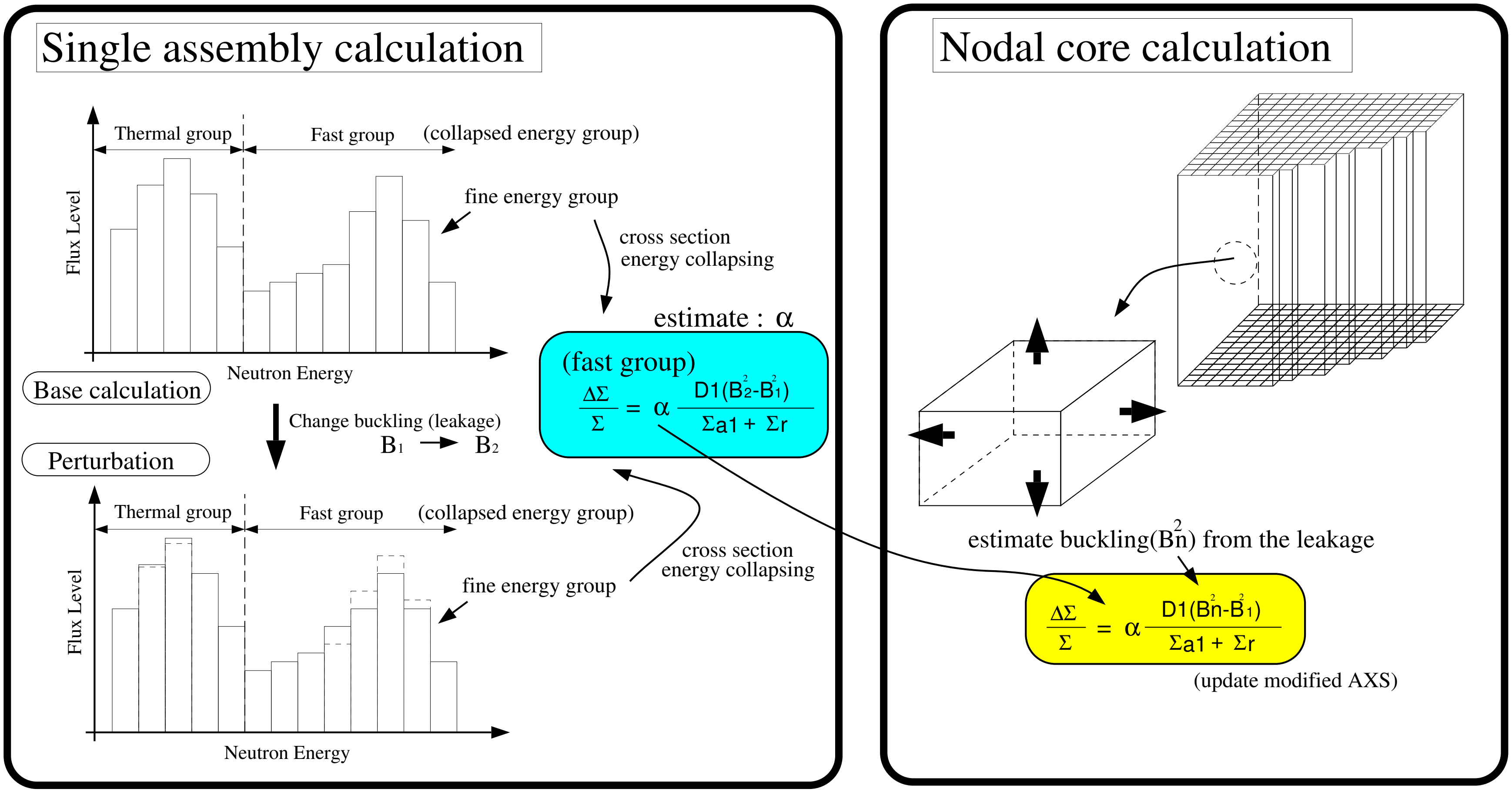
Enhancement models for BWR analysis

- Spatial rehomogenization model
  - Correction of assembly averaged cross section (spatially)
- Fast leakage model
  - Correction of collapsed cross section (energy spectrally)
- Dynamic discontinuity factor model
  - ADF adjustment by empirical procedure
  - Determined by results of reference explicit geometry transport core calculations

Spatial rehomogenization



Fast leakage model



Verification of SIMULATE neutronics model vs CASMO-4 reference core analyses

	RMS of Assembly Powers(%)		Error in rod worth (pcm)	
Case	Standard	Enhanced	Standard	Enhanced
BOC Un-Rodded	1.36	0.76	-22	+10
BOC Rodded	1.52	0.59		
MOC Un-Rodded	1.33	0.59	+49	+0
MOC Rodded	1.70	0.53		
EOC Un-Rodded	1.33	0.59	-26	-8
EOC Rodded	1.57	0.64		

Global Nuclear Fuel Japan

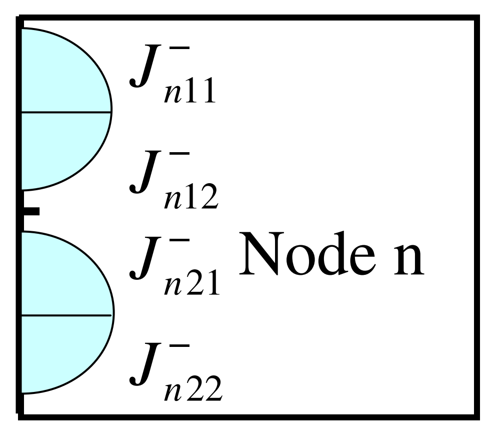
- A joint Venture of GE, Toshiba & Hitachi
  - In, January 2000, GNF-J started as a part of Global Nuclear Fuel LLC – A joint venture of GE, Toshiba & Hitachi.
  - GE Nuclear Fuel Joined as GNF-A
  - GNF provides nuclear fuel and services, from fuel design, manufacturing to the core management

Lattice Physics Code in GNF-J

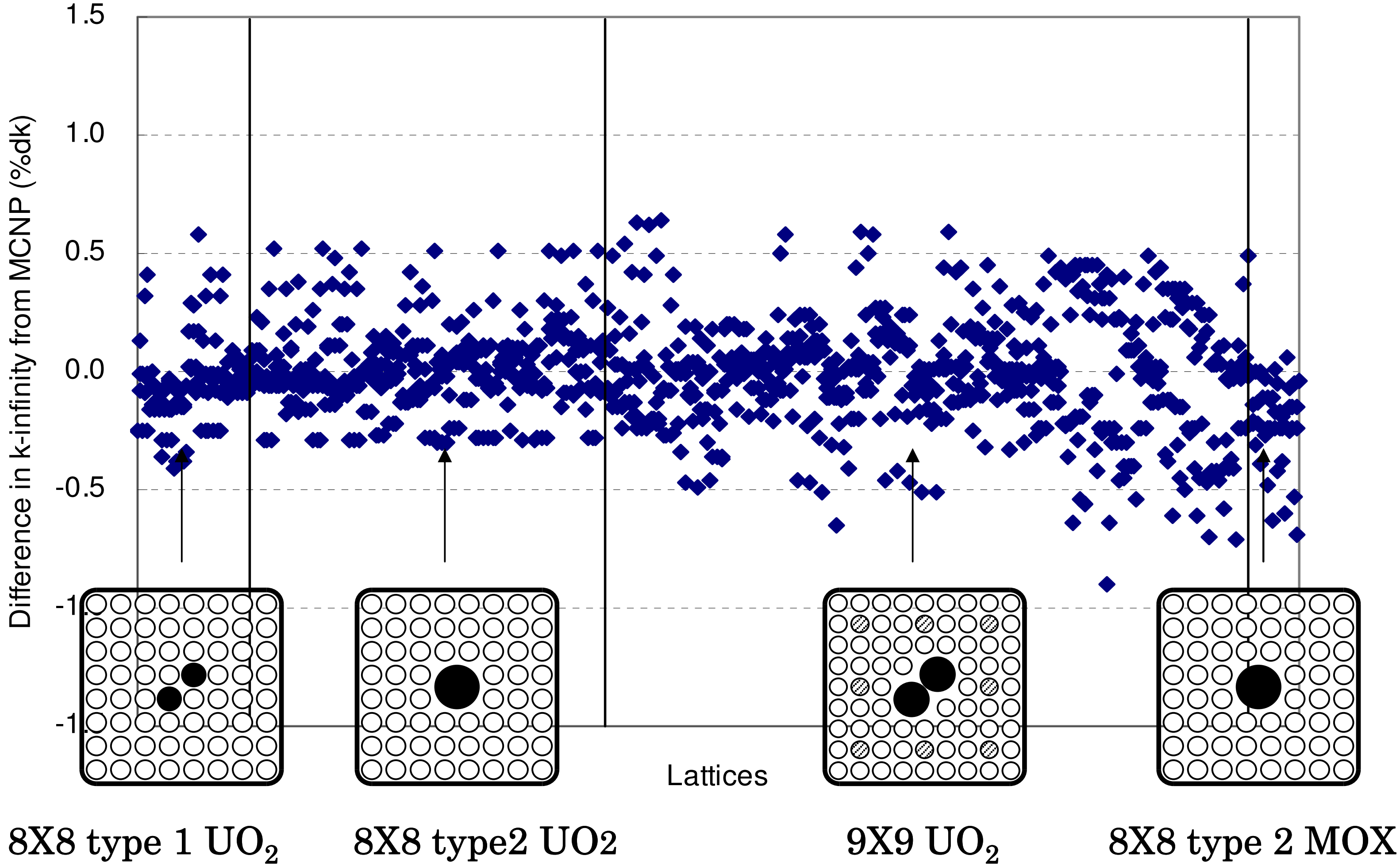
- TGBLA(1979-2004)
  - Pin transport + assembly diffusion
  - L-method (1984) accounts for spectrum mismatch effects in pin transport calc.
  - Gd rod asymmetric depletion (1987)
  - RICM option solves ultra-fine group slowing down for resonance shielding (1987)

Lattice Physics Code in GNF-J

- LANCER(2000-)
  - Improved CCCP method yields sufficient accuracy within reasonable computing costs.
  - 35 energy group 2D flux calculation provides accurate solution for highly heterogeneous lattices
  - Interface currents are expanded with 2 spatial X 2 angular modes.

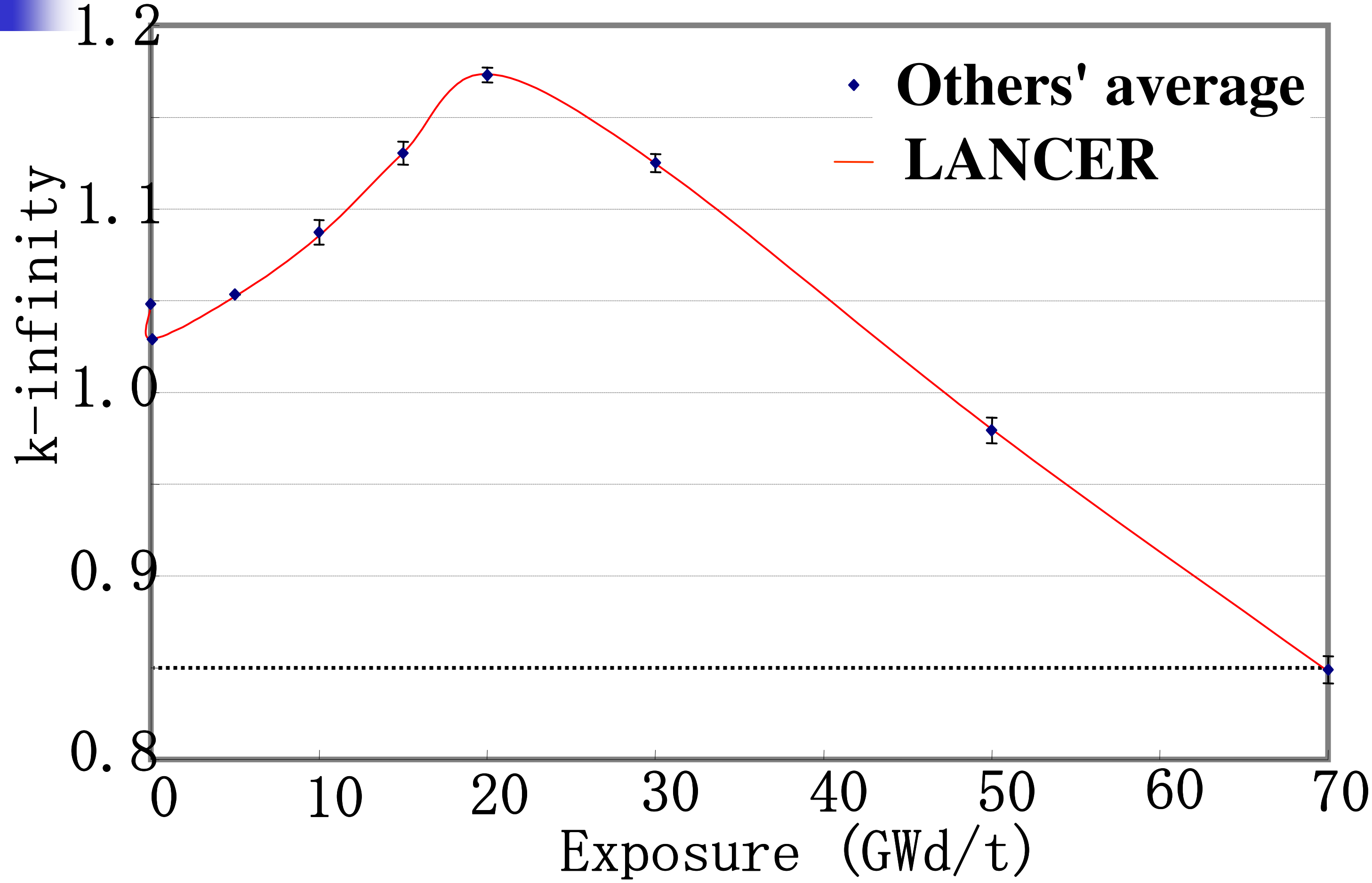


Lattice Physics Code in GNF-J



<u>Error in K-inf to MCNP4C in BOL Fuel Lattices</u>

Lattice Physics Code in GNF-J



K-inf of AESJ Benchmark UO2 Assembly

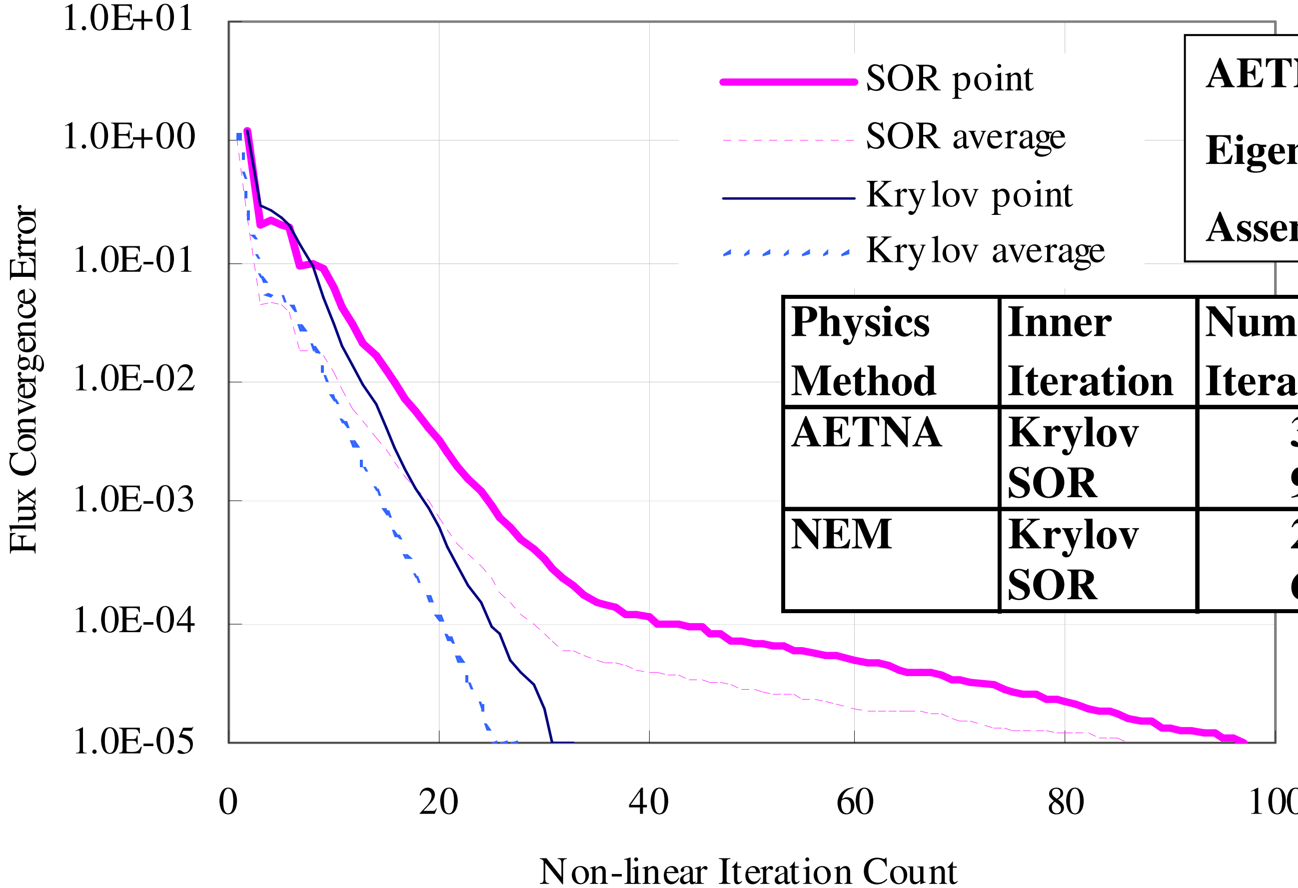
Core Physics Code in GNF-J

- LOGOS(1992-2004)
  - Developed based on PANACEA (GE).
  - Improved 1.5 group (quasi 2-group) model solves intranodal thermal flux analytically, accounts for spectral mismatch effects between assemblies
  - Pin power reconstruction model based on the analytic thermal flux expansion.
- PANACEA v.11 (1997-2004)
  - Almost the same improvements as LOGOS

