APPLICATION OF NUCLEAR DATA TO LIGHT WATER REACTOR CORE ANALYSIS

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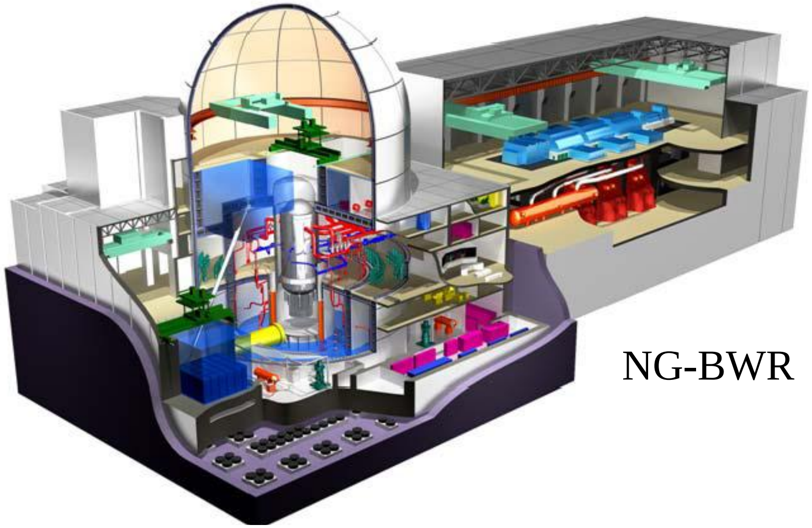
☐ Typical specification for next generation LWR

☐ V&V trends in design work

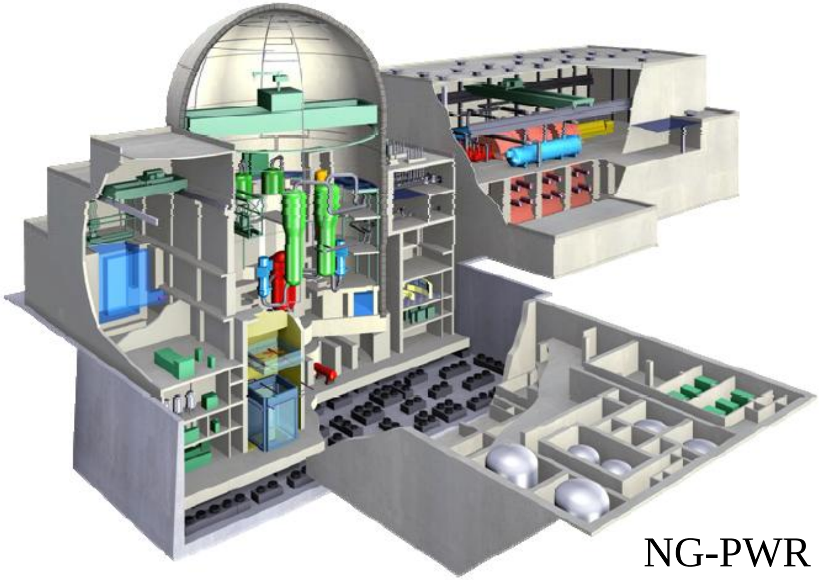
☐ Research in JAEA

- Overview of Monte Carlo code MVP & its application as reference solution code
- Overview of other V&V tools

Design Features of Next Generation-LWR



NG-BWR



NG-PWR

Item	NG-PWR
Power	1,780 MWe
Fuel burnup	70 - 90 GWd/t
Uranium enrichment	6 - 8 %
Fuel type	possible for full MOX
Cladding material	Zr, or Stainless steel
Coolant condition	Temperature: 330 ℃ Pressure: 15.4 MPa
Operation cycle	24 months
Plant life	80 years

(by Institute of Applied Energy, Japan, 2010)

Change of Design Policy in Nuclear Plant

☐ Requirement for Verification and Validation (V&V)

