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Methods and issues for the combined use of integral experiments and covariance data

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Adjustment Study in JAEA

Makoto ISHIKAWA and Kazuteru SUGINO

Japan Atomic Energy Agency (JAEA)

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Objective and History of Adjustment Study for FBR

1. Objective

To improve the prediction accuracy of FBR core nuclear design, the adjusted cross-section set, which combined integral experimental information with differential nuclear data, is being developed in Japan.

2. History

1) 1989 – 1991: A cooperative study between JAEA and JAPC was performed to create the **ADJ91** set based on JENDL-2 --> Applied to the design study of the Demonstration FBR.

2) 1992 – 1995: JAEA, Hitachi and Osaka Univ. developed the sensitivity method and analytical system for the burnup characteristics and Doppler reactivity.

3) 1999 – 2001: JAEA developed the **ADJ2000** set based on JENDL-3.2.

--> Being used in the future FBR project (FS and FaCT).

4) 2002 – 2009: Extension of integral experimental database, and improvement of nuclear data and covariances. --> for the next **ADJ2010** set based on JENDL-4.

JAEA

Main Features of Adjusted Cross-section Sets

	ADJ91	ADJ2000	ADJ2010
Basic library	JENDL-2 (Final 1989, First 1982)	JENDL-3.2 (1994)	JENDL-4 (2010)
Nuclear parameters to be adjusted	σ_∞ of 11 nuclides (32 reactions), χ of 2 nuclides, β of 6 nuclides	σ_∞ of 11 nuclides (41 reactions), χ of 2 nuclides, β of 6 nuclides, self-shielding factors of U-238	In addition, MA nuclides
Nuclear data covariance	Rough estimation from difference between measured values and JENDL-2	Covariance data file of JENDL-3.2, evaluated by the Japan Sigma Committee	Covariance data file of JENDL-4
Integral experimental data	82 data from JUPITER experiment	237 data from JUPITER, FCA, JOYO, BFS, MASURCA and Los Alamos, including burnup and Doppler data.	In addition, MA irradiation data, ZEBRA, SEFOR, Monju

Theory of Cross-section Adjustment

* J.B.Dragt, et al.: "Methods of Adjustment and Error Evaluation of Neutron Capture Cross Sections; Application to Fission Product Nuclides," NSE 62, pp.117-129, 1977.

- Based on the Bayes theorem, i.e., the conditional probability estimation method
  → To maximize the posterior probability that a cross-section set, ***T***, is true, under the condition that the information of integral experiment, ***Re***, is obtained.

$$J(T) = (T-T_0)^t M^{-1}(T-T_0) + [Re-Rc(T)]^t[Ve+Vm]^{-1}[Re-Rc(T)]$$

Minimize the function ***J(T)***. → $dJ(T)/dT = 0$

- The adjusted cross-section set ***T'***, and its uncertainty (covariance), ***M'*** *(Algebra)*

$$T' = T_0 + MG^t[GMG^t+Ve+Vm]^{-1}[Re-Rc(T_0)]$$

$$M' = M - MG^t[GMG^t+Ve+Vm]^{-1}GM$$

✓If $GMG^t \ll Ve+Vm$, $T' \fallingdotseq T_0$ and $GM'G^t \fallingdotseq GMG^t$

✓If $GMG^t \gg Ve+Vm$, $GM'G^t \fallingdotseq Ve+Vm$

✓If $GMG^t \fallingdotseq Ve+Vm$, $GM'G^t \fallingdotseq 1/2 \times GMG^t$

- Prediction error induced by the cross-section errors

Before adjustment: GMG^t After adjustment: $GM'G^t$

Where,
- T_0 : Cross-section set before adjustment
- M : Covariance before adjustment
- Re : Measured values of integral experiments
- Rc : Analytical values of integral experiments
- Ve : Experimental errors of integral experiments
- Vm : Analytical modeling errors of integral experiments
- G : Sensitivity coefficients, $(dR/R)/(d\sigma/\sigma)$

Sensitivity Coefficient of Doppler Reactivity

* M.Ishikawa, K.Sugino, et al.: "Development of a Unified Cross-section Set ADJ2000 based on Adjustment Technique for Fast Reactor Analysis," Proc. of ND2001, pp.1073-1076.