Core Physics Code in GNF-J

- AETNA (2000-)
  - 3 group analytic-polynomial nodal expansion method with iterative expansion coefficient update.
  - Krylov (GMRES) method for inner iteration along with NODEX type non-linear eigenvalue iteration provides faster convergence
  - Consistent pin power reconstruction model to the flux solution.
  - Pu-241 tracking model for decay correction

Core Physics Code in GNF-J



AETNA error

Eigenvalue 0.003%

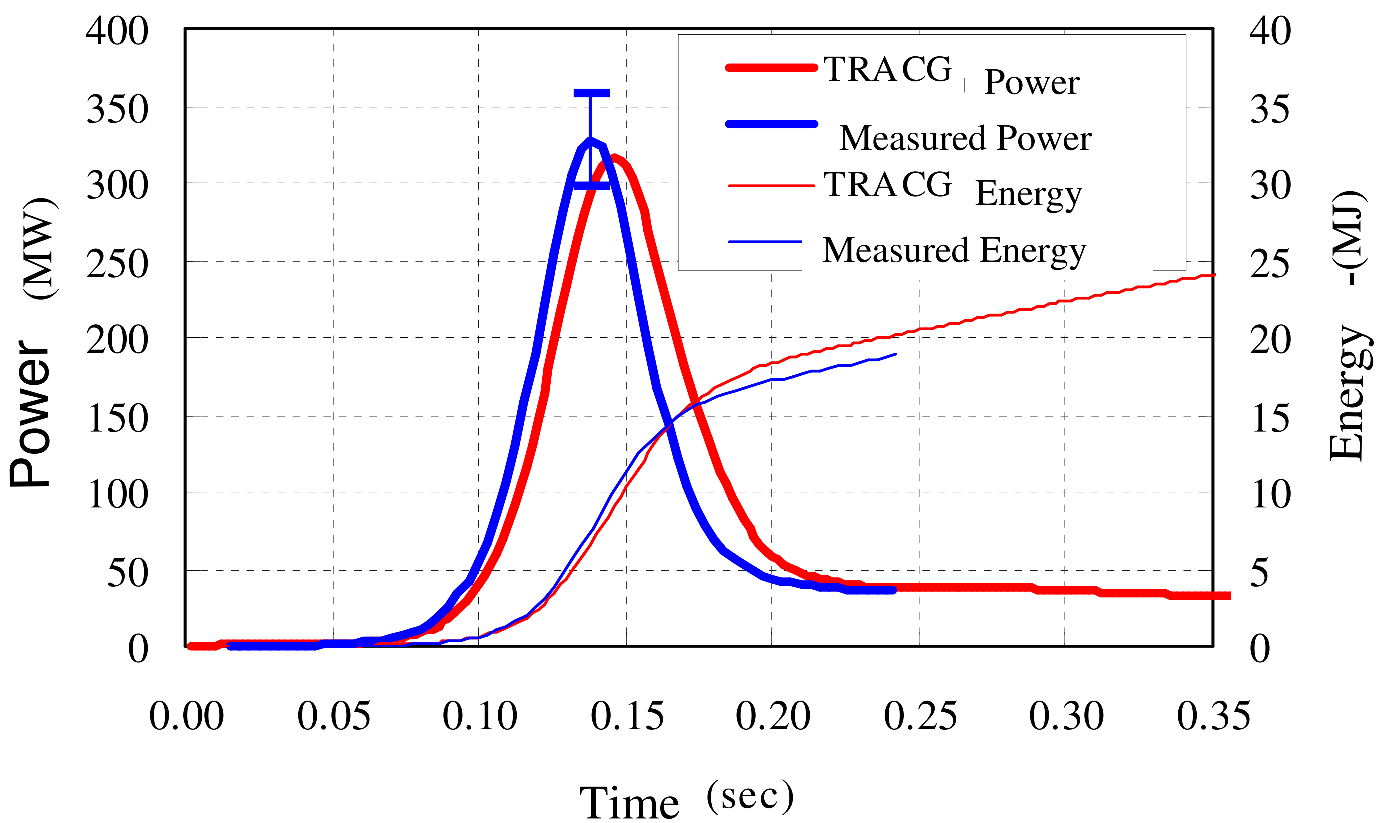
Assembly power 0.19%

Physics Method	Inner Iteration	Number of Iterations[1]	Computing Time[2] (sec)
AETNA	**Krylov**	**33**	**4.62**
	SOR	**97**	**9.81**
NEM	**Krylov**	**26**	**9.30**
	SOR	**69**	**22.60**

LRA 3D BWR Static Benchmark

Core Physics Code in GNF-J

AETNA/TRACG05 coupled nuclear thermal hydraulics code for plant transient analysis



SPERT III E-core RIA Test-81 (hot standby)

BWR Analysis Codes; NFI

- Nuclear Fuel Industries, Ltd.'s Standard Neutronics Code System
  - Assembly Analysis Code : NEUPHYS
    - Fine Group Collision Probability Calculation +
    - Three Groups Diffusion Model
  - BWR Core Simulator : COS3D
    - Modified One Group Diffusion Model
    - Power Correction with Spectrum Mismatch Model

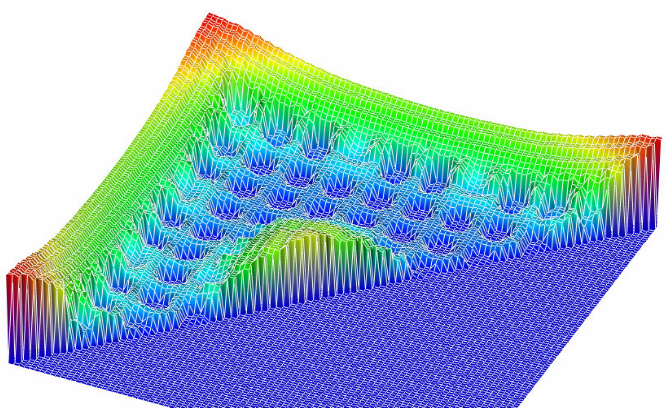
BWR Analysis Codes; NFI

- Applications of NFI Standard System
  - Design of 9X9 type B Assembly
    - Average Discharge Burnup 45GWD/t
    - Assembly Maximum Burnup 55GWD/t
  - BWR Core Management system
- For Further Advanced, More Complex and More Heterogeneous Design
  - NFI Supplies Advanced Neutronics Code System

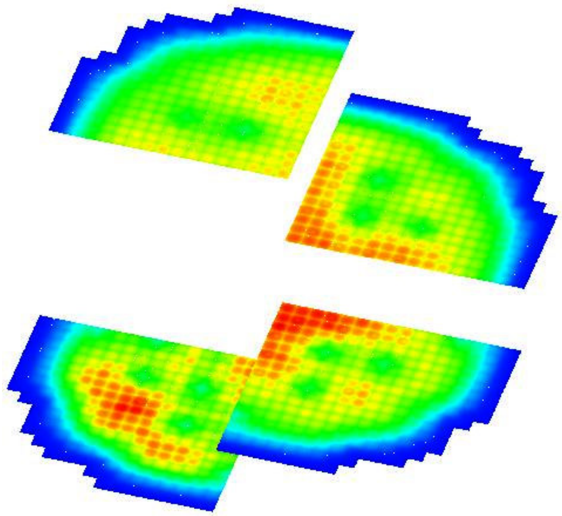
BWR Analysis Codes; NFI

- Advanced NFI Neutronics Codes

- Assembly Analysis Code
  - Arbitrary Geometry Multigroup MOC
  - Brand New Cross Section Library Based on JENDL-3.2
- BWR Core Simulator
  - Few Groups Advanced Nodal Method Core Calculation
  - Intra-Nodal Exposure and Spectrum History Distribution Model



Thermal Flux Distribution

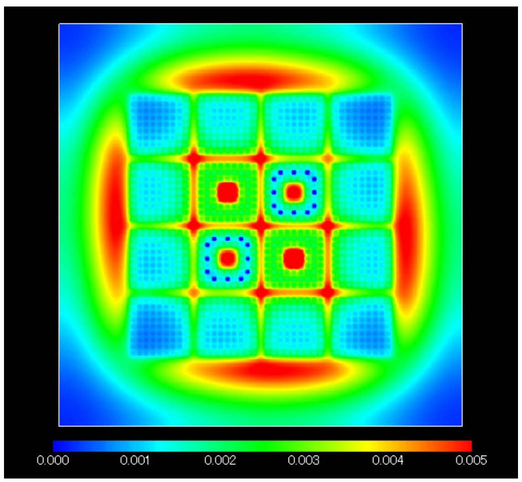


Axial Depending Whole Core Fast Flux Distribution

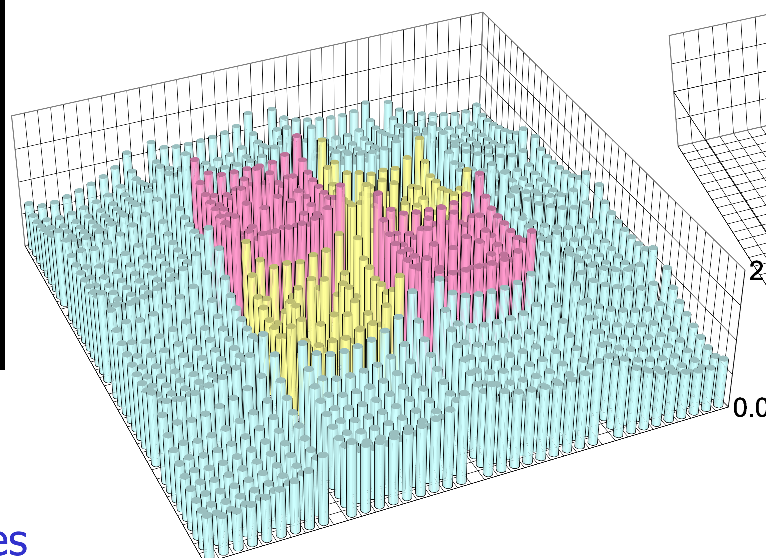
BWR Analysis Codes; NFI

Critical Experimental Analysis

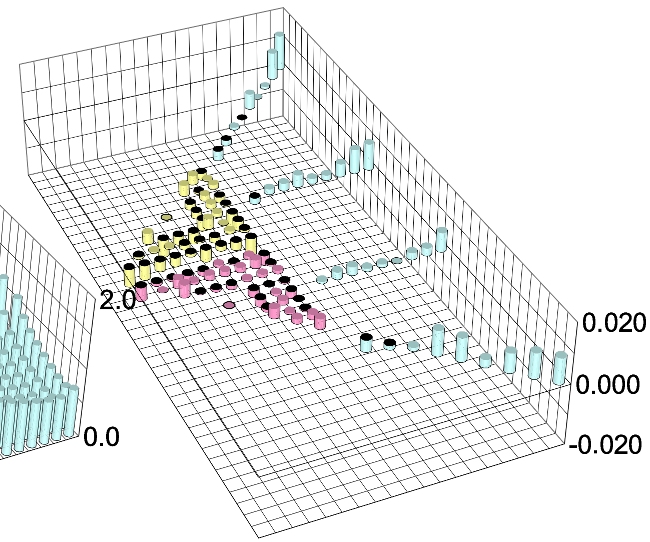
(Comparison with Monte-Carlo Calculation in 2-D Geometry)



Thermal flux distribution



Fission Rate Distribution



Fission Rate Relative Differences
（NFI System-GMVP）

R.M.S=0.0035

CONFIGURATION

- Three Types of Assemblies
  - -9X9 UO2 Driver
  - -9X9 with Large Water Channel
  - -9X9 W/C and Twelve Gd rod
- Water Reflector

3. PWRs in JAPAN

Operating : 23

2-Loop plant : 8

3-Loop plant : 8

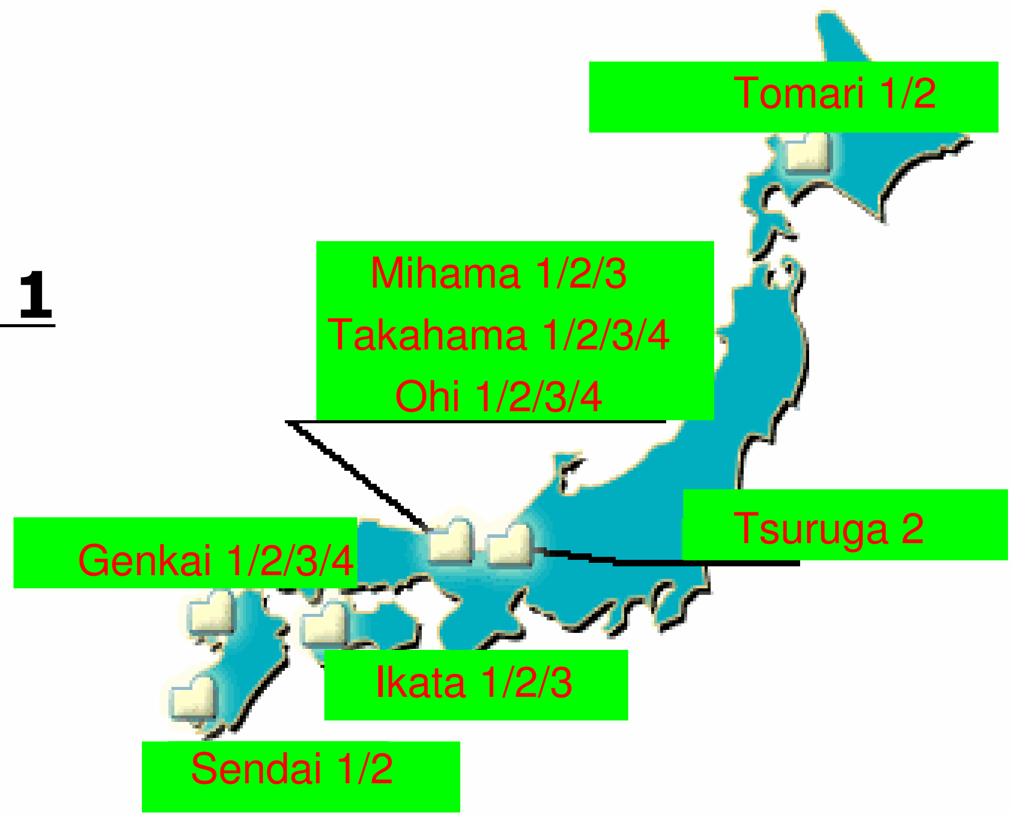
4-Loop plant : 7

Under Construction : 1

Tomari 3 (3-Loop)

Under Licensing : 2

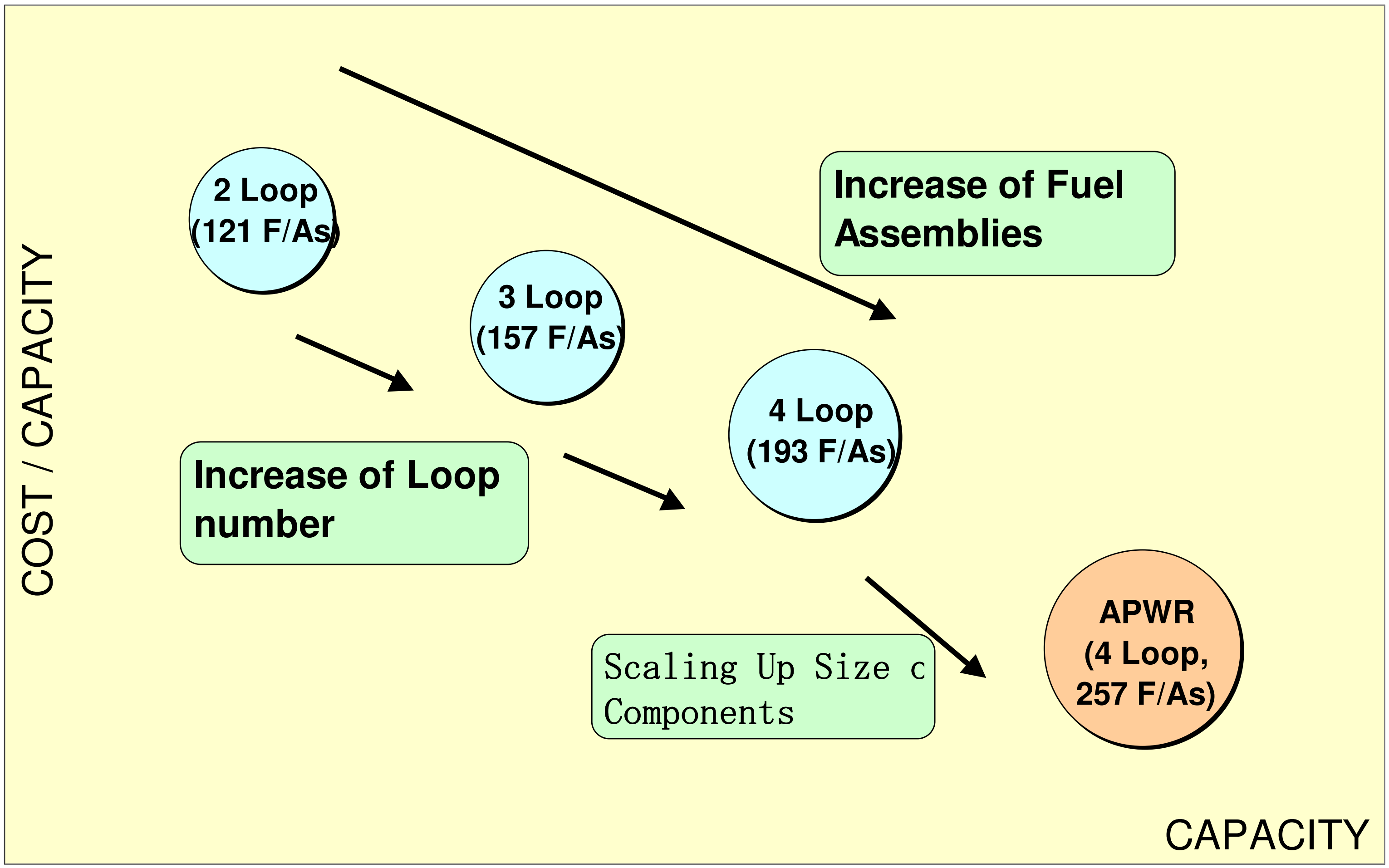
Tsuruga 3&4 (APWR)