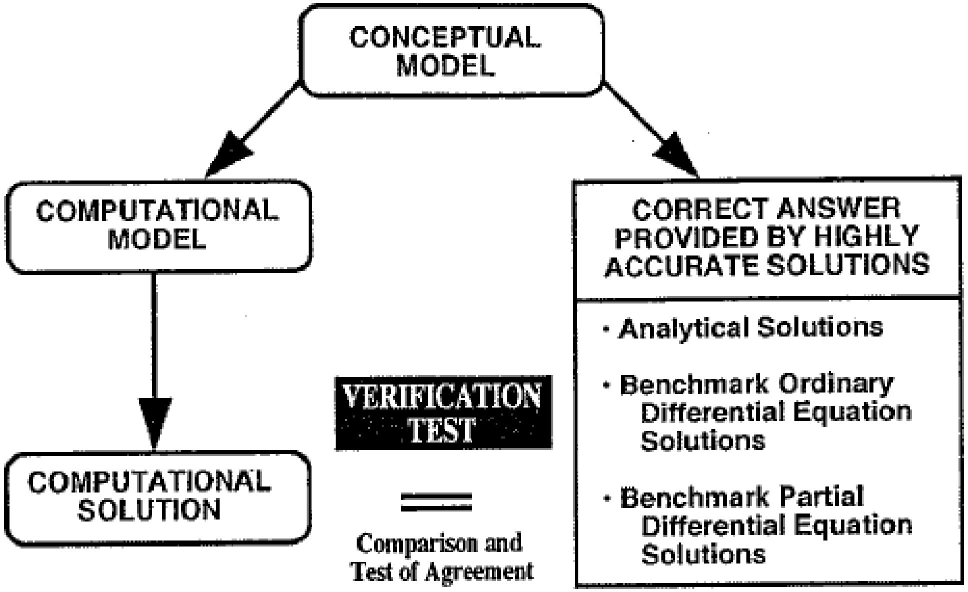


Figure 2
Verification Process

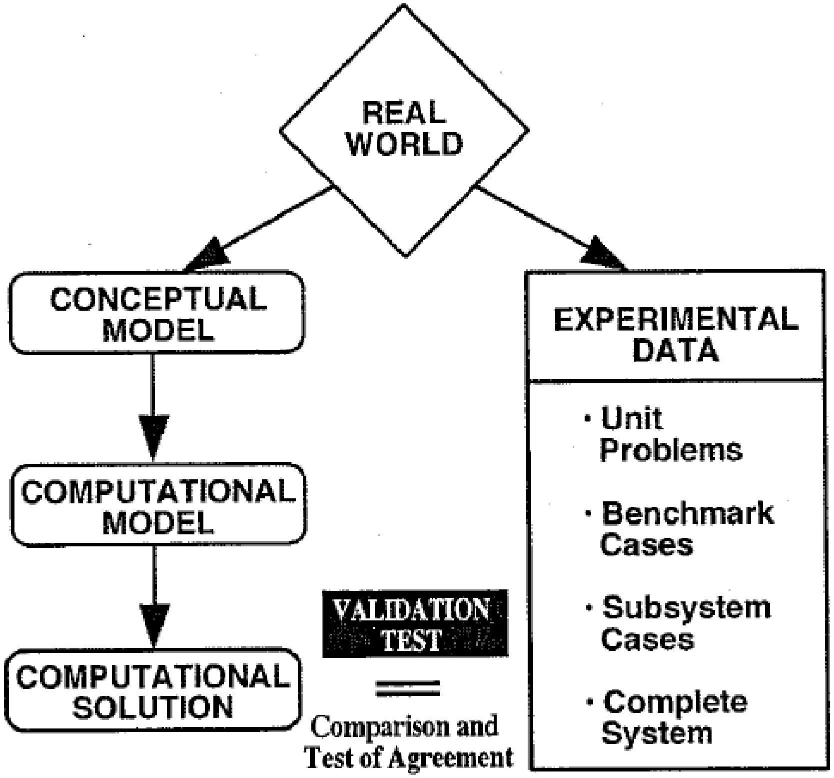


Figure 3
Validation Process

(from "Guide for the Verification and Validation of Computational Fluid Dynamics Simulations," American Institute of Aeronautics and Astronautics (AIAA), 2002)

Uncertainty Quantification for Nuclear Systems

Management of simulation uncertainties implies:

a) **Quantification of uncertainties** - that is assessing the accuracy of the simulation due to the various sources of errors by means of an uncertainty analysis.

b) **Understanding uncertainties** - that is identification of the key sources of uncertainties by means of a sensitivity analysis.

c) **Reduction of uncertainties** - that is using data assimilation by employing existing body of measurements to correct for the identified key sources of uncertainties.

d) **Routine application of uncertainties recognizing the increased demand from nuclear community** to provide confidence bounds with best-estimates predictions generated in support of the design, analysis, safety, and regulatory activities.

(by H.Abdel-Khalik, ***Verification and Validation for Nuclear Systems Analysis Workshop***, Idaho, July 2008)

Needs for Nuclear Data

☐ **To establish quality assurance system of nuclear data measurement and evaluation, as a part of V&V policy**

→ Accountability, Reproducibility, Traceability and Transparency

☐ **Quantitative reliability of covariances for major isotopes**

→ To be applied to uncertainty evaluation and improvement. Under rapid development in these few years

☐ **Improvement of minor actinides and fission products**

→ To manage the increased burnup, 70 - 90 GWd/t, and long operation period, 24 months

☐ **Improvement of thermal scattering law, S(α, β) data**

→ Mixing area with material science

Development of Monte Carlo Code MVP

☐ General Description

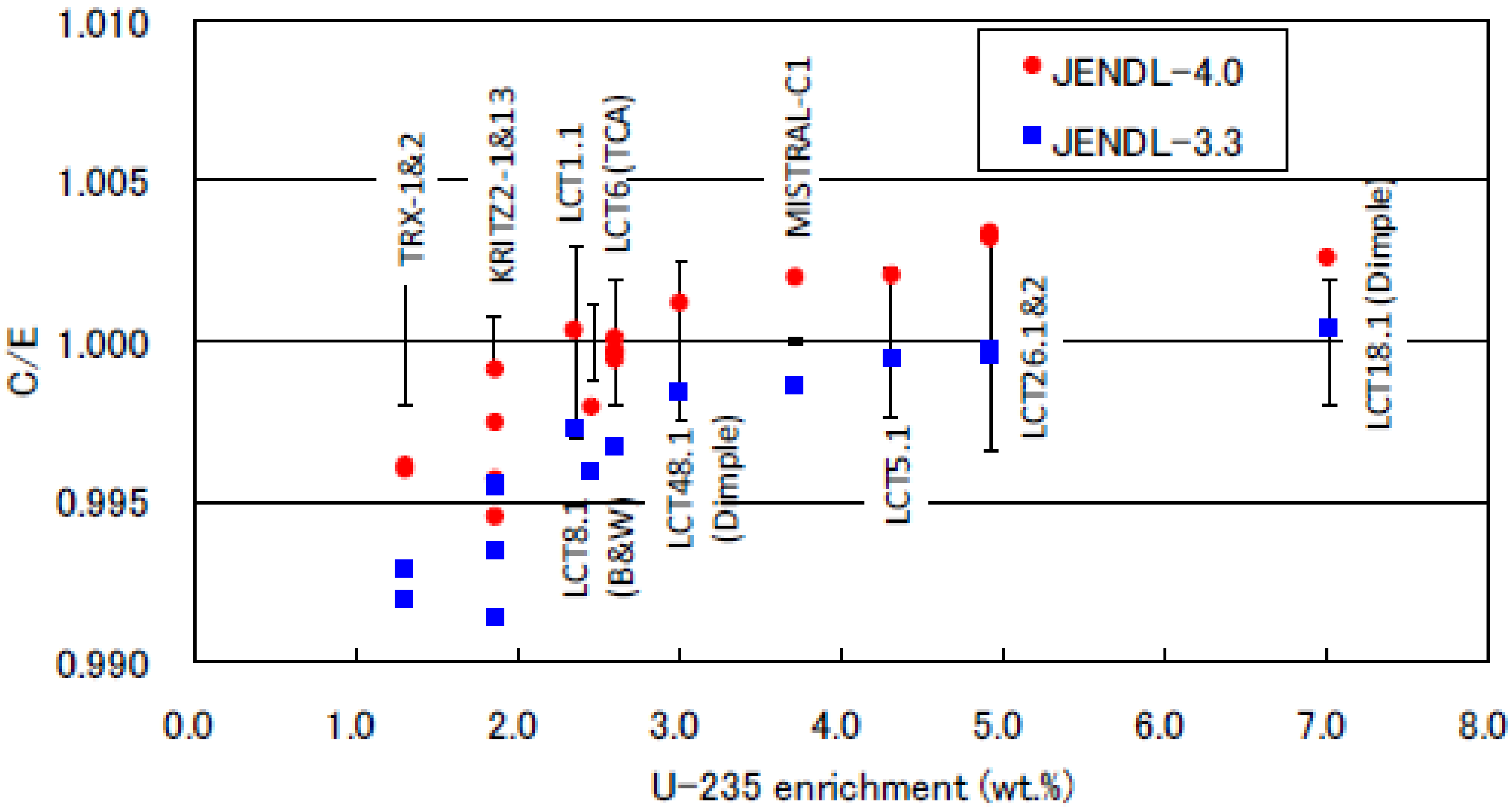
- Continuous-energy Monte Carlo code for neutron/photon transport.
- Implement various useful capabilities for nuclear reactor analysis of LWRs, FRs, HTGRs.
- Many experiences for benchmark/critical experiment analysis (V&V), reactor core design, etc.