- Conventional sensitivity method treats only **infinitely-diluted cross-sections.**

  → Impossible to evaluate Doppler reactivity.

- Doppler reactivity: $$R = \frac{1}{k_{eff,low}} - \frac{1}{k_{eff,high}}$$

- Relationship of effective cross-sections with temperature : $$\sigma_{eff,high} \approx \left[f_{low} + \left(\frac{df}{dT}\right)\Delta T \right]\sigma_{\infty,low} = (1 + f'\Delta T)\sigma_{eff,low}$$

  ↓

  where, $$f' \equiv \frac{1}{f_{low}}\left(\frac{df}{dT}\right) \equiv \alpha$$

- Introduction of ***pseudo-cross-sections, df/dT (=f')***:

$$S_{f'} \equiv \frac{dR/R}{df'/f'} = \left(\frac{\sigma_{eff,high} - \sigma_{eff,low}}{\sigma_{eff,high}}\right) \times \frac{1}{R} \times \frac{S_{k_{eff},high}}{k_{eff,high}}$$

where, $$S_{k_{eff},high} = \frac{dk_{eff,high}/k_{eff,high}}{d\sigma_{\infty,high}/\sigma_{\infty,high}}$$

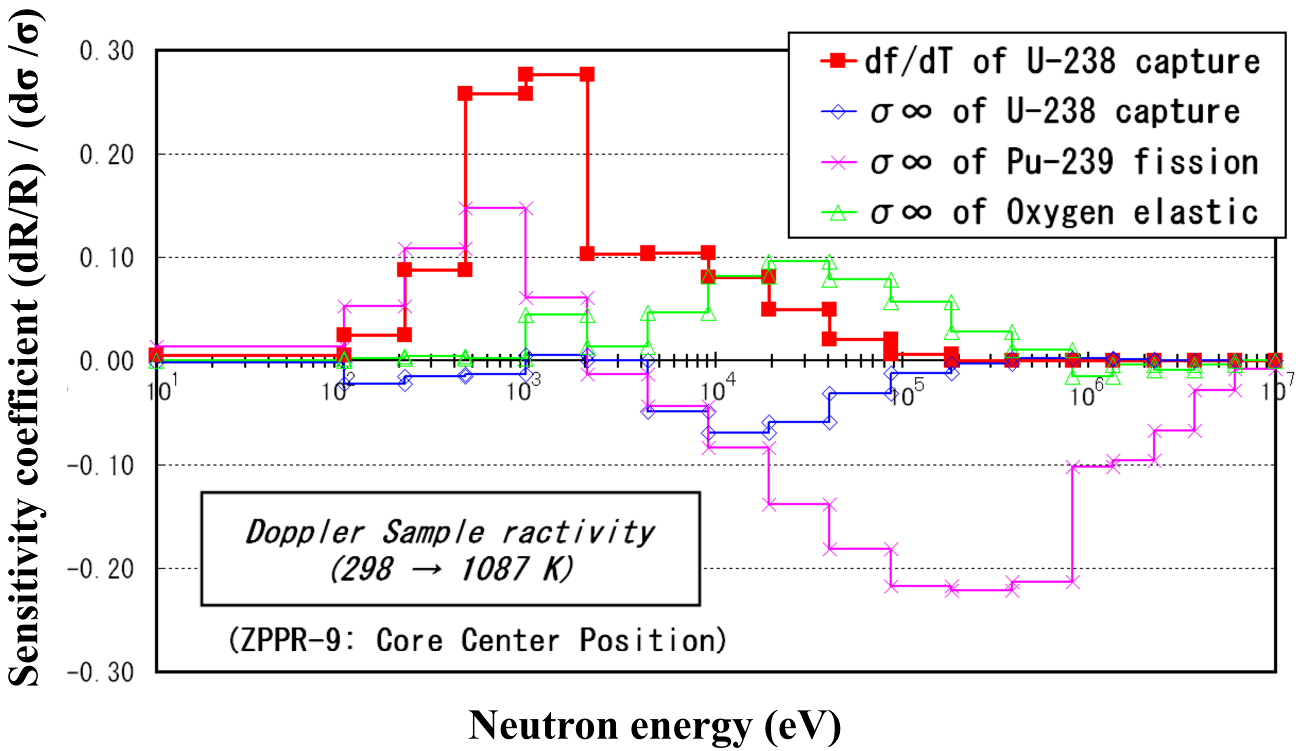
(Merits) 1) Easily calculated from the sensitivity of criticality, S_{keff},