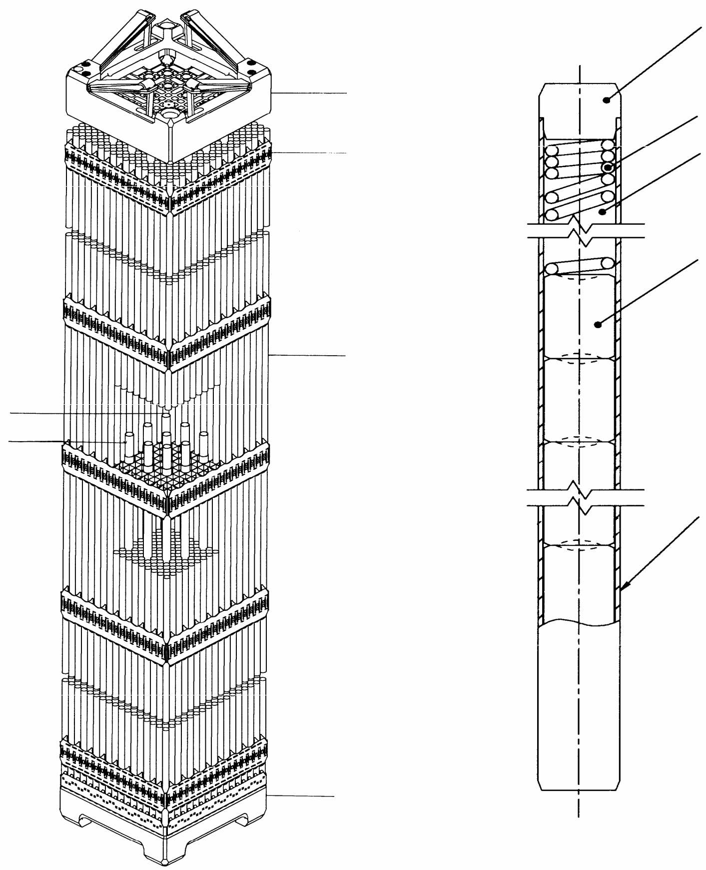
Main Specifications (1/3)

Parameters	Conventional PWR				APWR
No. of primary coolant loop	2	3	3	4	4
Thermal output(MWt)	1,650	2,432	2,652	3,411	4,451
No. of fuel assemblies	121	157	157	193	257
Core equivalent diameters(m)	2.46	3.04	3.04	3.37	3.89
Fuel length(m)	3.66	3.66	3.66	3.66	3.66
Fuel type	14x14	15x15	17x17	17x17	17x17
Ave. linear power density(kW/m)	20.4	20.3	17.1	17.9	17.6
Fuel rod diameter(mm)	10.7	10.7	9.5	9.5	9.5
No. of control rods	29	48	48	53	69

Main Specifications (2/3)



Main Specifications (3/3)



	14x14	15x15	17x17
Fuel rod diameter (mm)	10.7	10.7	9.5
Pellet diameter (mm)	9.3	9.3	8.2
Fuel length (m)	3.66	3.66	3.66
Ave. linear power density (kW/m)	20.4	20.3	17.1 ～ 17.9

Overview of fuel assembly and fuel rod (17x17-type)

Improvement in economical efficiency in conventional PWRs (1/2)

BURNUP LIMIT and CYCLE LENGTH

	～1990	1990～2004	2004～
Burnup Limit of Fuel Assembly	39 GWd/t	48 GWd/t	55 GWd/t
Fuel Enrichment	3.2～3.4	4.1	4.8
Cycle Length	9～12 months	12～13 months	13 ～ months
Type of Burnable Absorber	Cluster-type BP	Mainly Gd (6wt%)	Mainly Gd (10wt%)

Improvement in economical efficiency in conventional PWRs (2/2)

ADVANCED DESIGN FEATURES

(1) Fuel pellet density : 95%TD → 97%TD
(2) Grid : Inconell → Zircaloy
(3) Gadolinium content : 6wt% → 10wt%
(4) Advanced fuel cladding
(5) Debris filter at bottom of fuel assembly

Improvement of core design methodology and core management (1/2)

- Nuclear Design Methodology

  from 1D / 2D Calculation to 3D Calculation

- Thermal Design Methodology

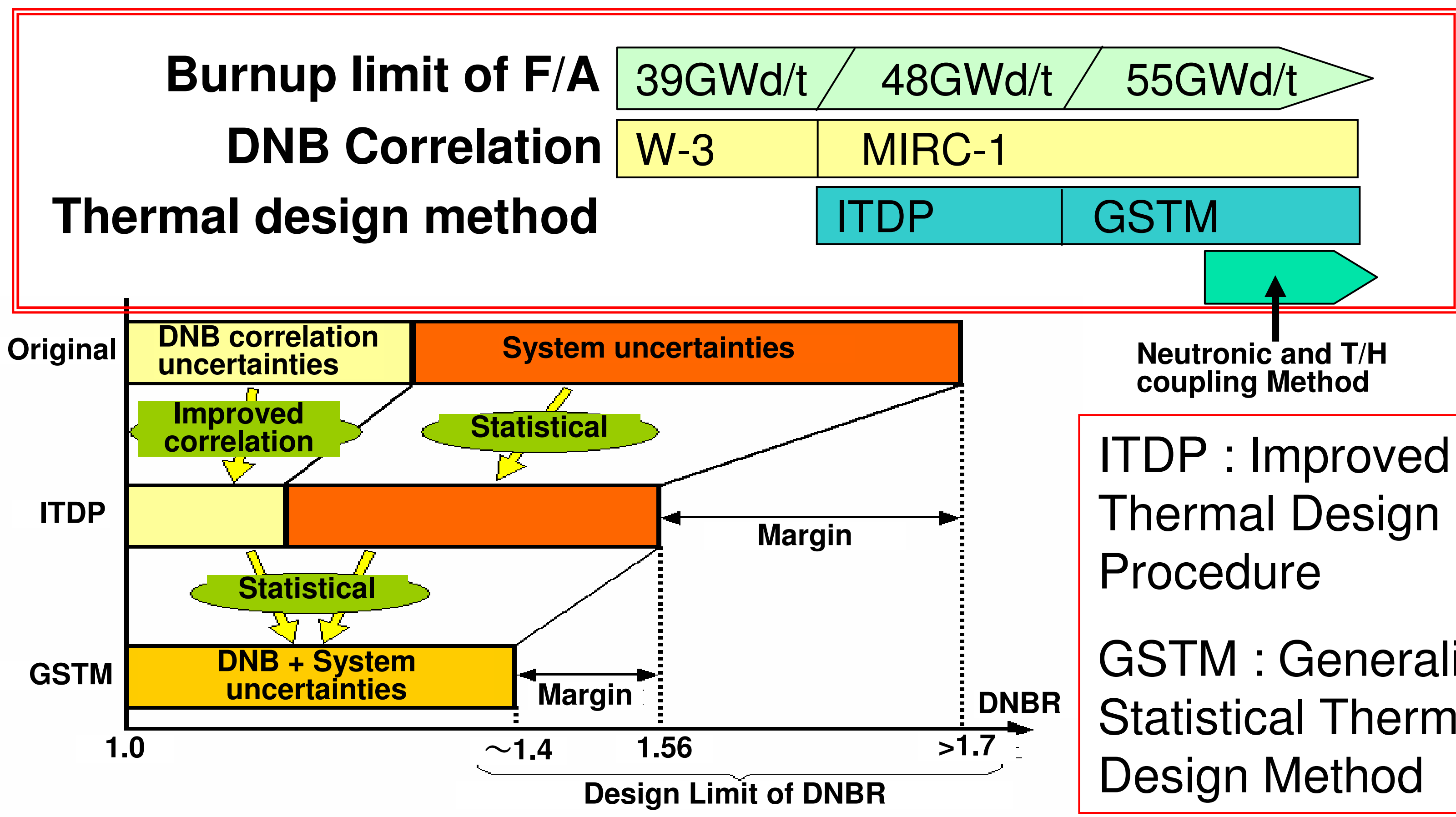
  from ITDP to GSTM

- Core Management

  from M/D, NIS, T/C to 3D On-Line Core Monitor (BEACON™, GARDEL)

BEACON : Best Estimate Analyzer for Core Operation of Nuclear plants

Improvement of core design methodology and core management (2/2)



ITDP : Improved Thermal Design Procedure

GSTM : Generalized Statistical Thermal Design Method

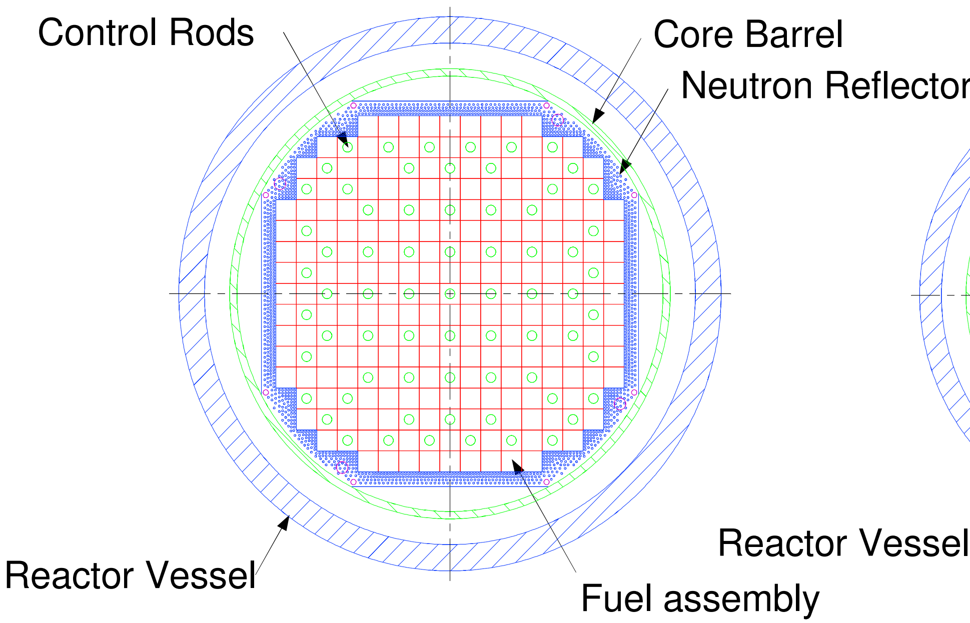
Design Features of APWR core (1/2)

(1) Large Core : consists of 257 fuel assemblies of an advanced 17x17-type

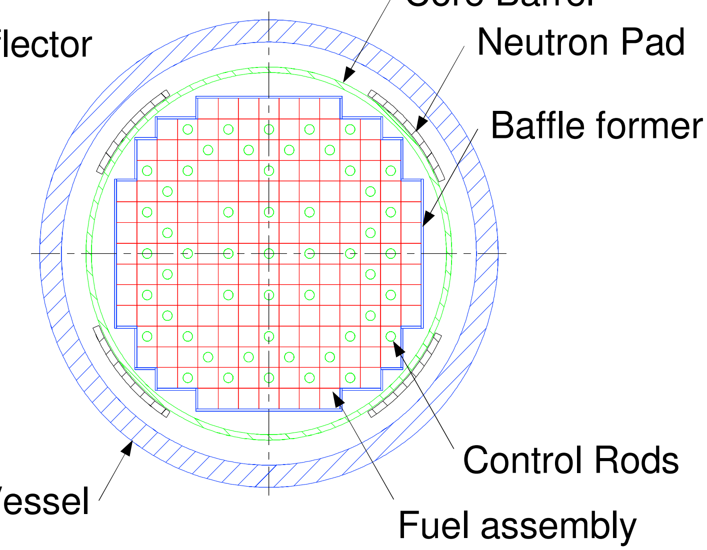
(2) Fuel : has lower gas plenum with the advanced design features as mentioned before

(3) N/R : has radial neutron reflector

Design Features of APWR core (2/2)



APWR

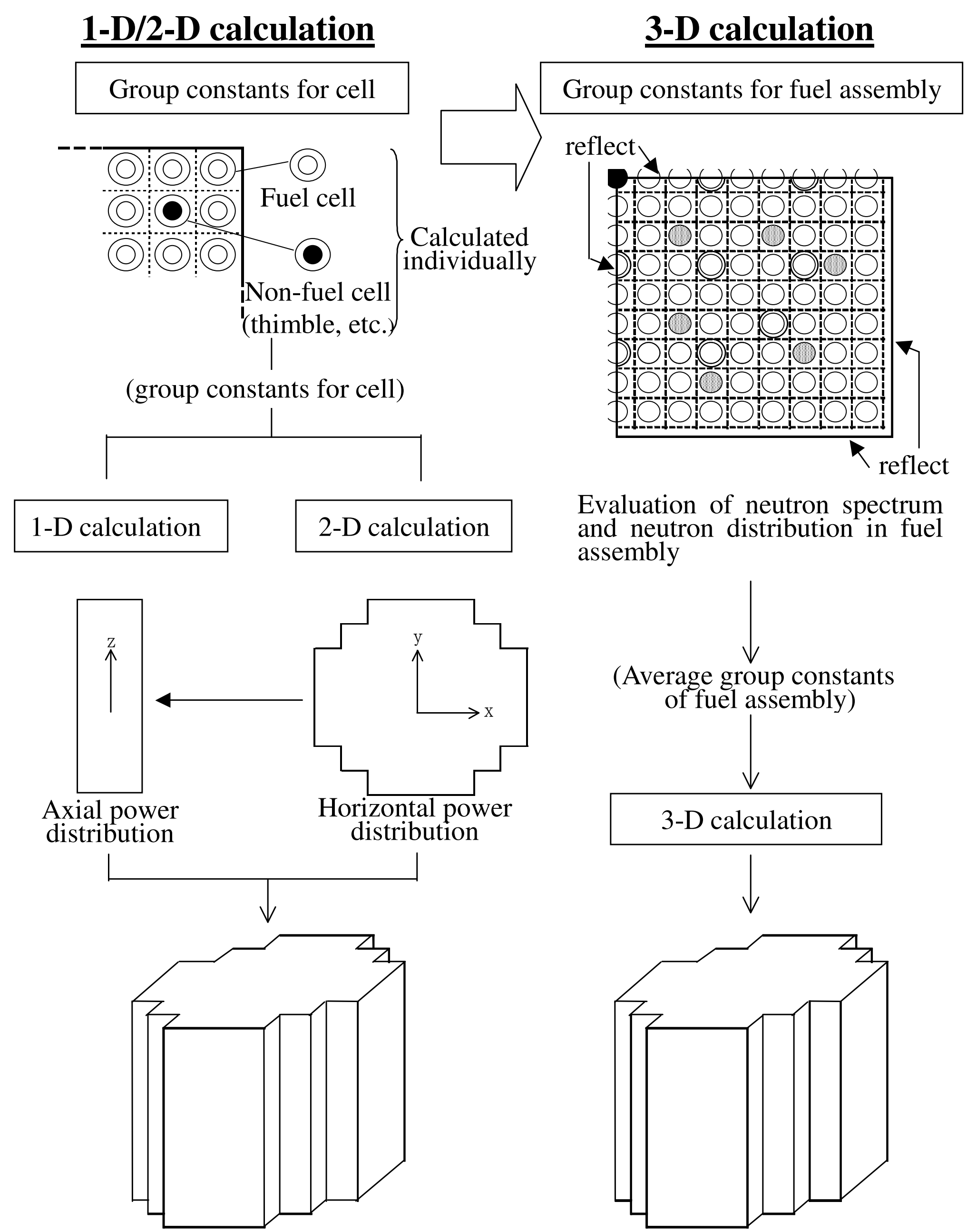


Conventional 4-loop plant

Comparison of Cores

Nuclear Design Methodology(1)

- 1D/2D Code System
  - PANDA/LEOPARD/HIDRA (MHI)
  - NULIF/SHARP-A/ SHARP-XY (NFI)
- 3D Code System
  - PHOENIX-P/ANC (MHI)
  - Improved NULIF system (NFI)
  - CASMO/SIMULATE (NEL)



Nuclear Design Methodology(2)

- **PHOENIX-P (MHI)**
  - Generating 2-group constants to be used in core calculation
  - Treating an single assembly or 4-assembly model with heterogeneity of fuel cells
  - Employing transport theory (response matrix and S_N)
  - Multi-group XS library

Nuclear Design Methodology(3)

- **ANC (MHI)**
  - Predicting PWR core performance parameters
  - Employing Diffusion Nodal Expansion and pin power recovery method
  - Nodal Expansion Method: Polynomial
  - Pin power recovery: Group theory

Nuclear Design Methodology(4)

- **Improved NULIF System (NFI)**
  - Generation of 2-group constants and Prediction of Core Performance
  - Spectrum interference effect by the Multi-assembly model
  - Pin power prediction by 2D pin-by-pin / 3D improved CM synthesis method
  - Improvements on burnup chain in the assembly calculation

Nuclear Design Methodology(5)

- **CASMO/SIMULATE (NEL)**
  - CASMO
    - Heterogeneous 2D transport calculation using the method of characteristics (MOC)
    - Capability of performing full-core heterogeneous transport and depletion calculation
  - SIMULATE
    - Advanced nodal diffusion model
    - MOX core analysis model
      - Semi-analytic nodal model
      - Nodal transport model
      - Spectral interaction model

Improvement of design system for near future : MHI Case (1)

[**Cross section library**]

ENDF/B-V 42-group to **ENDF/B-VI 70-group**
(**JENDL3.3 200-group** is under development)

[**Lattice Code**]

Response Matrix & S_N to **CCCP**
(PARAGON(**CCCP with DAF**) is ready to use.)