☐ Capabilities

- Eigenvalue and fixed-source calculations for neutron/photon transport.
- Continuous-energy calculations based on ENDF.
- Flexible input geometry by combinatorial geometry and lattice geometry (repeated geometry).
- Arbitrary-temperature calculations.
- Burnup calculations with a bundled solver in the source suite.

Integral Benchmark Test (JENDL-4.0)

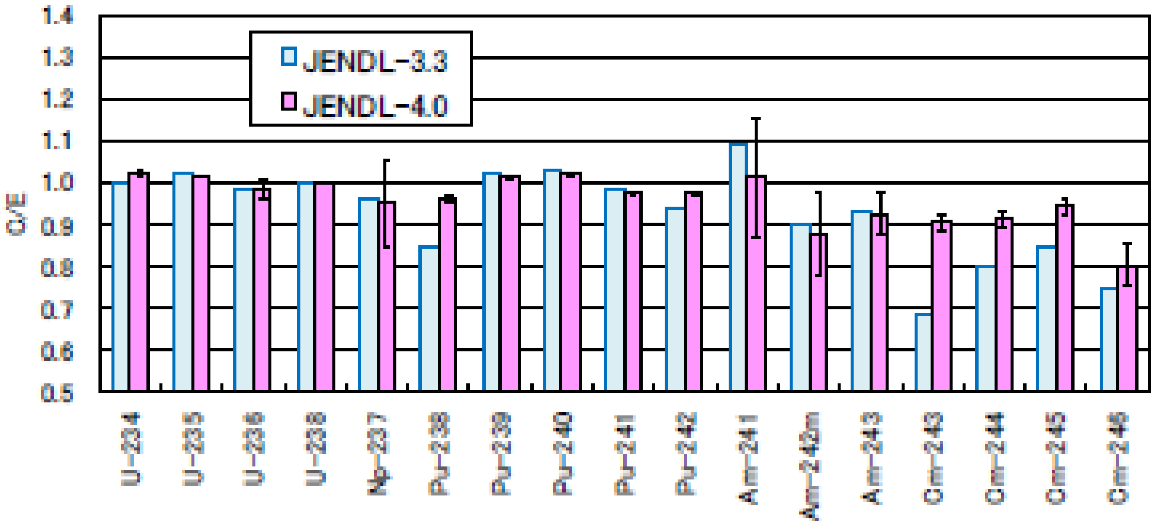
☐ An example for Criticality
(low enriched uranium-fueled light water moderated system)



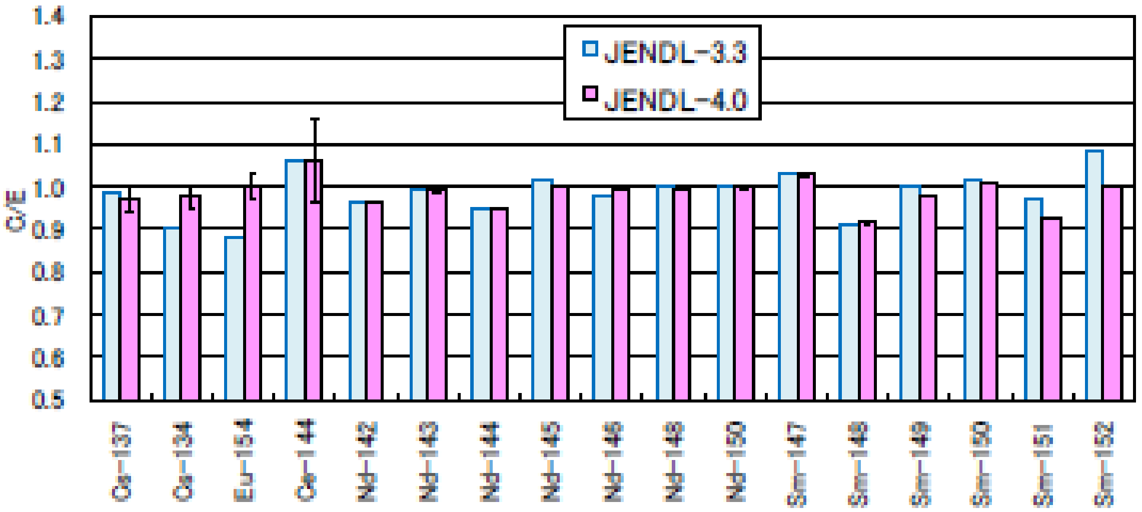
Integral Benchmark Test (JENDL-4.0)

☐ **An example for PIE analysis of PWR spent fuel**

Heavy nuclides



Fission products



New Capabilities in Next Release

- Simulation of delayed neutrons
- Multigroup cross section generation
  - Improvement of design calculation route
  - Identification of analytical errors in deterministic design methods
- Perturbation calculations for k_{eff} value
  - Accurate evaluation of feedback reactivity worth
  - Improvement of usefulness of integral experimental database in V&V activities
- Calculation of kinetics parameters β_{eff} and Λ
  - Key parameters for safety analysis
  - Scaling factor between measured and calculated reactivity worths
- Exact resonance elastic scattering model
  - Accurate evaluation of Doppler reactivity worth

Development of Tools for V&V

- ☐ Nuclear Data Covariance Processing Tools
  - ➢ ERRORJ : to calculate covariance of multi-group cross section
  - ➢ ERRORF : to calculate covariance of self-shielding factor and its temperature gradient used for temerature-related core characteristics
- ☐ Sensitivity Analysis Tools (Generalized perturbation theory)
  - ➢ Diffusion (SAGEP) : used for FBR design work
  - ➢ Transport : Depletion perturbation theory used for LWR core analysis
- ☐ Uncertainty Evaluation Methods
  - ➢ Cross section adjustment method : used for FBR design work
  - ➢ Extension of bias factor method

    : newly developed for evaluating error sources and for minimizing prediction uncertainty
  - ➢ Extension of cross section adjustment method

    : newly developed for minimizing prediction uncertainty in the framework of the cross section adjustment method

Burnup sensitivity coefficients

$$G\left(\sigma_x^g\right) = \frac{\sigma_x^g}{R}\left\{T_D + T_N + T_\phi + T_{\phi^*} + T_P\right\}$$

G: sensitivity coefficient

σ_x^g: microscopic cross section(XS) (x: reaction, g: energy range)

R: neutronic characteristic (including number densities)