2) No influence to the self-shielding factors at room temperature.

JAEA

Sensitivity for Sample Doppler Reactivity (ZPPR-9 Core)

- The df/dT of U-238 capture has largely positive sensitivity at keV energy region,
- Sensitivity of Pu-239 fission is negative, since it increases the denominator of perturbation,
- There is also a certain sensitivity to space-related reactions, since it also has the characteristics of local sample reactivity.



Generalized Perturbation Theory for Burnup characteristics

* M.L.Williams: " Development of Depletion Perturbation Theory for Coupled Neutron/Nuclide Fields, " NSE 70, pp.20-36, 1979.

- Needs: 1) Use of power reactor data such as burnup reactivity loss or composition changes of fuel nuclides
  2) Evaluation of FBR nuclear design accuracy

- Net sensitivity: $$S(\sigma_x^g) = \frac{dR/R}{d\sigma_x^g/\sigma_x^g} = \frac{\sigma_x^g}{R} \times \left\{ S_D + S_N + S_\phi + S_{\phi^*} + S_P \right\}$$

$$S_D = \sum_{i=1}^{I} \left[\int_t^{t_{i+1}} dt \frac{\partial R}{\partial \sigma_x^g} \right]_{E,V}$$: Direct term

$$S_N = \sum_{i=1}^{I} \int_t^{t_{i+1}} dt \left[N^* \frac{\partial M}{\partial \sigma_x^g} N \right]_{E,V}$$: Atomic number density term

$$S_\phi = \sum_{i=1}^{I+1} \left[\Gamma_i^* \frac{\partial B}{\partial \sigma_x^g} \phi_i \right]_{E,V}$$: Normal flux term

$$S_{\phi^*} = \sum_{i=1}^{I+1} \left[\Gamma_i \frac{\partial B}{\partial \sigma_x^g} \phi_i^* \right]_{E,V}$$: Adjoint flux term

$$S_P = \sum_{i=1}^{I+1} \left[P_i^* \frac{\partial P_i}{\partial \sigma_x^g} \right]_{E,V}$$: Power normalization term

where,

$\frac{\partial}{\partial t} N(t) = M \times N(t)$: Burnup equation

$P_i = \int_{E,V} dE dV \left[\kappa \sigma_f N \phi_i \right]$: Reactor power

P^* : Adjoint power

N_i^* : Adjoint atomic number density

Sensitivity for Burnup Reactivity Loss (JOYO Mk-I Core)

- Direct term is negative because it increases Denominator
- Atomic number density term is positive since it acceralates the decrease of Pu-239.
- Power normalization term is negative because it lower the neutron flux level.