[**Core Calculation Code**]

Nodal method to **Semi-analytical Nodal method**
Group theory to **Fourier Transformation**
Xe and Sm feedback to **micro-depletion method**
4-assembly model to **single assembly with SPC**

Improvement of design system for near future : MHI Case (2)

- PARAGON: New Lattice Physics Code developed by MHI and Westinghouse
  - Current Coupling Collision Probability with Discrete Angular Flux
  - Function to generate power profile within fuel rod by Spatially Dependent Dancoff Method (SDDM)
  - Full FORTRAN90 features.
    - Dynamic memory allocation etc..
    - Any number of space mesh and energy groups can be handled.

Improvement of design system for near future : MHI Case (3)

- PARAGON basic theory

Current **C**oupling **C**ollision **P**robability with **D**iscrete **A**ngular **F**lux equation

$$V_i \phi_i = \sum_{\alpha} \sum_{n,m} J_{-,\alpha}^{nm} P_{i,s_\alpha}^{nm} + \sum_j V_j F_j p_{ij}$$

$$J_{+,\alpha}^{nm} = \sum_{\beta} \sum_{k,\ell} J_{-,\alpha}^{k,\ell} P_{s_\alpha,s_\beta}^{nm,k\ell} + \sum_j V_j F_j p_{s_\alpha,i}^{nm}$$

$$J_{+,\alpha}^{nm} = \beta_{\alpha}^{nmk\ell} J_{-,\alpha}^{k\ell}$$

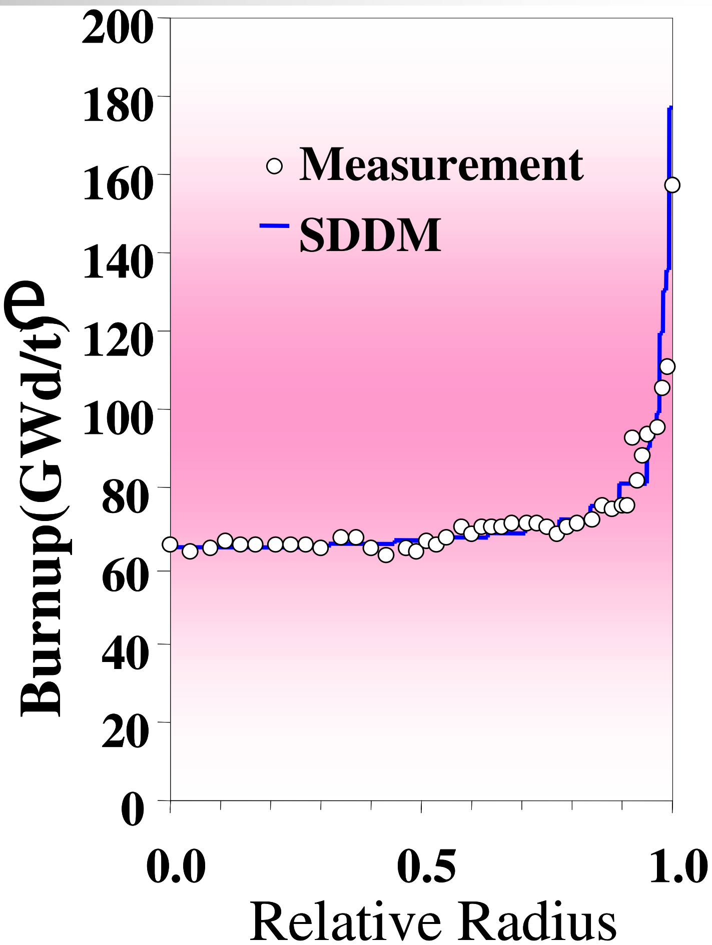
Improvement of design system for near future : MHI Case (4)

- **Resonance XS**

  for Lattice code

Spatially Dependent Dancoff Method(SDDM) was developed. (2002)

The method has been put in PARAGON



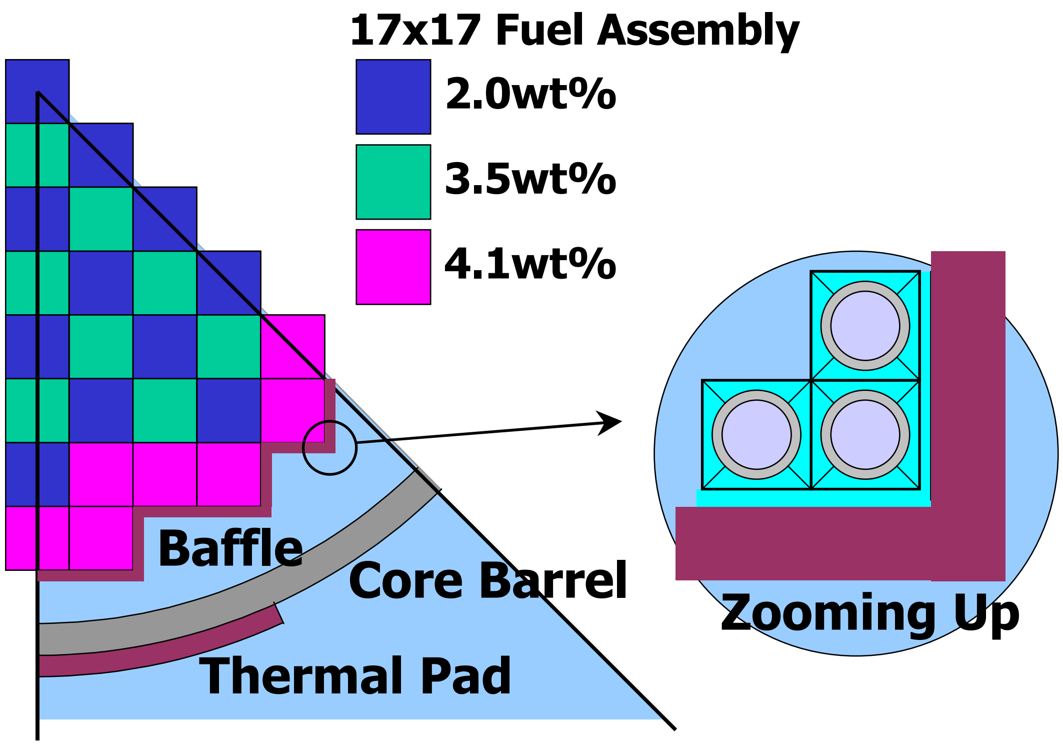
Measurement: CRIEPI

Burnup distribution within Fuel Rod(Average 74.5GWd/t)

Improvement of design system for near future : MHI Case (5)

- Large Scale Calculation

K-eff
PARAGON: 0.9980
MCNP :0.9991(1σ0.02%)



A Japanese typical 3loop initial Core

-2.2				
-2.1	-1.2			
-1.3	-0.5	-0.4		
-1.2	-0.8	-0.7	-0.9	
-1.7	0.0	-0.7	2.1	2.2
-1.8	-1.7	1.4	3.0	3.5
-1.5	0.3	0.8	1.8	
-1.1	-1.2			

Assembly Power Comp.
(PARAGON-MCNP)/
MCNP x 100(%)

Improvement of design system for near future : MHI Case (6)

Core Calculation code for near future Micro-Depletion Correction Model

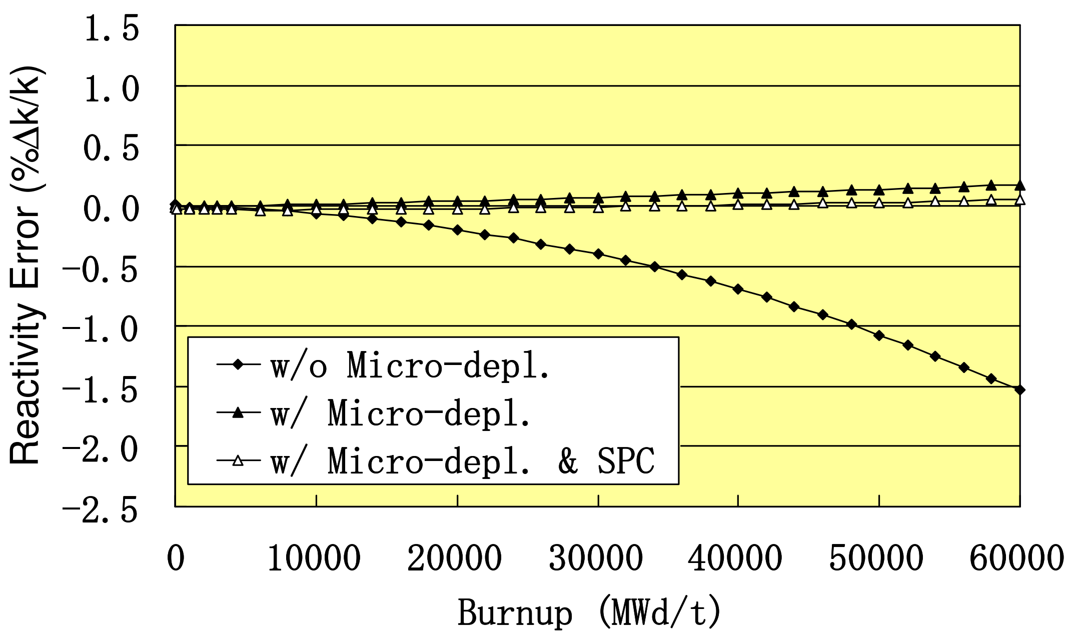
- **Explicit Tracking of Number Density**
  - Based on Two-Group Flux and Micro XS
  - Predictor-Corrector Method for Depletion (except for Gadolinia chain)
- **Correction to Macro XS**
  - Determine the Difference of Contribution to the Macro XS

$$\Delta \Sigma_{ag} = \sum_{i} \left(\sigma_{ag,i}^{actual} N_{i,actual} - \sigma_{ag,i}^{ref} N_{i,ref} \right)$$

  - Applied to Absorption and Fission XS

Improvement of design system for near future : MHI Case (7)

- Reactivity Error with Spectral Correction (Relative Power=1.0, Boron=1000ppm)



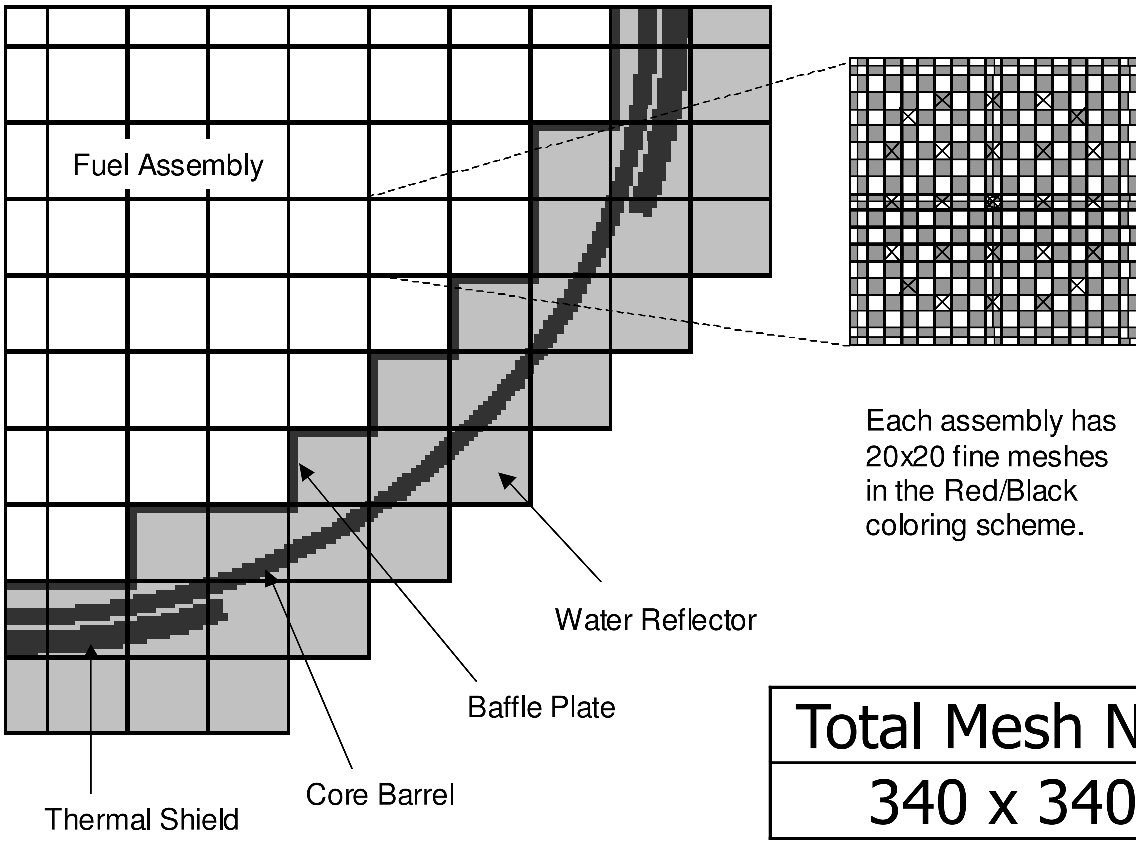
Two-group constants were generated with PHOENIX-P where 1.0 power and **500ppm** boron were assumed.

Improvement of design system for near future : NFI Case (1)

- SCOPE2: PWR Core Calculation Code for the Next Generation Developed by NFI
  - One of the most precise method ever
    - 3-D Pin-by-Pin geometry
    - Multi-group (currently 9 groups)
    - Nodal SP3 transport theory method
  - Fully featured for reload core design
    - Reloading, grid treatment, RCC cusping, etc.
  - Realistic computing time for depletion calc.
    - ~2.5 hours by 24 processors (P-4 2GHz)

Improvement of design system for near future : NFI Case (2)

- 3-Dimensional Pin-by-Pin Geometry



Pin-cell averaged XS

Correction Factor
By
Improved-SPH method

Total Mesh Numbers	Total Memory
340 x 340 x 26	Approx. 5GB

Improvement of design system for near future : NFI Case (3)

3-Loop Type PWR HZP Power Distribution by SCOPE2

Initial Core

	H	G	F	E	D	C	B	A
8	0.3	0.1	-0.2	-0.6	-0.8	-0.4	0.2	0.7
9	-0.1	-0.3	-0.4	-0.5	-0.7	-0.3	0.0	0.7
10	-0.4	-0.5	0.0	-0.1	0.0	0.0	0.4	
11	-1.0	-0.9	-0.3	0.1	0.0	0.3	0.6	
12	-1.0	-0.9	-0.1	0.3	0.4	0.6		
13	-0.6	-0.6	0.1	0.4	0.6			
14	0.3	0.1	0.7	0.5				
15	1.1	1.1						

-- Error[%]

RMS=0.5%
MAX=1.1%

Reload Core

	H	G	F	E	D	C	B	A
8	0.2	0.6	0.0	-0.1	0.1	0.7	0.6	0.3
9	0.8	1.0	0.7	0.5	0.3	0.7	0.6	0.1
10	0.3	0.6	-0.6	-1.3	-1.6	0.2	0.6	
11	0.9	1.0	-0.9	-1.5	-1.3	0.4	0.4	
12	0.8	1.0	-1.5	-1.7	-1.6	-0.3		
13	1.2	1.1	0.1	0.6	-0.1			
14	0.1	0.1	-0.8	0.2				
15	-0.6	-0.7						

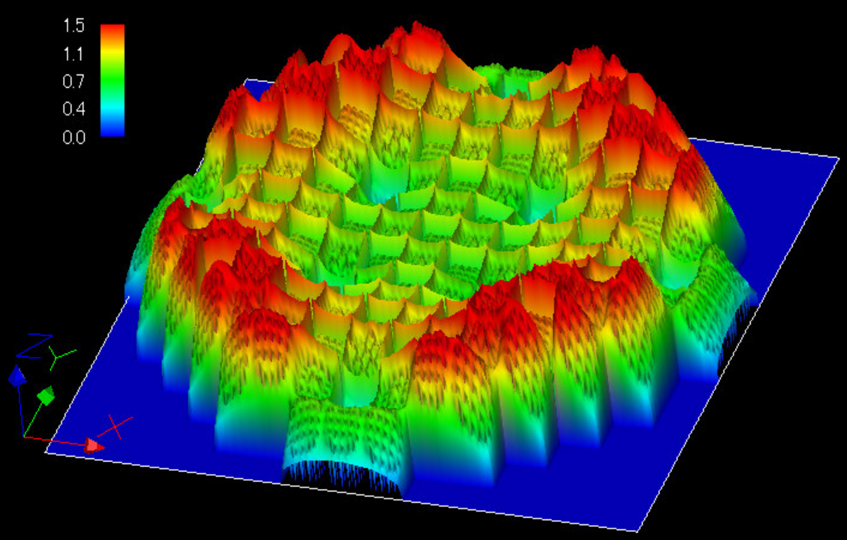
-- Error[%]