T_D: Direct term : Direct Effect

T_N: Number density term : Effect via nuclide's transmutation (including fission yields, decay constants)

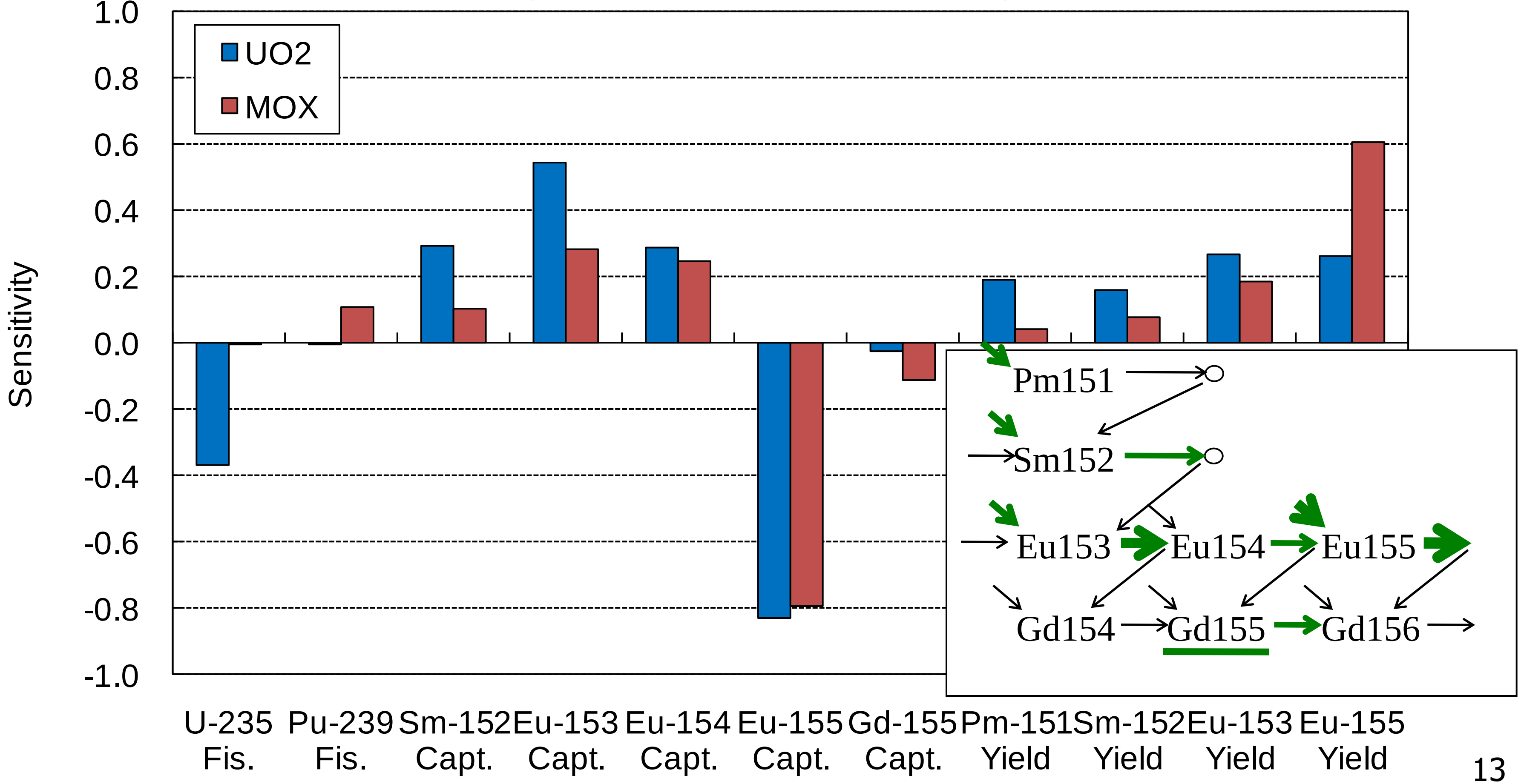
T_ϕ: Flux term : Effect through the neutron flux change

T_{ϕ^*}: Adjoint flux term : Effect through the adjoint flux change

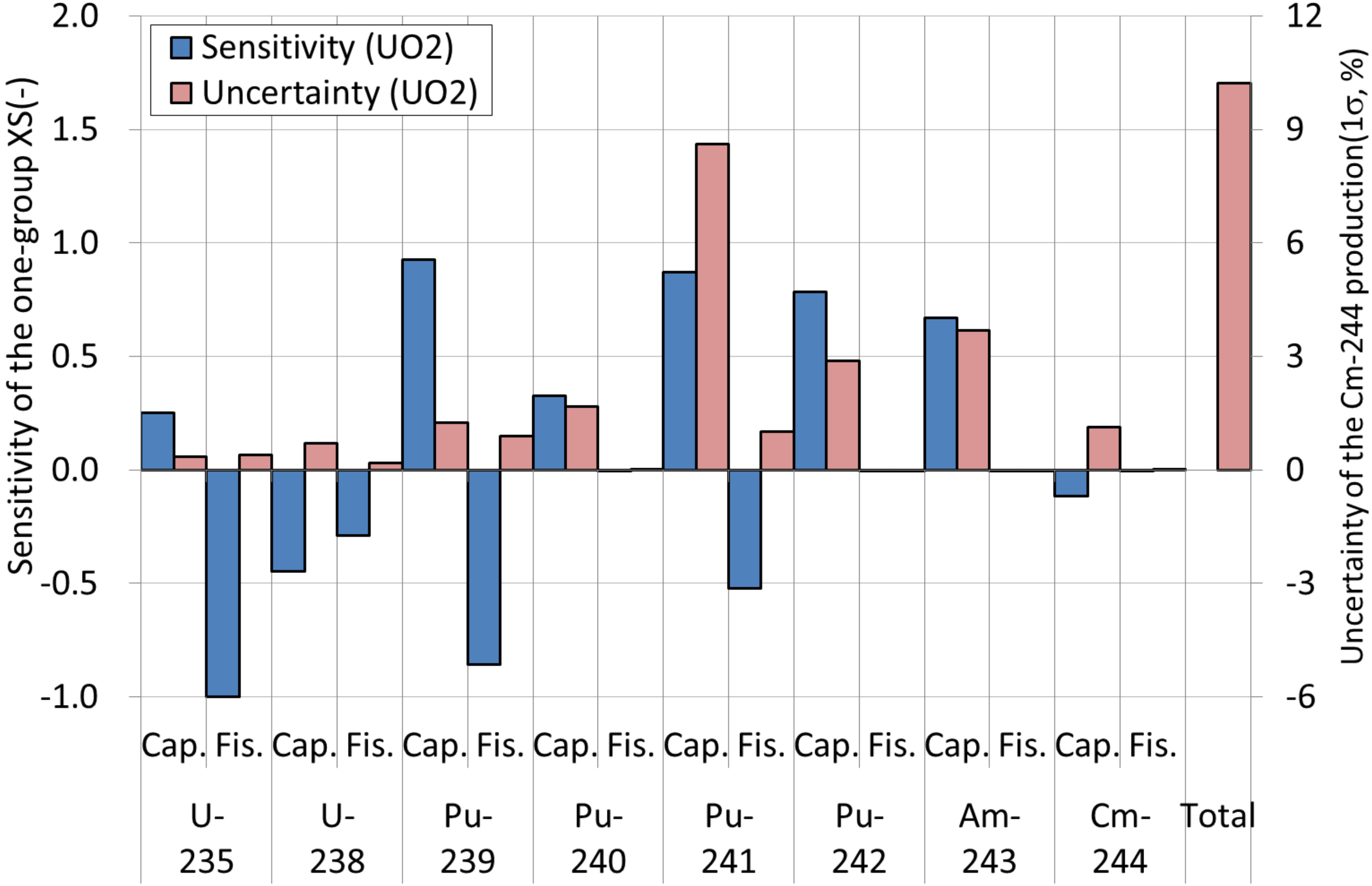
T_P: Power normalization term : Effect through the correlation between fission XS and flux (to keep power constant)