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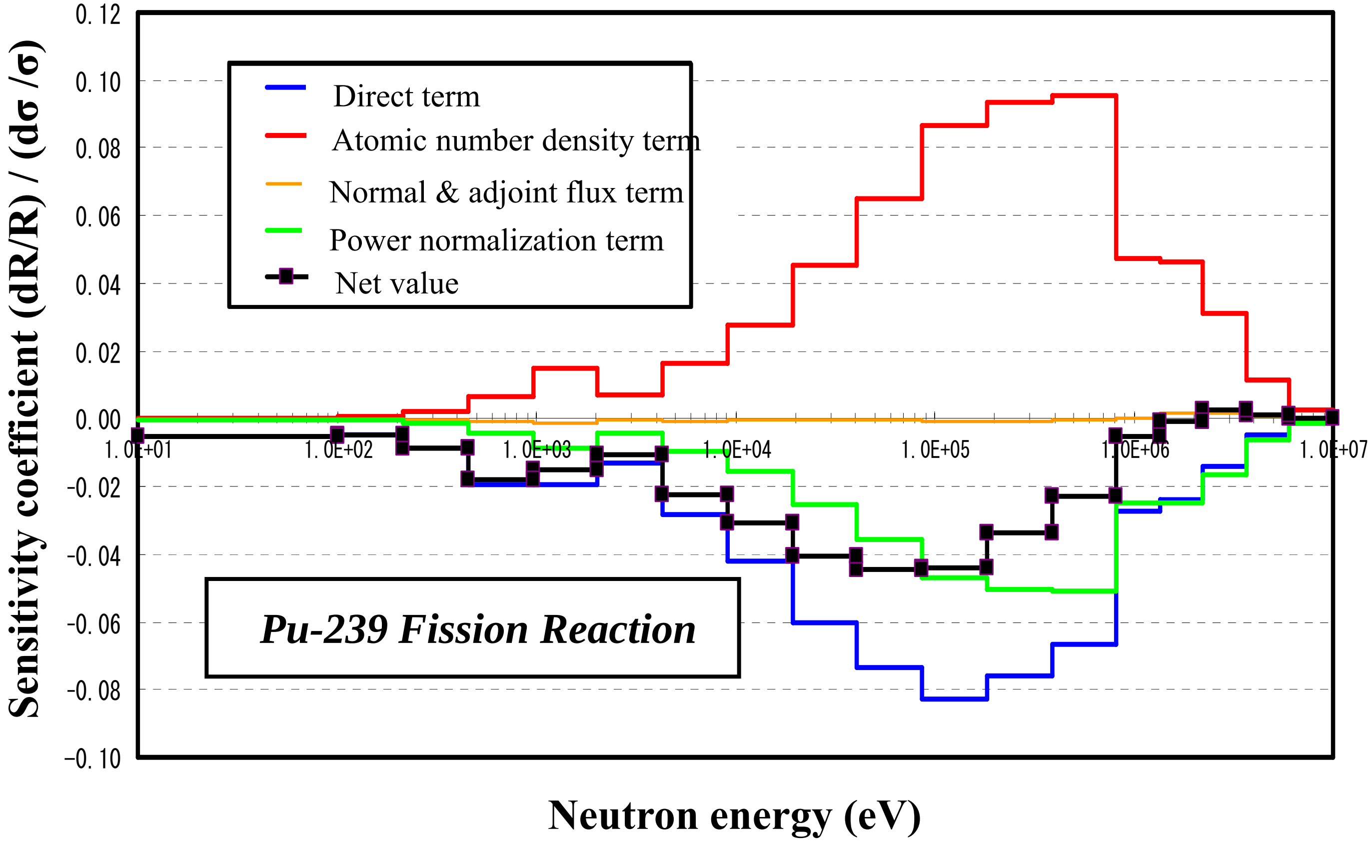