RMS= 0.9%
MAX=-1.7%

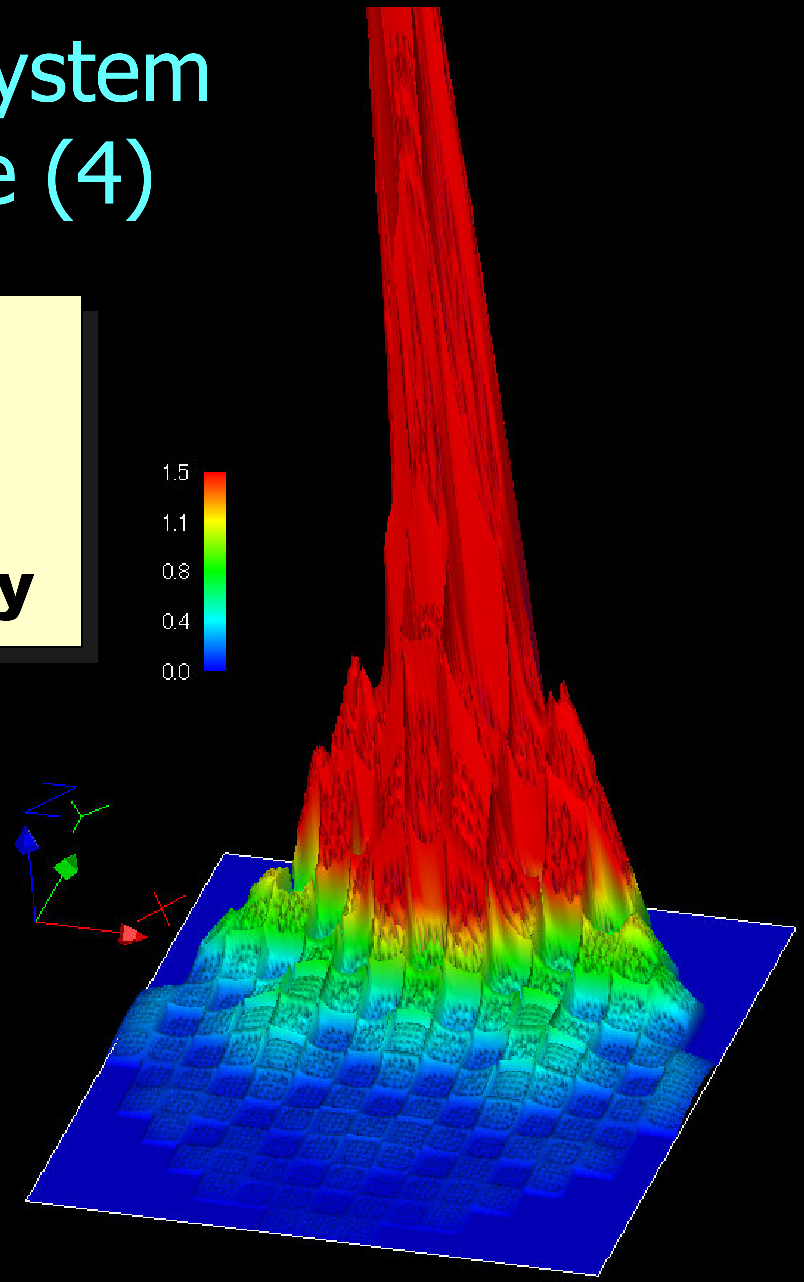
＊Error=(Meas.-SCOPE2)/SCOPE2×100(%)

Improvement of design system for near future : NFI Case (4)

3-Loop Type Initial Core
HZP Power Distribution
by SCOPE2
in 3-D Pin-by-Pin Geometry



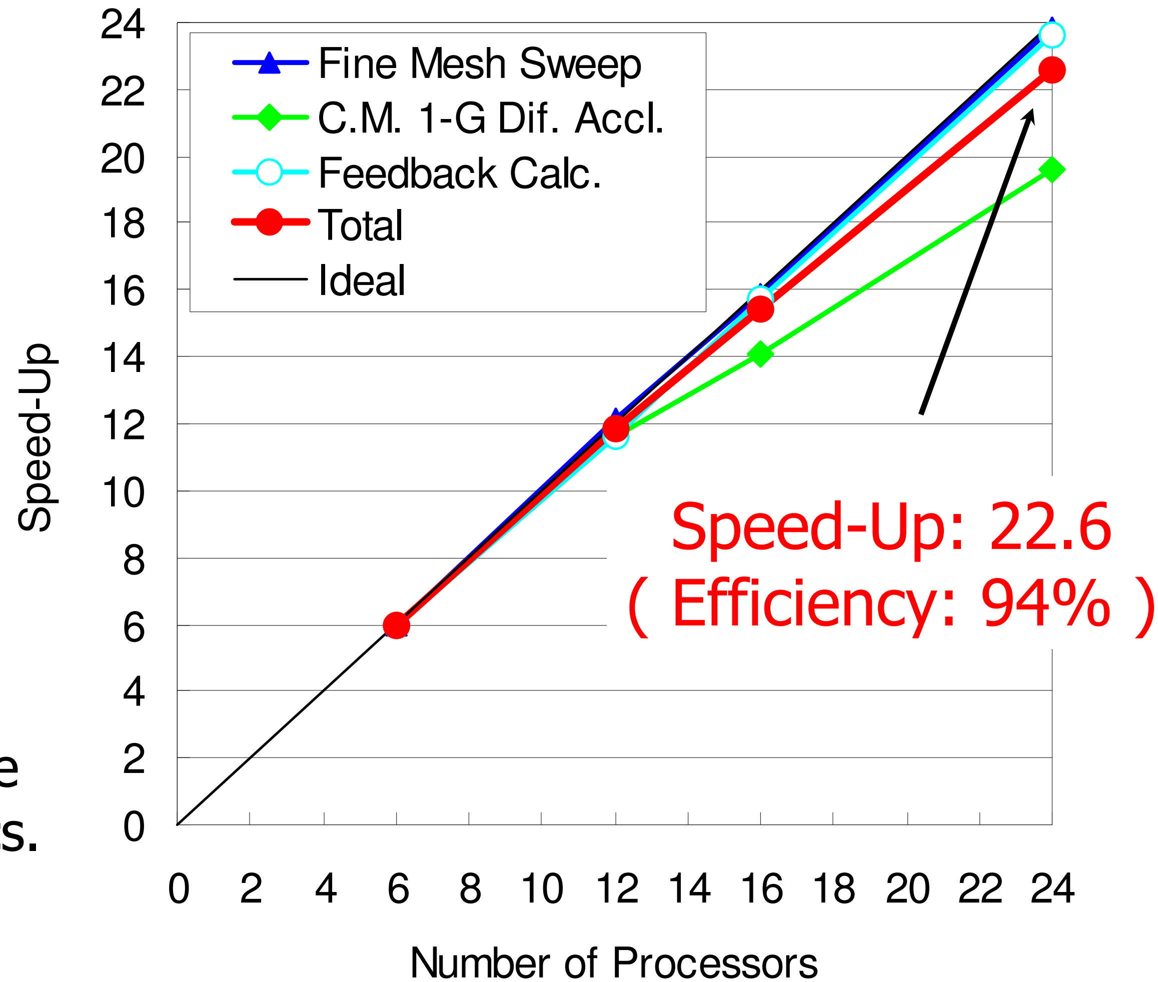
RCC D-Bank In



All-Rods-In with 1-Rod Stuck

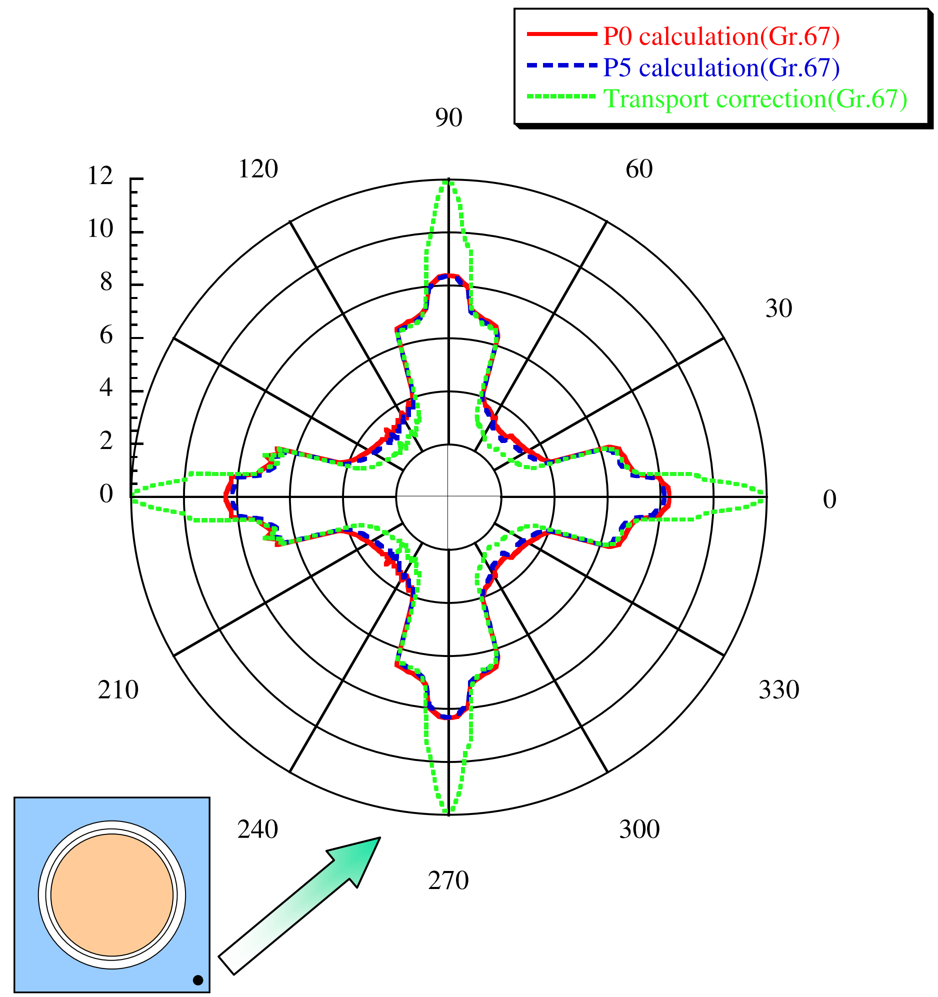
Improvement of design system for near future : NFI Case (5)

- Identical Results in Parallel Computing
  - Important in view of quality assurance for design calculations
  - Fine-grained Parallel Algorithm
- Good Scalability
  - Enough for middle-size cluster with ~ 24 procs.



Improvement of design system for near future : NEL Case (1)

- **NELAT: Rigorous Lattice Physics Code for the Next Generation Developed by NEL**
  - Heterogeneous 2D Transport Calculation based on the Characteristics Method
  - Efficient Ray Tracing using the Macroband Method
  - Efficient Angular Integration Scheme using the Polynomial Expansion for Neutron Flux
  - Capability of Treating the Anisotropic Scattering Source Up to P_5 Order



Angular Flux Distribution (12wt%Pu^{tot} MOX Fuel Cell)

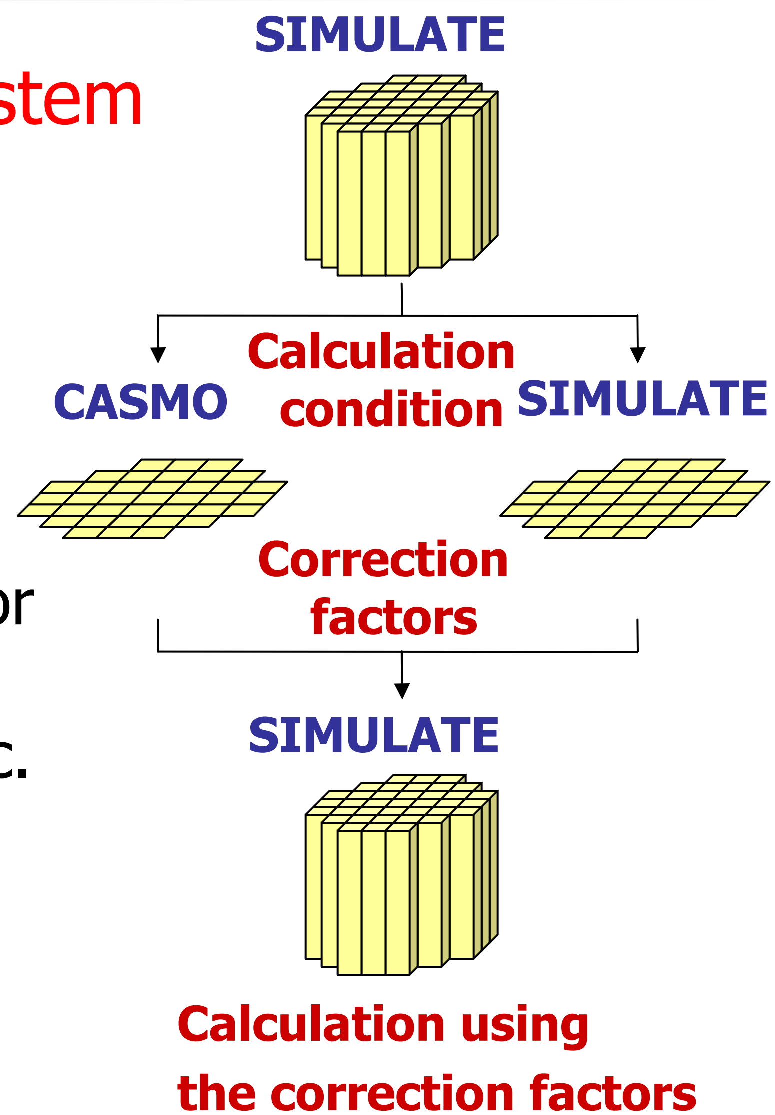
Improvement of design system for near future : NEL Case (2)

Hybrid Core Calculation System

Compare the CASMO and SIMULATE *2-D* core calculation results under the predetermined *consistent* calculation condition

Generate the *correction factors* for assembly homogenized cross sections, discontinuity factors, etc.

Apply them for SIMULATE 3-D calculation



4. FR Analysis Codes

- Cell Codes

  SLAROM,

  CASUP
- Core Calculation Codes
  - CITATION
  - TRITAC(XYZ)

    DSA

    → JUPITER
  - NSHEX

    Nodal Transport in Hex.-z

- Detailed Core Calculation

  BACH---MOC+Axial Leakage 3D (Hex-z)
- Sensitivity Codes

  SAGEP---2D, 3D

  ---GPT(Generalized Perturbation Theory Based on Diffusion Theory)

  SAINT---1D, 2D

  Sensitivity in Cells and Assemblies, GPT Based on FFCP

  ABLE

  Cross Section Adjustment with Method Uncertainty

Calculation Flow of BACH

CASUP

Calculate effective macro cross section

MOC code for hexagonal geometry

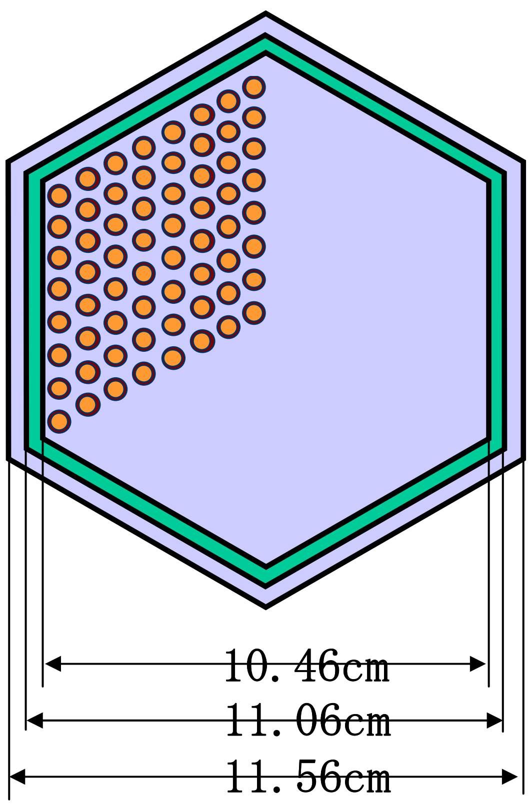
- Calculate intersection between Pass line and geometry and length of the segment
- Correct the width of Pass line so that each area is kept constant
- Calculate the angular flux along Pass line
- Calculate the scalar flux