Sensitivity of Gd-155 Concentrations

- Sensitive to nuclear data of various nuclides
- Not so much difference between UO2 and MOX fuels
- Nuclides far from target nuclide have larger sensitivity in UO2



Sensitivity and Uncertainty analysis for LWR fuel compositions (Cm-244)



Summary

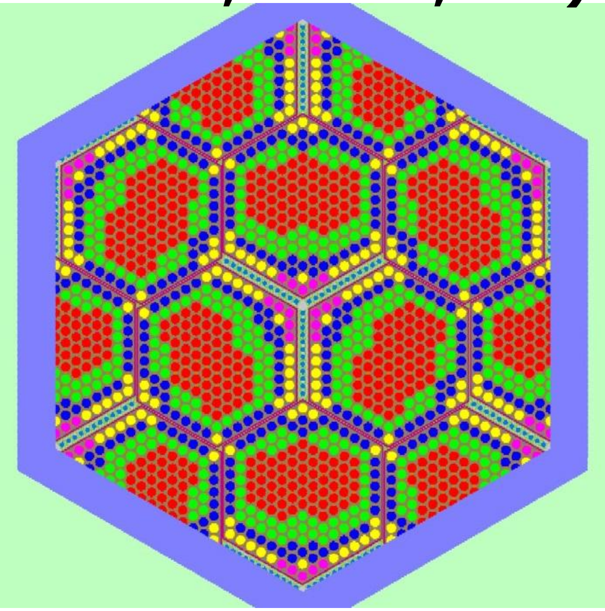
- ☐ In the LWR application, the nuclear data has already been placed as an industrial material, rather than a scientific art. This means the strong requirement for both the quality and the quantity of nuclear data, such as the covariance, and/or the V&V system.
- ☐ Research in JAEA for the V&V system in reactor physics field
  - Improvement of Monte Carlo code, MVP
  - Development of V&V tools
    - Nuclear data covariance processing tools
    - Extension of sensitivity analysis tools
    - Uncertainty analysis methods for reactor core design
    - Extension of reactor physics experimental data

Thank you for attention.

Experiences with MVP

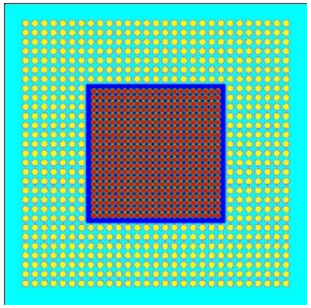
Testing Evaluated Nuclear Data, Integral Benchmark Test (Development of JENDL-4.0, Verification of ENDF/B, JEFF, etc.)

Reference solution for reactor design (FaCT, RMWRs, HTGRs, etc.)

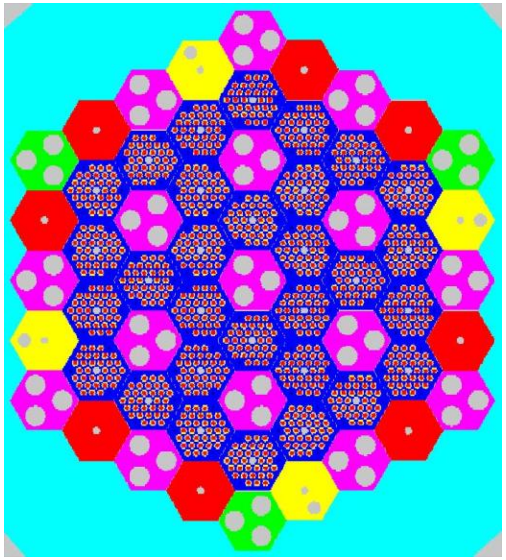


Reduced-Moderation Water Reactor (a Y-shaped control rod is located at the center.)

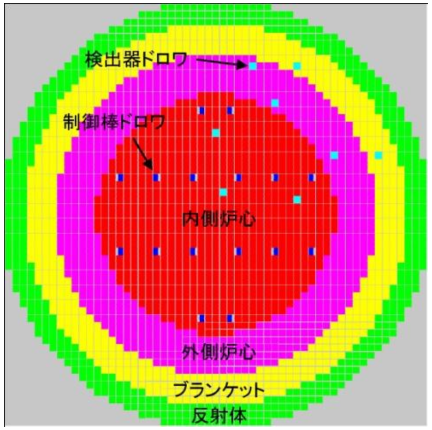
Analysis of various critical experiments (TCA, FCA, STACY, HTTR , ZPPR, etc.)