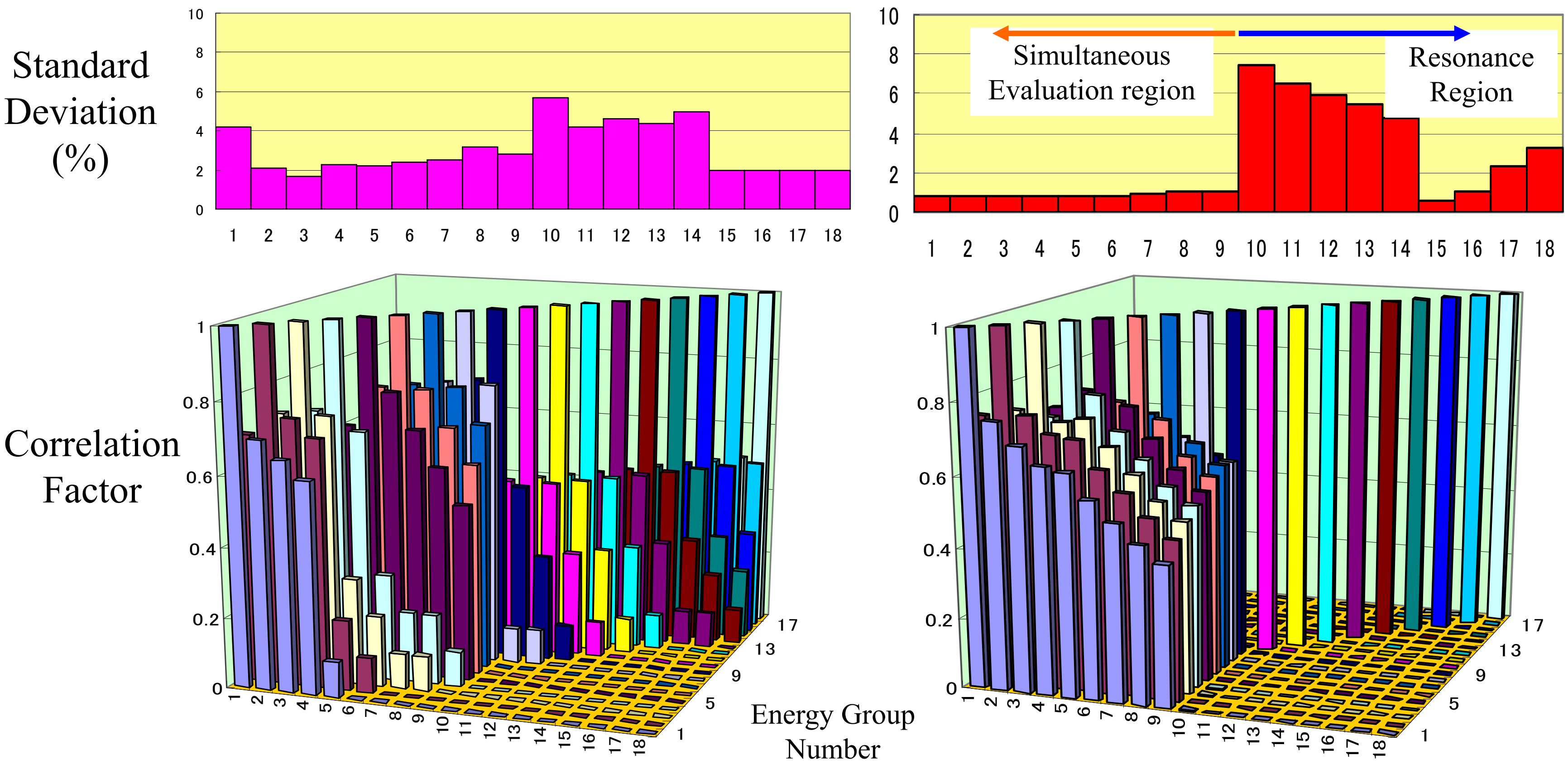
Net sensitivity of Pu-239 fission is slightly negative because of these cancellation.