Make homogenized macro cross section

HEX-Z nodal transport code

Calculate axial leakage and power distribution

Infinite assembly problem



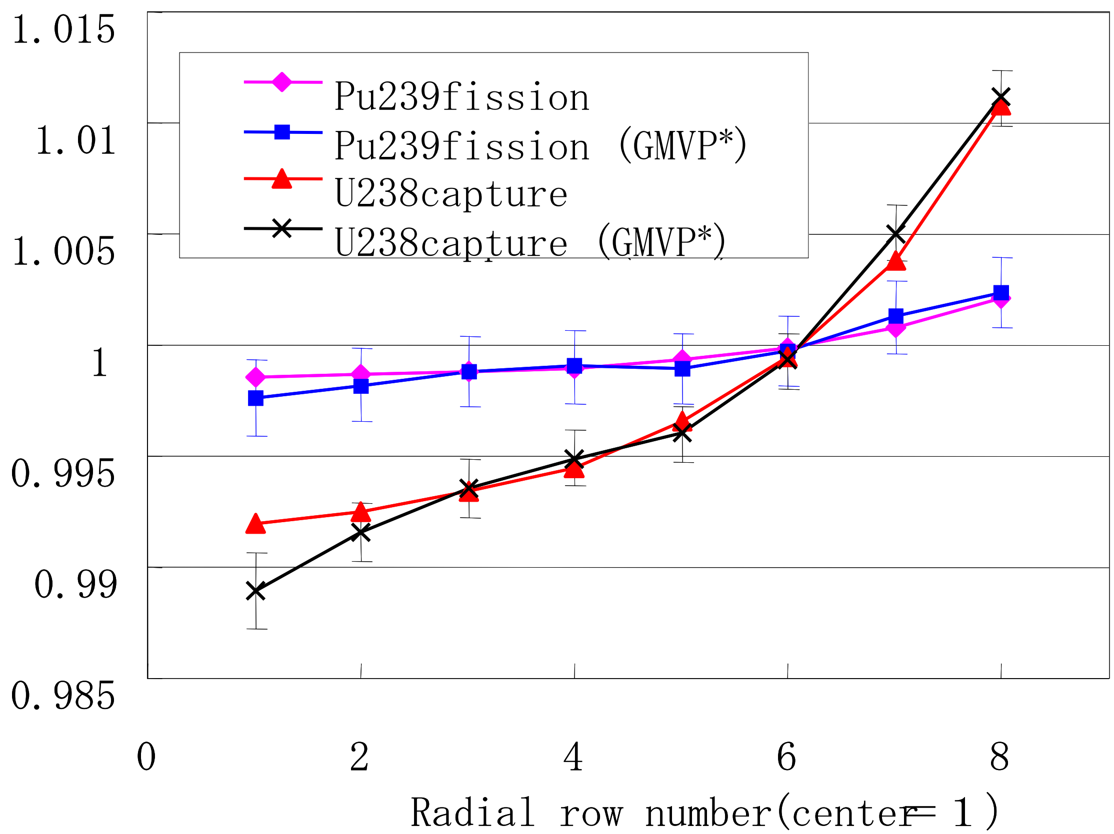
Calculation Model

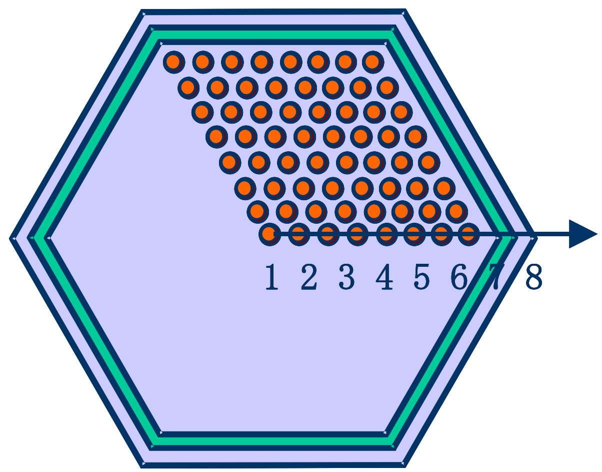
Fuel Rod	
Number	169
Diameter of Fuel	0.54 cm
Diameter of Cladding	0.65 cm
Pin Pitch	0.787 cm

Material

Fuel	MOX
Cladding & Wrapper Tube	Stainless
Coolant	Na

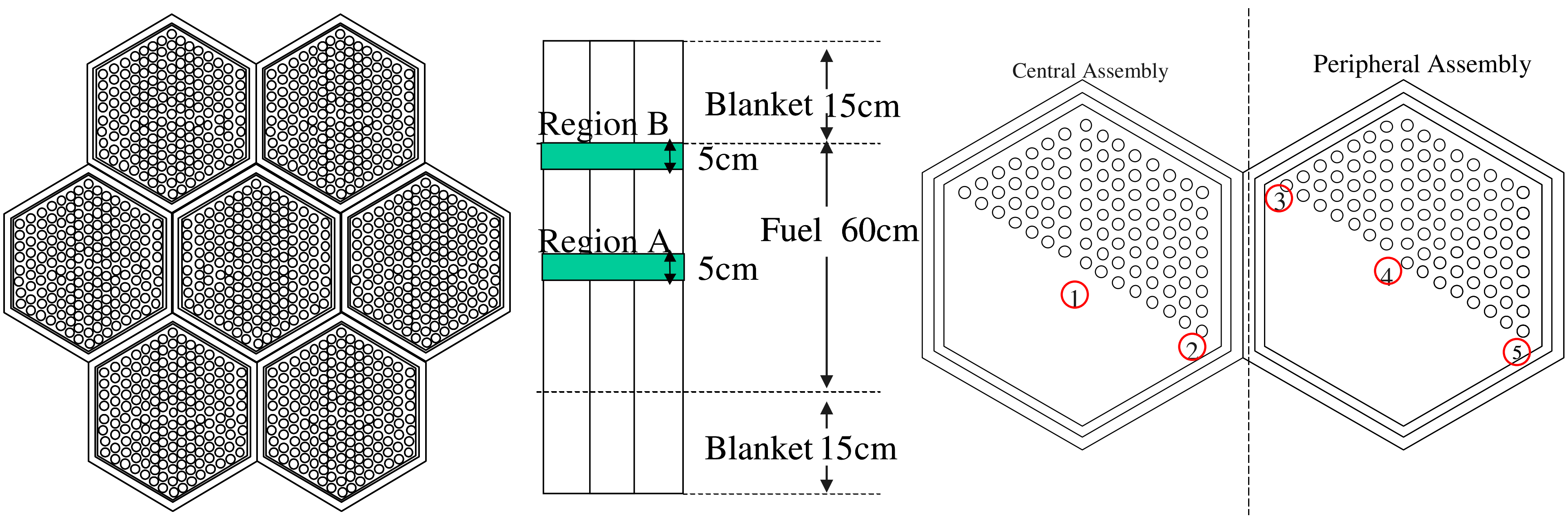
Reaction rate distributions





* GMVP: Monte-Carlo code

3D Calculation



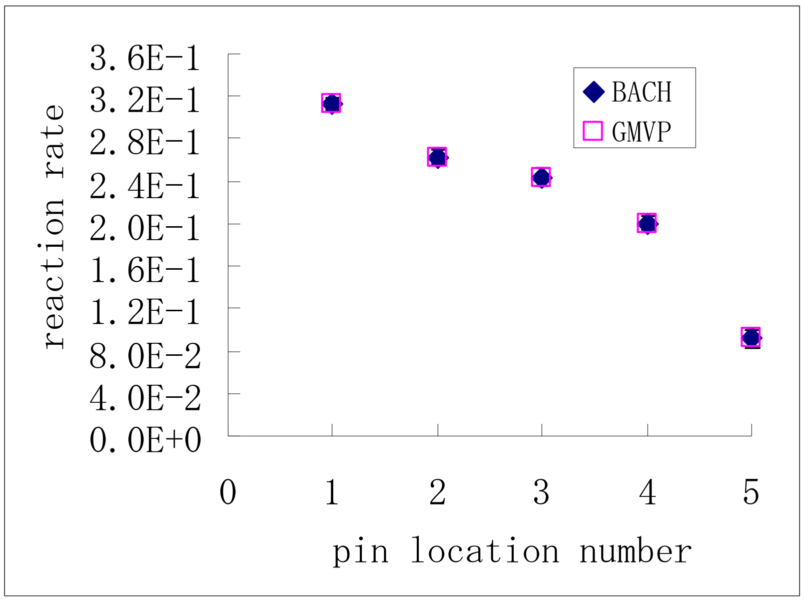
Boundary:vacuum

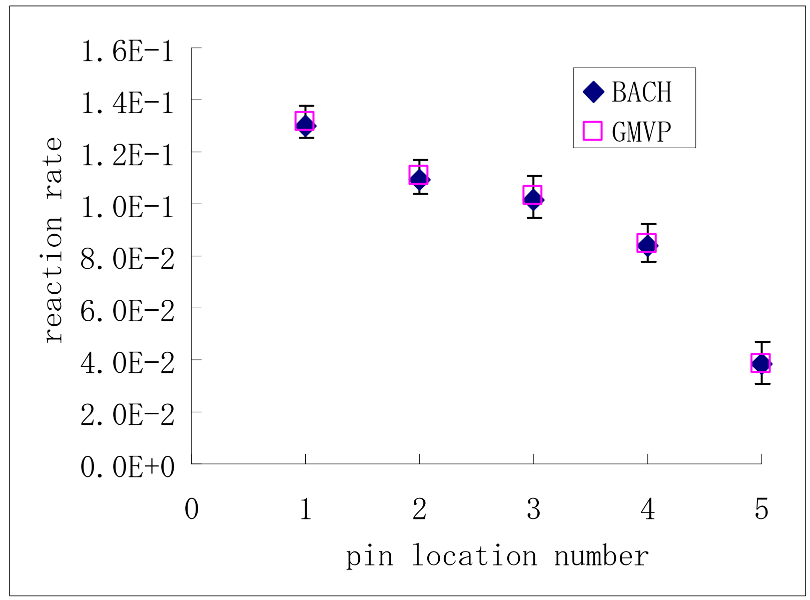
Calculation Model

Reaction rate in individual pins

Np-237 Capture

GMVP: 5000*10000 history





Region A

Region B

JNC Activities

- Development of SLAROM-UF
- Covariance Processing Code : ERRORJ
- Reactor Constant Adjustment
- Programme of PIE Analysis at JNC
- NGSC (Next Generation Simulation Code) in JNC

Development of SLAROM-UF

- To improve the treatment of the resonance self-shielding in a heterogeneous cell

- To improve the treatment of the resonance self-shielding, and interaction between <u>different nuclides</u> , <u>regions</u>, and <u>temperatures</u>

- To apply various fast reactor core analyses for <u>the feasibility study</u> in JNC

(Coolant/Fuel) Na/MOX, Pb-Bi/MN, Gas/MN reactor, etc.

Developed new cell calculation code “SLAROM-UF”

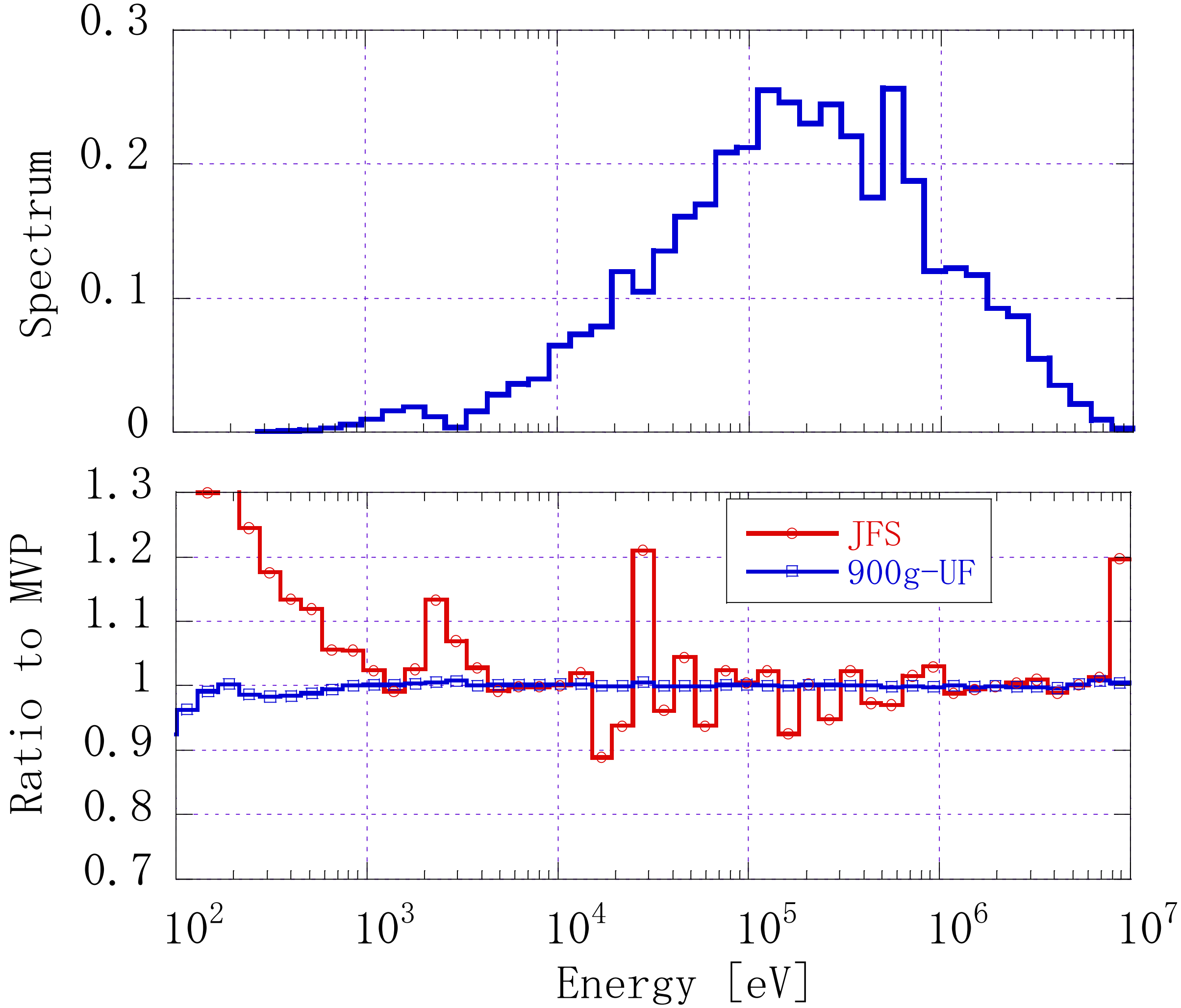
Combination use of

- Slowing down equation in ultra fine (100,000) group structure for < 50keV
- Pij calculation in 900-group structure

  Δu=0.008 for > 50keV (comparable to the ECCO 1,968-group library)

  Reflect ultra fine calculation result for < 50keV

Application of SLAROM-UF



JOYO sub-assembly
Infinite cell spectrum

Ratio to MVP

JFS : 70 group constant with f-table
900g-UF : new result

Neutron spectrum by SLAROM-UF coincides with that by continuous energy Monte Carlo method.

Covariance Processing Code:ERRORJ

ERRORJ

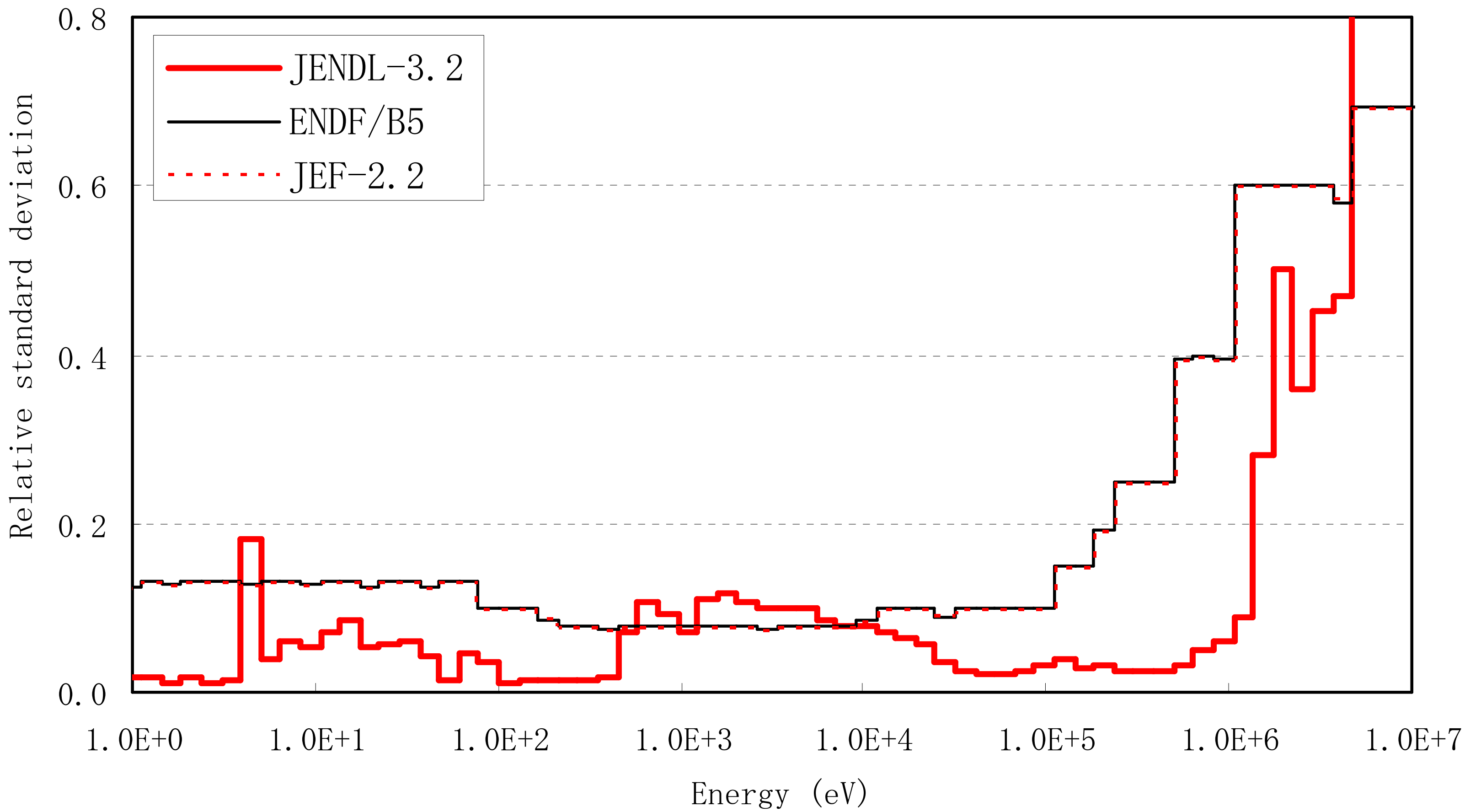
Covariance data given in nuclear data file $\longrightarrow$ Group-averaged covariance data for application

- Expanded processing for covariance of resonance parameters

  Sensitivity coefficient of group constant to resonance parameter is calculated numerically.