Tank-type Critical Assembly



High-Temperature engineering Test Reactor

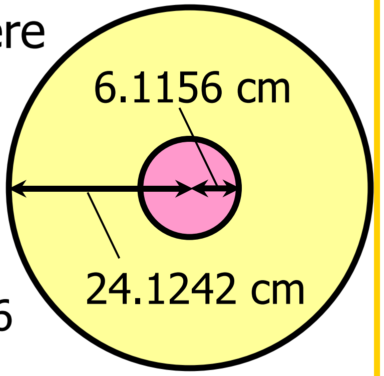


Fast Critical Assembly, ZPPR-9

New Capability : β_{eff} Calculation

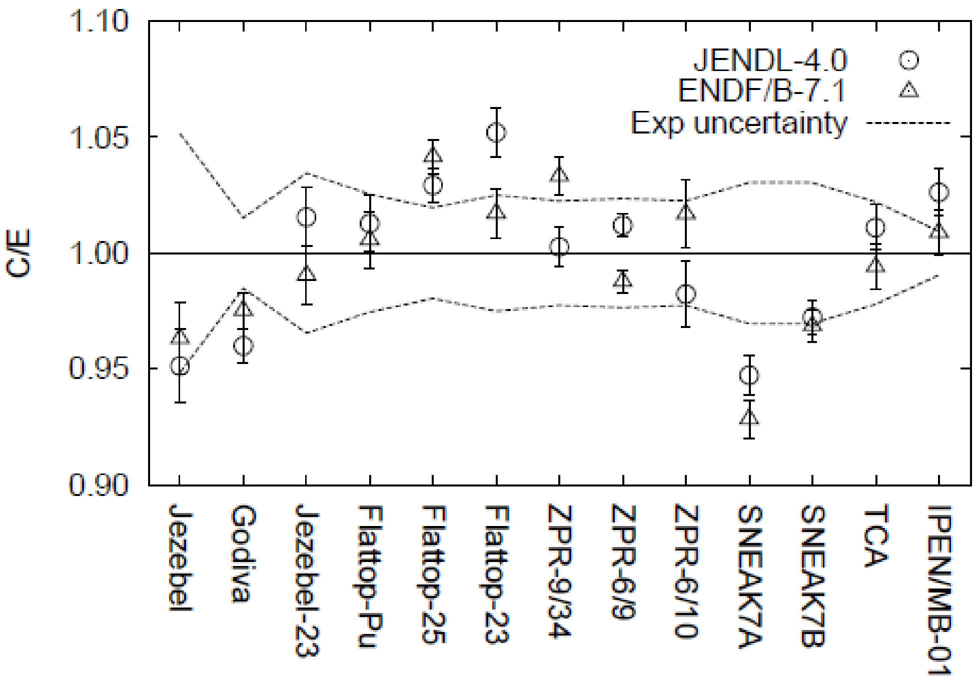
Benchmark for Flattop-U core

- Reflected U sphere
- Core
  ~93 wt% ^{235}U
- Reflector
  Normal uranium
- HEU-MET-FAST-006



	β_{eff} (pcm)
Deterministic (Reference)	678.5
Diff. Oper. Samp.	**679.8(3.4)**
Meulekamp (Next Fis. Prob.)	643.9(2.4)
Nauchi (Next Fis. Prob.)	609.4(2.3)

Integral benchmark test for effective delayed neutron fraction



Overestimation for the uranium-233 systems

New Resonant Scattering Model

Scattering models used conventionally in MC codes

- Asymptotic Slowing-Down Scattering Model (Asymptotic Model)
- Free-Monatomic-Gas Model with Constant Cross Section (Free-gas Model)

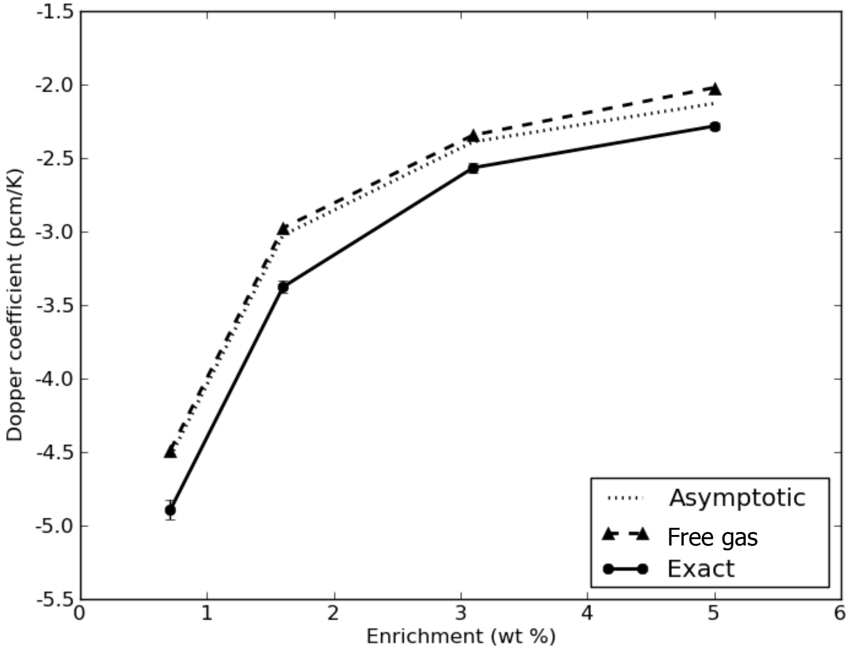
Approx. of Free-gas model

$$\sigma_s^{\mathrm{eff}}(v) = \frac{1}{v}\iint \sigma_s(v_{\mathrm{rel}})v_{\mathrm{rel}}M(V,\mu)dVd\mu$$

$$\approx \frac{\sigma_{s0}}{v}\iint v_{\mathrm{rel}}M(V,\mu)dVd\mu$$

- Free-Monatomic-Gas Model with Resonance Cross Section (Exact Model)

Mosteller Doppler defect benchmark for UO_2 pin cell geometry



The asymptotic & free-gas models negatively underestimate the reference Doppler coeffcients by ~10%.

New Capability : β_{eff} Calculation

Methodology

Consider a system where the number of delayed neutrons can be changed.

$$L\Phi(a) = \frac{1}{k(a)}\left[F^{p}\Phi(a) + (1+a)F^{d}\Phi(a)\right]$$

a is the fractional change in the number of delayed neutrons.

a = 0 : unperturbed system

a = -1 : no delayed neutrons

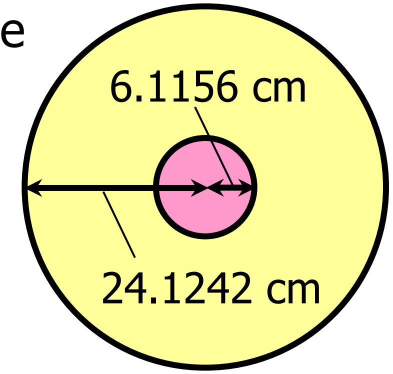
$$\beta_{\text{eff}} = \lim_{a\to 0}\frac{1}{k(0)}\frac{k(a)-k(0)}{a} = \frac{1}{k(0)}\boxed{\left.\frac{\partial k(a)}{\partial a}\right|_{a=0}}$$

Estimate differential coefficient

Need to estimate the perturbed source effect.