Group-structure Covariances
(Comparison of Pu-239 Fission Cross-section)

* K.Shibata, et al.: “JENDL-3.2 Covariance File,” Proc. of ND2001, pp.40-43.



Old Covariance by Rough Estimation (1991)

New Covariance by Sigma Committee (2000)

to JENDL-3.2

Integral Data for Fast Reactors (1/4)

- **JUPITER Critical Experiment**
  - ✓ Cooperative study of DOE and JNC in 1978～1988, using ZPPR facility at ANL, USA.
  - ✓ The largest FBR mockup experiment in history, 4,600 – 8,500 liters.
  - ✓ Various core concepts, sizes, and structures:
    - ◆ 600～800MWe-class two-region homogeneous cores,
    - ◆ 650MWe-class radially-heterogeneous cores,
    - ◆ 650MWe-class axially-heterogeneous cores,
    - ◆ and, 1000MWe-class homogeneous cores with enriched uranium regions.
  - ✓ Many kinds of measured parameters.

↓

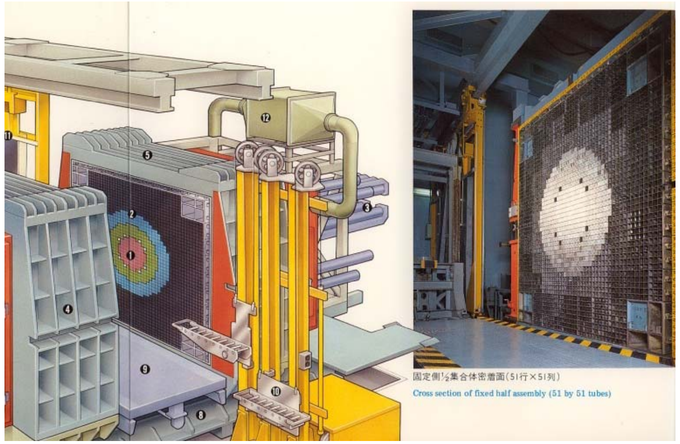
As-built experimental information is available for the public.

ZPPR Critical Assembly（ANL）

Integral Data for Fast Reactors (2/4)

FCA Critical Experiment

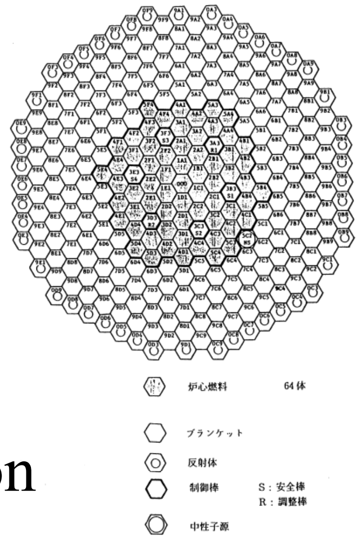
- ✓ Fast Critical Assembly at JAEA, Japan.
- ✓ To simulate small FBR cores with plutonium and enriched uranium fuels.
  - ◆ FCA XVII-1 Core (1993) -650 liters.
  - ◆ FCA X-1 Core (1982) -130 liters.



Experimental Reactor JOYO

- ✓ First Japanese FBR (1st Criticality in 1977) ーBurnup, pin-wrapper structure.
  - ◆ Mixed one-region plutonium and enriched uranium core with 240 liter-size.
  - ◆ Criticality, fuel-blanket replacement reactivity, and burnup reactivity were adopted.

As-built experimental information is available for the public.



JOYO Mk-I Core (JAEA)

(Minimum critical core)

Integral Data for Fast Reactors (3/4)