- Processing for μ **and fission spectrum**

Comparison of Covariance:U-235(n, γ)



Uncertainty in criticality of BFS-62-1 core (1 σ)

340 pcm (JENDL-3.2) **800 pcm** (ENDF/B-V)

Reactor Constant Adjustment

■**ADJ2000R** has been used in the feasibility study

- Adjust JFS 70g constant based on **Bayesian theory**
- Use various experimental data of small to large Na-MOX core including **burn-up** and **temperature** related reactivity
- Use covariance data consistent with JENDL-3.2

↓

Apply the adjustment technique to uncertainty evaluation of BN-600 hybrid core analysis with BFS-62 experiment

Confirm effectiveness of the technique

(1) No-Information method (No experimental data is considered.)

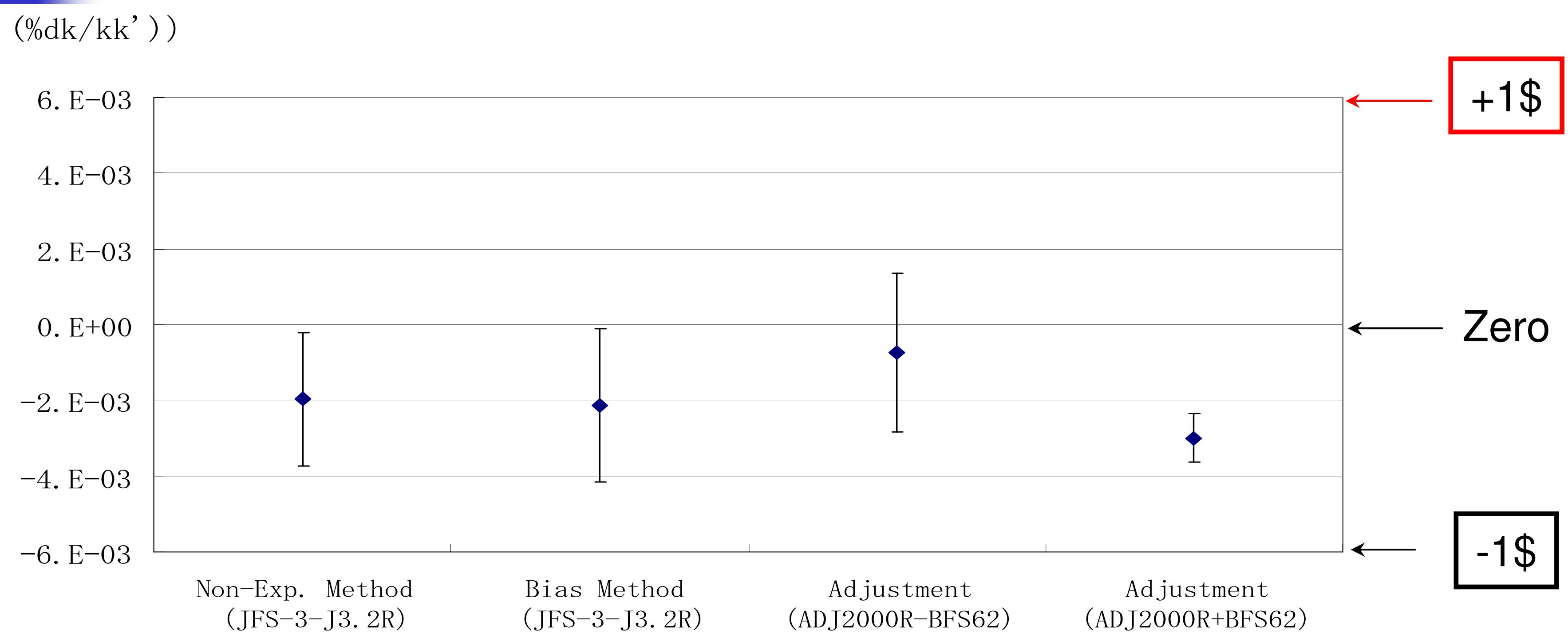
(2) E/C Bias Method = "Bias factor method"

(3) Cross-Section Adjustment Method

Not considering BFS-62 data = "ADJ2000R - BFS62"

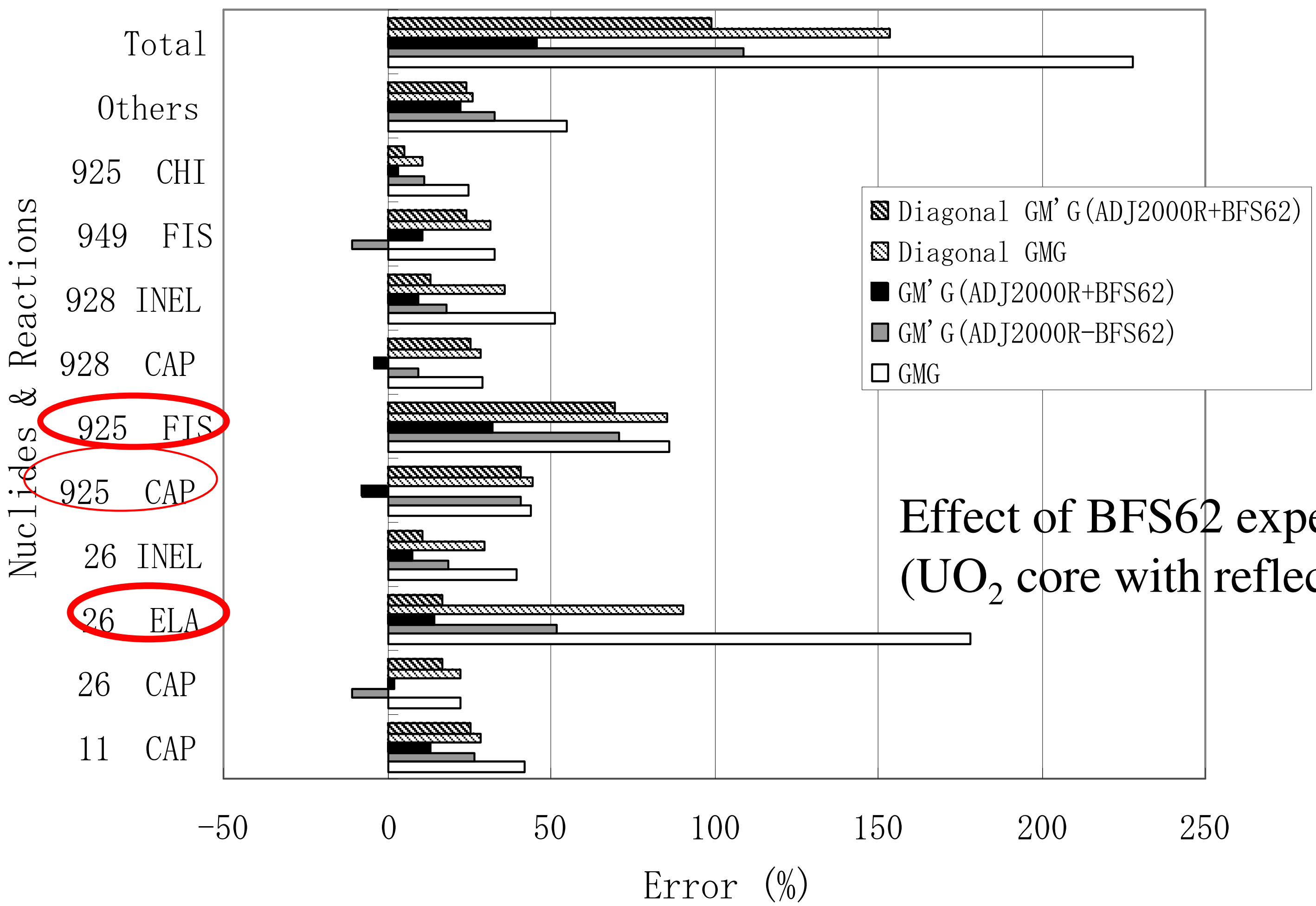
Considering BFS-62 data = "ADJ2000R + BFS62"

Predicted Uncertainty on Sodium Void Reactiv



Uncertainty (¢)	**30**	**34**	**36**	**11**
X-section-induced error (¢)	27	31	25	5

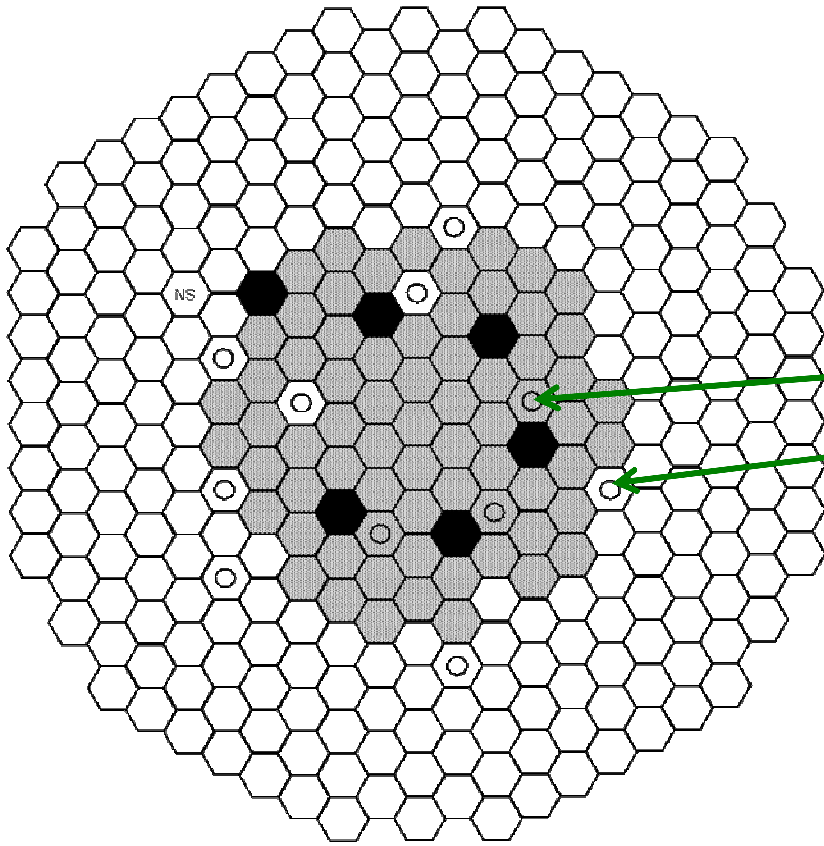
Breakdown of the Predicted Uncertainty



Effect of BFS62 experiment (UO_2 core with reflector)

Programme of PIE Analysis at JNC

- JOYO MK-I core fuel
- JOYO MK-II driver fuel
- MA samples irradiated at JOYO MK-II core



MA sample irradiation

^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm

(Fourteen samples are in a PIE stage.)

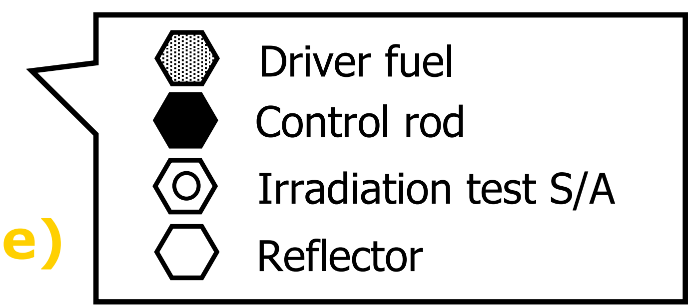
Irradiation positions

3rd row, 208 efpd (29th-32nd cycle)

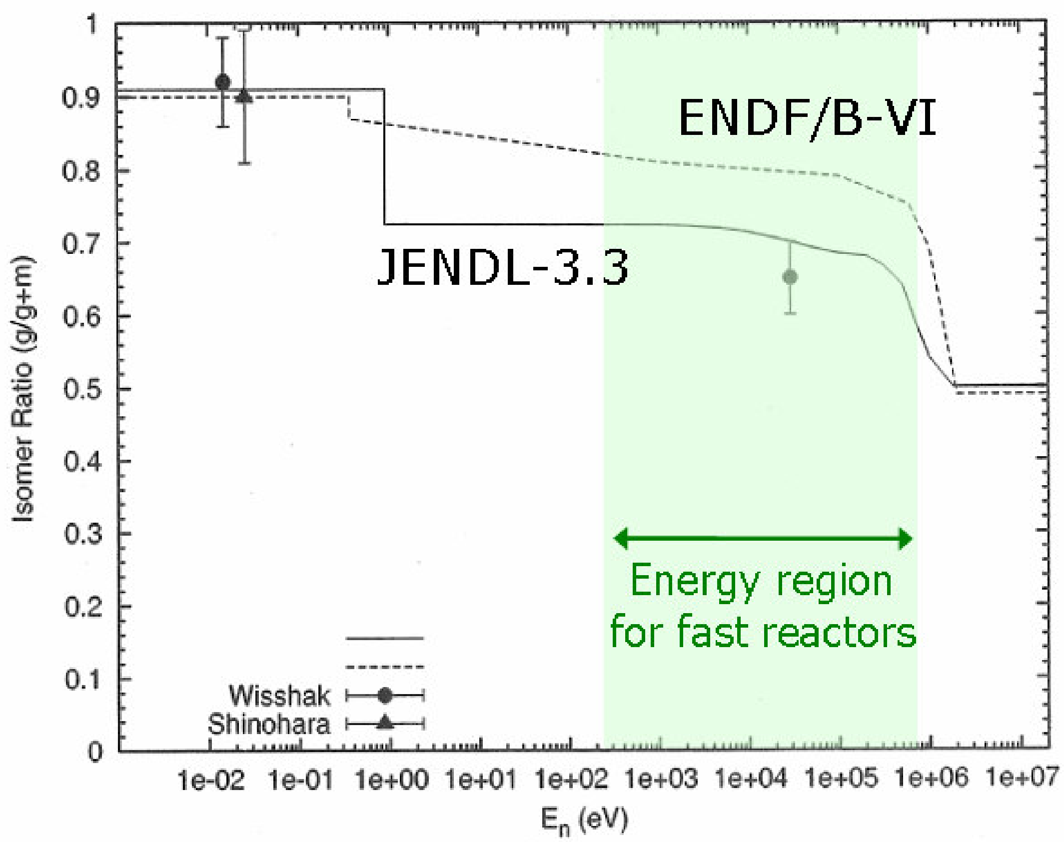
5th row, 251 efpd (30th-33rd cycle)

Neutron flux: 1～3 × 10^{15} n/cm^2 sec

JOYO MK-II core (32nd cycle)



Preliminary C/E Values on ^{242m}Am / ^{241}Am Abundance Ratio



Present status of the isomeric ratio for ^{241}Am capture reaction

Nuclear Data Library	^{241}Am Isomer Ratio	C/E on ^{242m}Am/ ^{241}Am
JENDL-3.2	0.80	1.30
JENDL-3.2	0.85	1.00
JENDL-3.3	0.85	1.02
ENDF/B-VI	0.85	0.94
JEF-2.2	0.85	1.03

The above results implied the possibility that the ^{241}Am isomeric ratio lies around 0.85.

NGSC (Next Generation Simulation Code) in JN

Background

- In 2002, feasibility study on computer technologies in support of NGSC concept was started inside JNC.
- In 2003, an expert group was set up in each of CEA and JNC to discuss jointly the development direction of future simulation codes and to put together a common report.