Benchmark for Flattop-U core

- Reflected U sphere
- Core
  ~93 wt% ^{235}U
- Reflector
  Normal uranium
- HEU-MET-FAST-006



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Example of Sensitivity to Fission Product Concentrations

- Sensitivity to capture cross section

  (Ex. Gd-155 concentration at EOL

  of g-group capture cross section of Eu-155)

$$\frac{dN^{EOL}_{Gd-155}}{N^{EOL}_{Gd-155}} \Bigg/ \frac{d\sigma^{g}_{cap,Eu-155}}{\sigma^{g}_{cap,Eu-155}}$$

- Sensitivity to fission yields

  (Ex. Gd-155 concentration at EOL

  of Eu-155 cumulative fission yield of U-235)

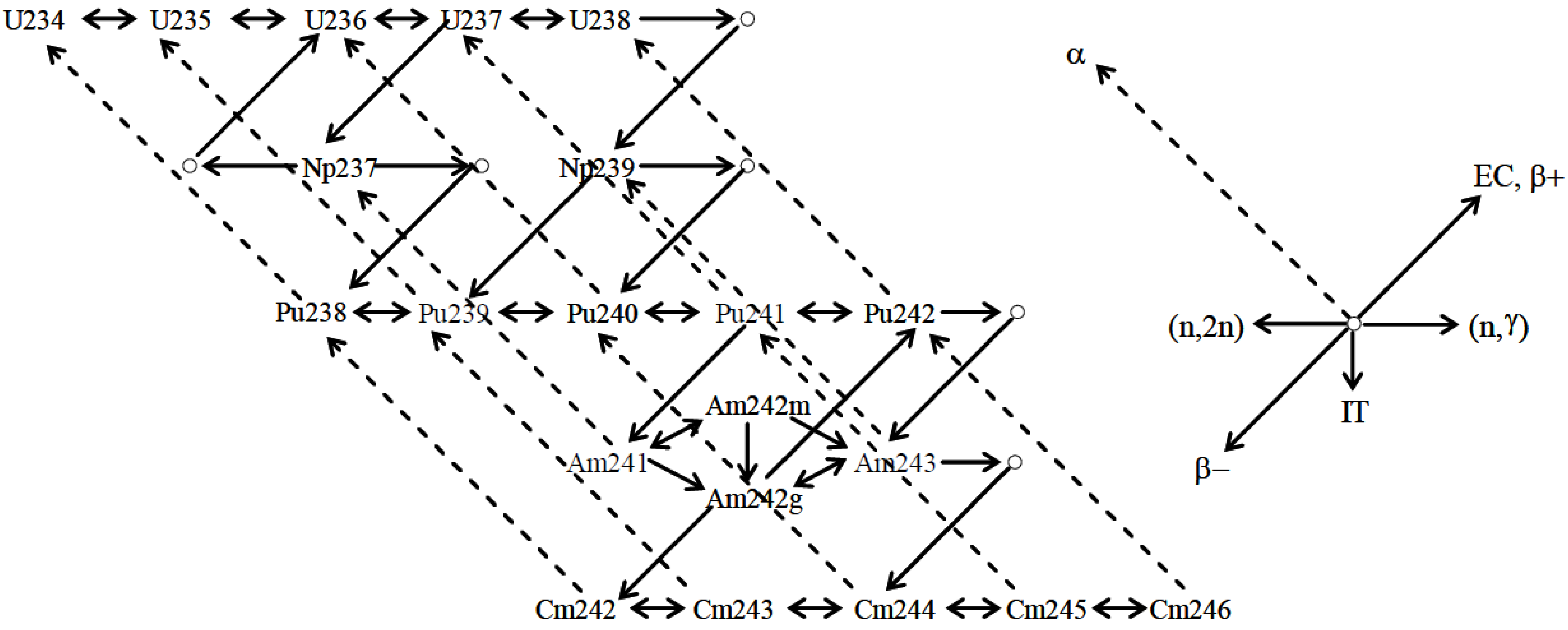
$$\frac{dN^{EOL}_{Gd-155}}{N^{EOL}_{Gd-155}} \Bigg/ \frac{dY^{U-235}_{Eu-155}}{Y^{U-235}_{Eu-155}}$$

- Sensitivity to fission cross section

  (Ex. Gd-155 concentration at EOL

  of g-group fission cross section of U-235)

$$\frac{dN^{EOL}_{Gd-155}}{N^{EOL}_{Gd-155}} \Bigg/ \frac{d\sigma^{g}_{fis,U-235}}{\sigma^{g}_{fis,U-235}}$$



U234 ↔ U235 ↔ U236 ↔ U237 ↔ U238

Np237 Np239

Pu238 ↔ Pu239 ↔ Pu240 ↔ Pu241 ↔ Pu242

Am242m

Am241 Am243

Am242g

Cm242 ↔ Cm243 ↔ Cm244 ↔ Cm245 ↔ Cm246

α

EC, β+

(n,2n)

(n,γ)

IT

β−

Uncertainty Analysis Methods

- Conventional Bias Factor Method (with Mock-up exp.) $f = \dfrac{E}{C}$
  (Only experimental uncertainty is introduced)

↓ (without Mock-up exp.)

$$f_{EB} = \frac{E_{EB}}{C_{EB}} = \frac{\prod_{i=1}^{N} E_i^{F_i}}{\prod_{i=1}^{N} C_i^{F_i}}$$

- Extended Bias Factor Method (EB)
  - Minimize prediction uncertainty by maximizing the correlation between the design core and a number of past experimental data with respect to cross section induced and analytical uncertainties
- Conventional Cross Section Adjustment Method (CA)
  - Adjust the cross sections based on the Bayesian theory with a least-square technique (C/E values, experimental, analytical and cross section induced uncertainties of various experimental cores and parameters are synthesized)
- Extended Cross Section Adjustment Method (EA)
  - Achieve the minimized prediction uncertainty of EB method in cross section adjustment method

Application of uncertainty Analysis Methods

- Comparison of effectiveness in FBR core design

	CA		EB		EA	
	Pre-diction	Un-certainty (%)	Pre-diction	Un-certainty (%)	Pre-diction	Un-certainty (%)
keff	**0.9977**	**0.30**	**0.9982**	**0.21**	**0.9982**	**0.21**
Power Distribu-tion	**1.000**	**1.4**	**0.996**	**0.9**	**0.996**	**0.9**
Control rod worth	**0.969**	**1.4**	**0.968**	**1.1**	**0.968**	**1.1**

Variance by Extended Bias Factor Method

- Variance by EB method

$$V\left(\frac{R_R}{R_t}\right)=\left(S_R-\sum_{i=1}^{N}F_iS_i\right)V_\sigma\left(S_R-\sum_{i=1}^{N}F_iS_i\right)^t+V\left(\Delta M_R-\sum_{i=1}^{N}F_i\Delta M_i\right)+V\left(\sum_{i=1}^{N}F_i\Delta E_i\right)$$