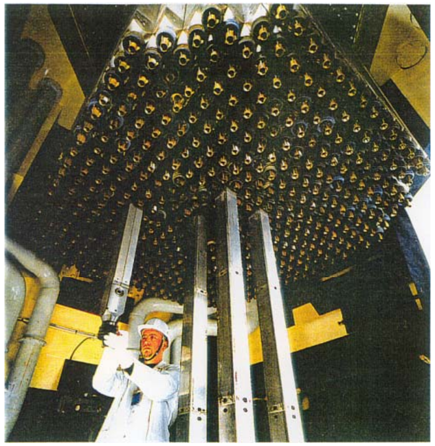
BFS-1, 2 Critical Experiment

- ✓ Fast Critical Assembly at IPPE, Obninsk, Russia.
  - ◆ BFS-58-1-I1 Core (1996) —Uranium-free region in core center and enriched uranium region in periphery.
  - ◆ BFS-62-1 ~5, 66-1 Core (1999 ~ 2002) —Three enriched uranium region core to study Pu-disposition in BN-600 core.
  - ◆ BFS-67, 69, 66-2 Core (1993 ~ 2003) —10 kg of NpO_2 loading cores in central MOX region with weapon-grade Pu, high enriched Pu, and degraded Pu.

BFS-2 Critical Assembly (IPPE)

MASURCA Critical Experiment

- ✓ Fast Critical Assembly at CEA, Cadarache, France.
  - ◆ ZONA2B Core (1996) —a 380 liter-size MOX fuel core with reflectors, which simulated Pu-burner.



MASURCA Critical Assembly (CEA)

Integral Data for Fast Reactors (4/4)

■ Los Alamos Small Core Experiment

- ✓ Sphere-shaped cores of approx. ten centimeter in diameter with metallic fuel consisted of Pu-239, or degraded Pu, or U-235.
  - ◆ FLATTOP-Pu, FLATTOP-25, JEZEBEL, JEZEBEL-Pu, GODIVA (1950s – early 60s)

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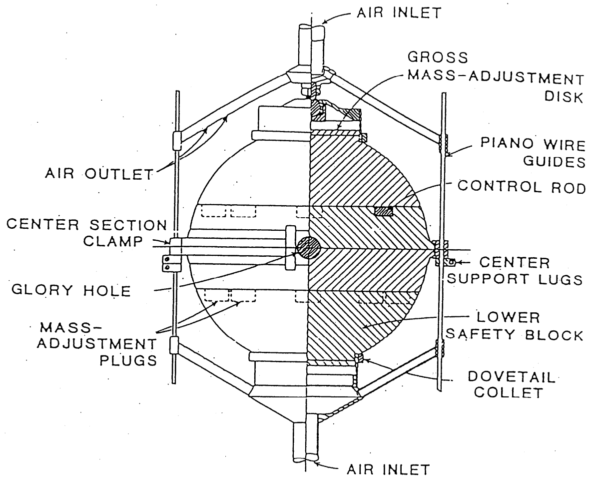
Benchmark models have already opened.



Jezebel under operating conditions. The nearly spherical assembly is supported by lightweight clamps and guides.

Radius 6.3cm
Mass 17kg Pu
LA – 9685 – H Unclassified

Los Alamos Small Core (JEZEBEL)

■ Other Experiments

- ✓ **ZEBRA** (MOZART program, 1971-73, UK) - a 560 liter-sized single-region and a 1,800 liter-sized two-region MOX cores .
- ✓ **SEFOR** (General Electric, 1969-72, USA) - a 20MWt fast power reactor core fueled with mixed PuO_2-UO_2 and cooled with sodium.
- ✓ **Monju** (JAEA, 1st Criticality in 1994, Japan) - a 280MWe FBR with MOX fuel.