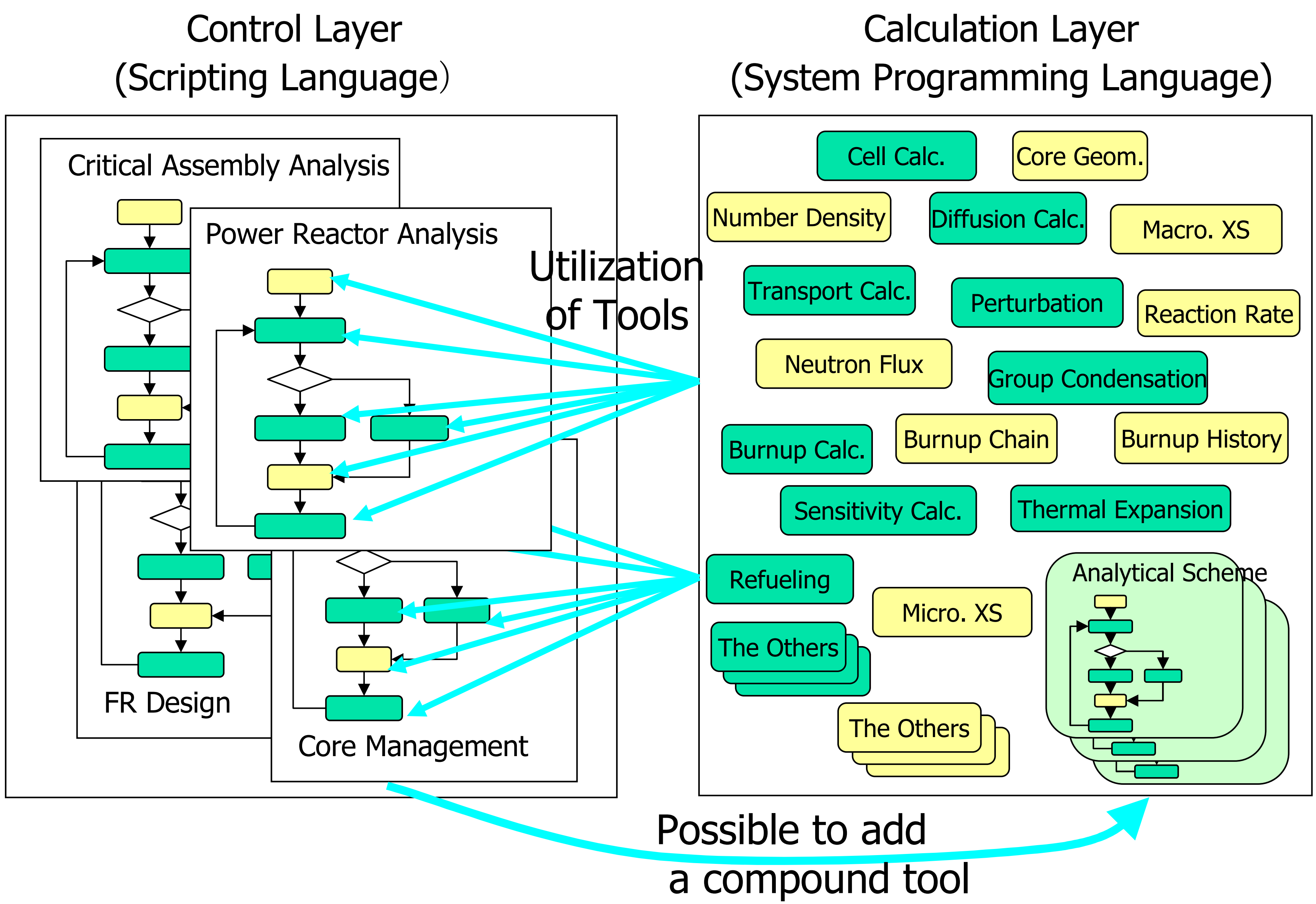
Needs

- Development of advanced physics models
- Improvement of numerical solution methods and procedures
- Simulation methods for multi-physics phenomena
- Code validation by experiments

Main points of concept

- Existing software properties or legacy codes are to be efficiently utilized (efficiency)
- De facto standard (universality) and accessible by engineers and researchers in nuclear societies (openness)
- Flexible and expandable (expansibility)
- Validation and qualification of the system (completeness)

Architecture of NGSC for Reactor Physics



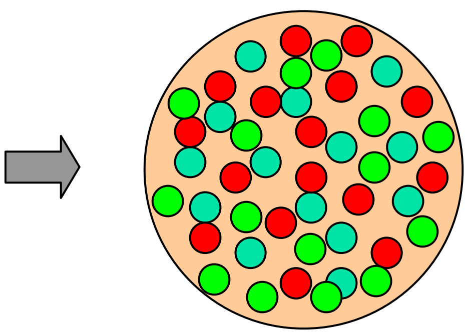
5. General Analysis Codes

MVP:general purpose Monte Carlo code based on the continuous-energy model

- **Time-dependent neutron and photon transport problems**
- **External-source and fission-source (keff) problems**

Features

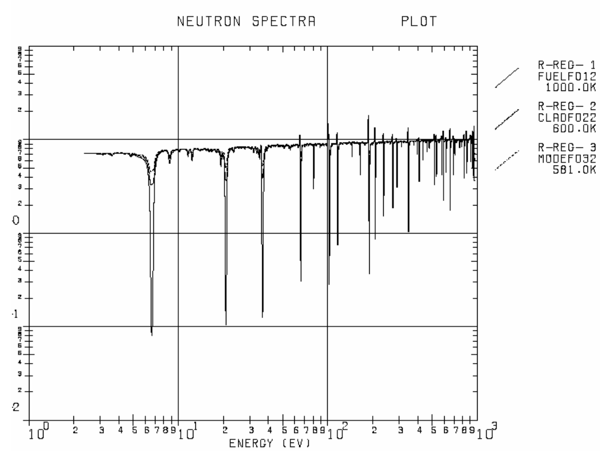
- **Fast computation algorism for vector and/or parallel computers**
- **Arbitrary temperature calculation**
  (Internal production of temperature dependent libraries)
- **Burn-up calculation**
- **Statistical geometry model**
  for randomly distributed lots of particles
  (analyses of HTGR, plutonium spots etc.)
- **Accurate perturbation calculation**
- **Noise analysis (Feynman-α)**



Location and type of particles are sampled along a flight path of each neutron

SRAC: neutronics calculation code system for various types of thermal reactors

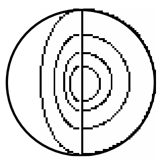
- **Production of effective microscopic and macroscopic group cross sections**
- **Static cell and core calculations including burn-up analyses**



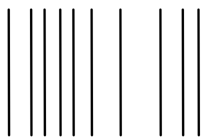
Features

- **Collision probability calculation (PIJ) applicable to 16 types of lattice geometries**
- **PEACO option which solves a multi-region cell problem by PIJ using hyper-fine lethargy mesh in resonance energy range**
- **S_N transport codes ANISN(1D), TWOTRAN(2D) and multi-dimensional diffusion code CITATION are integrated into the system to enable many choices of calculation flow depending on problem**

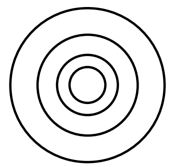
Geometrical models of PIJ



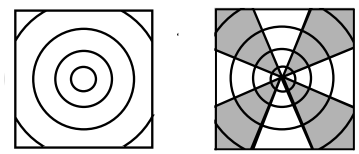
Sphere
(Pebble, HTGR)



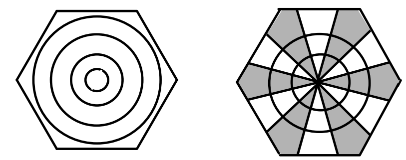
1D-Plate
(JRR, JMTR)



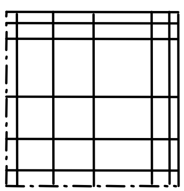
1D-Cylinder
(any pin type fuel)



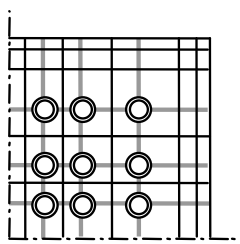
Square unit pin cell
(PWR, BWR)



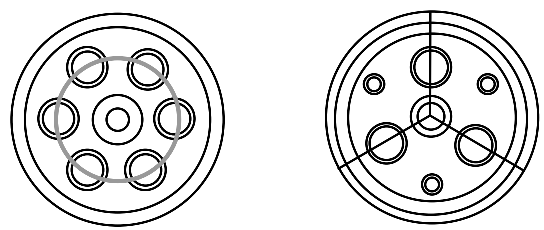
Hexagonal unit pin cell
(FBR, VVER, HCLWR)



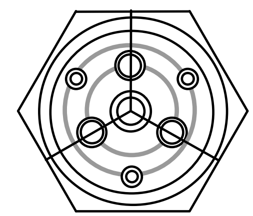
2D square plate fuel assembly (KUCA)



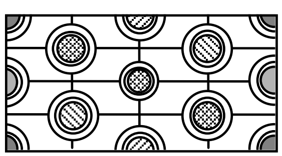
2D square assembly with pin rods (PWR)



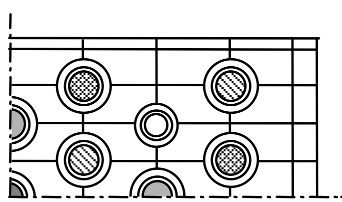
Annular assembly with annular arrays of pin rods
(CANDU, ATR, RBMK)



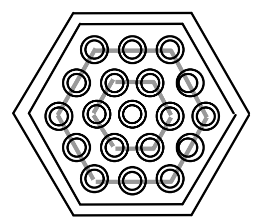
Hexagonal assembly with annular arrays of pin rods (HTTR, VHTRC)



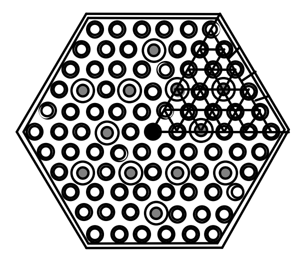
Periodic 2D X-Y array with different pin rods (PROTEUS-LWHCR)



Reflective 2D X-Y array with different pin rods(PWR, BWR,etc.)



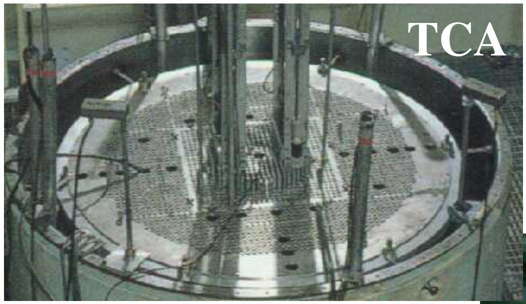
Hexagonal fuel assembly with pin rods (FBR)



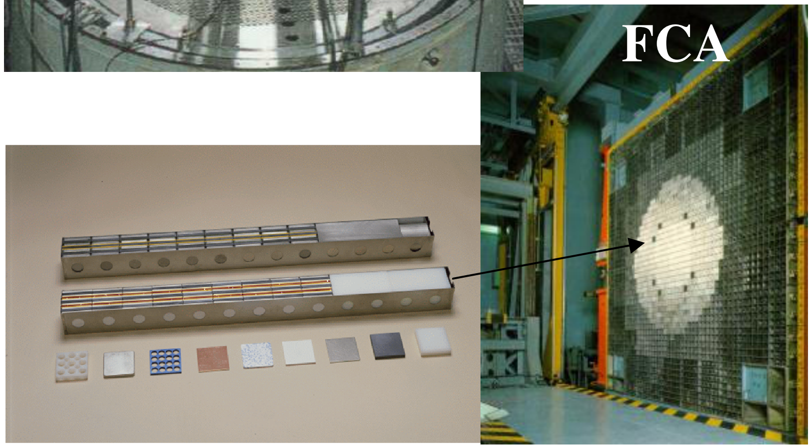
Hexagonal Assembly with different types of pin rods (VVER,HCLWR)

Applications of MVP and SRAC in Japan

Experimental Analyses of Critical Assemblies (CA) and Testing Reactors



***TCA [JAERI]: Tank type critical assembly**
Pin type fuels (low enriched UO_2/MOX),
H_2O moderator

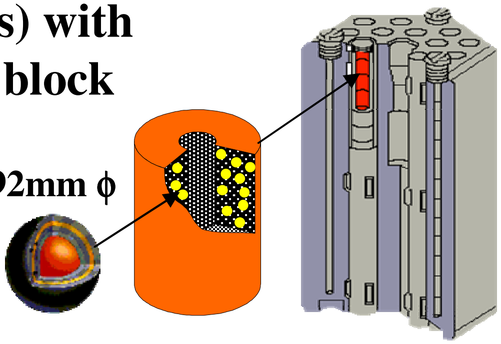


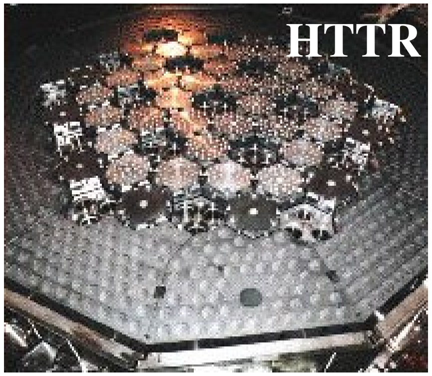
***FCA [JAERI]: Fast Critical Assembly**
Many kinds of plate type fuels
and structural materials;
uranium, plutonium, sodium,
stainless steel, polyethylene, etc.

***HTTR [JAERI]: High Temperature engineering Test Reactor**
TRISO coated fuel particles (CFPs) with
UO_2 kernel in hexagonal graphite block
fuel assembly

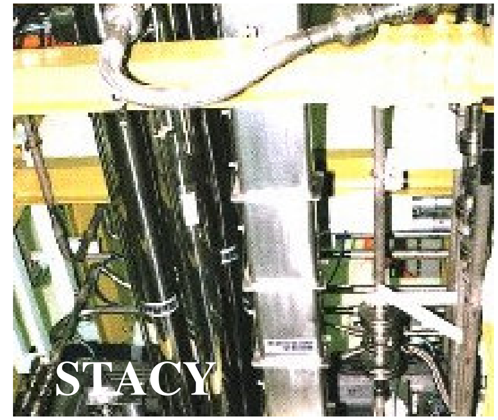
0.92mm ϕ

about 1 billion CFPs
in HTTR



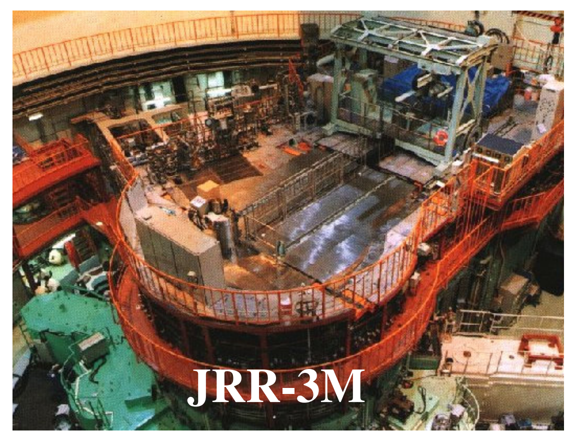


*JMTRC [JAERI] :CA for JAERI Material Testing Reactor
UAl$_x$-Al plate type fuel,
H_2O moderator
*STACY [JAERI]: CA of Nucl. Fuel Cycle Safety
Engineering Research Facility (NUCEF)
10% enriched uranil nitrate solution fuel
*KUCA [Kyoto Univ.]
High enriched U-Al alloy plate type fuel,
polyethylene moderator
*EOLE [Cadarash]: Programs by CEA(France) and NUPEC, JNES
MOX-LWR mockup experiments ; MISTRAL, BASALA and FUBILA



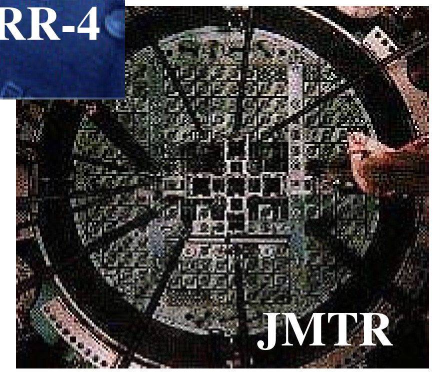
Core Management and Upgrading of Research Reactor

*JRR-2 [JAERI]: research reactor (decommissioned)
45% enriched UAl$_x$-Al cylindrical plate type fuel,
D_2O moderator
*JRR-3M [JAERI]: research reactor
20% enriched UAl$_x$-Al plate type fuel,
H_2O moderator



*JRR-4 [JAERI]: research reactor
93% enriched U, U-Al alloy fuel, (~1996)
20% enriched U, U_3Si_2-Al dispersed alloy fuel (~1998), H_2O moderator

*JMTR [JAERI]: materials testing reactor
20% enriched U_3Si_2-Al dispersed alloy fuel, H_2O moderator

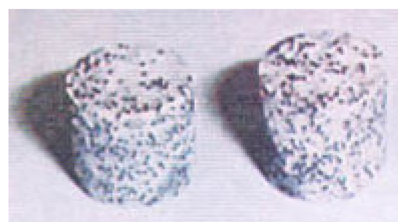


Analyses of Post Irradiation Experiments

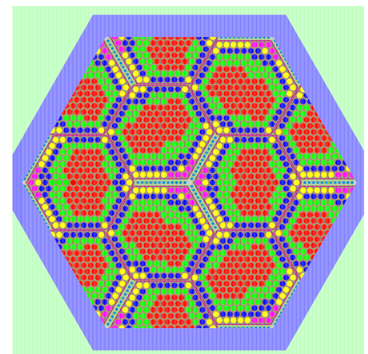
*PWR (Mihama NPP/Unit-3, Takahama NPP/Unit-3) by JAERI
*BWR (Fukushima NPP/Unit-1/3) by NUPEC
*REBUS program analyses by JNES

Conceptual Nuclear Design Study of Future Reactors

*Reduced-Moderation Water Reactors
*Space Power Reactor
*Rock-like oxide (ROX) fueled Reactor



ROX fuel pellets



Fuel lattice of RMWR

Integral Test of JENDL

*Benchmark calculations for more than 1000 experiments is now in progress for next JENDL

6. Future Plan and Conclusions

- LWR Assembly Calculation
  - Multi-group transport theory method with heterogeneous geometry
  - Anisotropic scattering
- LWR Core Calculation
  - 3-D multi-group cell averaged pin-by-pin transport calculation (SCOPE2)

- Boundary between Assembly and Core calculations will disappear.

<u>Example</u>

  - CASMO-4 heterogeneous fuel-core transport calculation
  - 2-D core calculation by PARAGON
- Micro Reactor Physics
  - Osaka University (Doppler Reactivity Effect)
  - PARAGON

- General
  - Monte-Carlo transient
  - Detailed Coupled system with Neutronic and Thermal-Hydraulic
- FR
  - Multi-group pin-by-pin fuel core transport

Conclusion

A lot of R&D Activities are being carried out for the improvement for LWR and FR Applications.