Cross section error Analysis method error Exp. error

- Determine the exponents to minimize the variance

  (1st order derivatives of variance with respect to the exponents are set to be zero)

$$\frac{\partial V\left(\frac{R_R}{R_t}\right)}{\partial F_i}=0$$

  - Maximize the correlation between the target core and the experimental data with respect to uncertainties arising from cross section and analytical errors
  - Avoid introduction of experimental error
  - Totally minimize uncetainty due to cross section, analysis method and experimental errors

Uncertainty Evaluation by Cross-section Adjustment Method

☐ Conventional Cross-section Adjustment Method

$$\mathbf{R}^{(2)}_{CA} = \mathbf{R}^{(2)}_{c}(\mathbf{T}_{CA}) = \mathbf{R}^{(2)}_{c}(\mathbf{T}_{0}) + \mathbf{G}^{(2)}[\mathbf{T}_{CA} - \mathbf{T}_{0}]$$

$$V(\mathbf{R}^{(2)}_{CA}) = \mathbf{G}^{(2)}\mathbf{M}_{CA}\mathbf{G}^{(2)T} + \mathbf{V}^{(2)}_{m} - \mathbf{K}_{CA}\mathbf{V}^{(12)}_{m} - \mathbf{V}^{(12)T}_{m}\mathbf{K}^{T}_{CA}$$

$$\mathbf{K}_{CA} = \mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T}[\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(1)}_{m} + \mathbf{V}^{(1)}_{e}]^{-1}$$

$$\mathbf{T}_{CA} = \mathbf{T}_{0} + \mathbf{M}\mathbf{G}^{(1)T}[\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(1)}_{m} + \mathbf{V}^{(1)}_{e}]^{-1}[\mathbf{R}^{(1)}_{e} - \mathbf{R}^{(1)}_{c}(\mathbf{T}_{0})]$$

$$V(\mathbf{T}_{CA}) \equiv \mathbf{M}_{CA} = \mathbf{M}\mathbf{G}^{(1)T}[\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(1)}_{m} + \mathbf{V}^{(1)}_{e}]^{-1}\mathbf{G}^{(1)}\mathbf{M}$$

☐ Extended Cross-section Adjustment Method

$$\mathbf{R}^{(2)}_{EA} = \mathbf{R}^{(2)}_{c}(\mathbf{T}_{EA}) = \mathbf{R}^{(2)}_{c}(\mathbf{T}_{0}) + \mathbf{G}[\mathbf{T}_{EA} - \mathbf{T}_{0}]$$

$$V(\mathbf{R}^{(2)}_{CA}) = \mathbf{G}^{(2)}\mathbf{M}_{EA}\mathbf{G}^{(2)T} + \mathbf{V}^{(2)}_{m} - \mathbf{K}_{EA}\mathbf{V}^{(12)}_{m} - \mathbf{V}^{(12)T}_{m}\mathbf{K}^{T}_{EA}$$

$$\mathbf{K}_{EA} = [\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(12)T}_{m}][\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(1)}_{e} + \mathbf{V}^{(1)}_{m}]^{-1}$$

$$\mathbf{T}_{EA} = \mathbf{T}_{0} + [\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{G}^{(2)+}\mathbf{V}^{(12)T}_{m}][\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(1)}_{m} + \mathbf{V}^{(1)}_{e}]^{-1}[\mathbf{R}^{(1)}_{e} - \mathbf{R}^{(1)}_{c}(\mathbf{T}_{0})]$$

$$\begin{aligned} V(\mathbf{T}_{EA}) \equiv \mathbf{M}_{EA} = {} & [\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{G}^{(2)+}\mathbf{V}^{(12)T}_{m}][\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(1)}_{m} + \mathbf{V}^{(1)}_{e}]^{-1}[\mathbf{G}^{(2)+}\mathbf{V}^{(12)T}_{m}] \\ & -[\mathbf{M}\mathbf{G}^{(1)T}][\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{V}^{(1)}_{m} + \mathbf{V}^{(1)}_{e}][\mathbf{M}\mathbf{G}^{(1)T} + \mathbf{G}^{(2)+}\mathbf{V}^{(12)T}_{m}]^{T} \end{aligned}$$

Uncertainty Evaluation by Cross-section Adjustment Method

☐ Conventional Cross-section Adjustment Method

- Adjustedcross-section set

$$\mathbf{T}_{CA}=\mathbf{T}_0+\mathbf{M}\mathbf{G}^{(1)T}[\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T}+\mathbf{V}_m^{(1)}+\mathbf{V}_e^{(1)}]^{-1}[\mathbf{R}_e^{(1)}-\mathbf{R}_c^{(1)}(\mathbf{T}_0)]$$

- Variance

$$V(\mathbf{R}_{CA}^{(2)})=\mathbf{G}^{(2)}\mathbf{M}_{CA}\mathbf{G}^{(2)T}+\mathbf{V}_m^{(2)}-\mathbf{K}_{CA}\mathbf{V}_m^{(12)}-\mathbf{V}_m^{(12)T}\mathbf{K}_{CA}^T$$

$$\text{where, }\mathbf{K}_{CA}=\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T}[\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T}+\mathbf{V}_m^{(1)}+\mathbf{V}_e^{(1)}]^{-1}$$

☐ Extended Cross-section Adjustment Method

- Adjusted cross-section set

$$\mathbf{T}_{EA}=\mathbf{T}_0+[\mathbf{M}\mathbf{G}^{(1)T}+\mathbf{G}^{(2)+}\mathbf{V}_m^{(12)T}][\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T}+\mathbf{V}_m^{(1)}+\mathbf{V}_e^{(1)}]^{-1}[\mathbf{R}_e^{(1)}-\mathbf{R}_c^{(1)}(\mathbf{T}_0)]$$

- Variance

$$V(\mathbf{R}_{CA}^{(2)})=\mathbf{G}^{(2)}\mathbf{M}_{EA}\mathbf{G}^{(2)T}+\mathbf{V}_m^{(2)}-\mathbf{K}_{EA}\mathbf{V}_m^{(12)}-\mathbf{V}_m^{(12)T}\mathbf{K}_{EA}^T$$

$$\text{where, }\mathbf{K}_{EA}=[\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T}+\mathbf{V}_m^{(12)T}][\mathbf{G}^{(1)}\mathbf{M}\mathbf{G}^{(1)T}+\mathbf{V}_e^{(1)}+\mathbf{V}_m^{(1)}]^{-1}$$