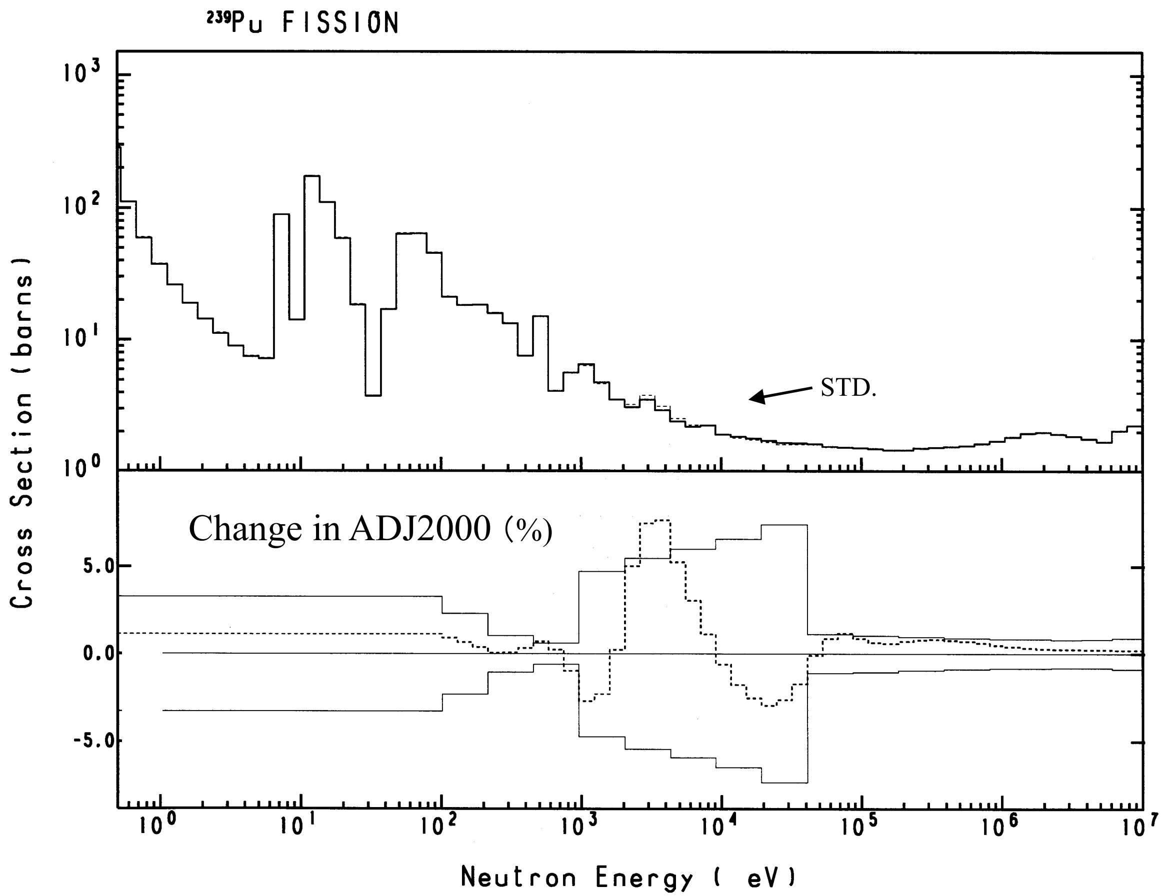
Determination of Experimental and Analytical Uncertainties

- Experimental uncertainty
  - Follows the evaluation by experimenters like ANL.
- Analytical modeling uncertainty
  - Assumes it is proportional to the sensitivity against the degree of modeling detail,
  - Absolute value was decided to make the ratio of the chi-square value to the freedom approx. unity.
- Elimination of abnormal data
  - Excludes if the deviation of C/E value from unity is three times larger than the total uncertainty value.

(confidence level : 1σ)

Core Parameter		Experimental uncertainty	Analytical Modeling uncertainty
Criti-cality	JUPITER, FCA, etc.	0.04%	0.17%
	Los Alamos	0.1～0.18%	0.15%
F28/F49 Ratio		2.5%	1.1%
F25/F49、C28/F49 Ratio		2.2%	0.55%
F49 Distribution		1.0%	0.6～1.2%
Control Rod Worth		1.2%	1.3%
Sodium Void Reactivity		2%	5.5～8.8%
Doppler Reactivity		2.0～3.0%	5.0～6.6%

Change of Nuclear Data
- Pu-239 Fission -



- In the simultaneous evaluation region, the error values are small.
- On the other hand, the errors are large in the resonance region.

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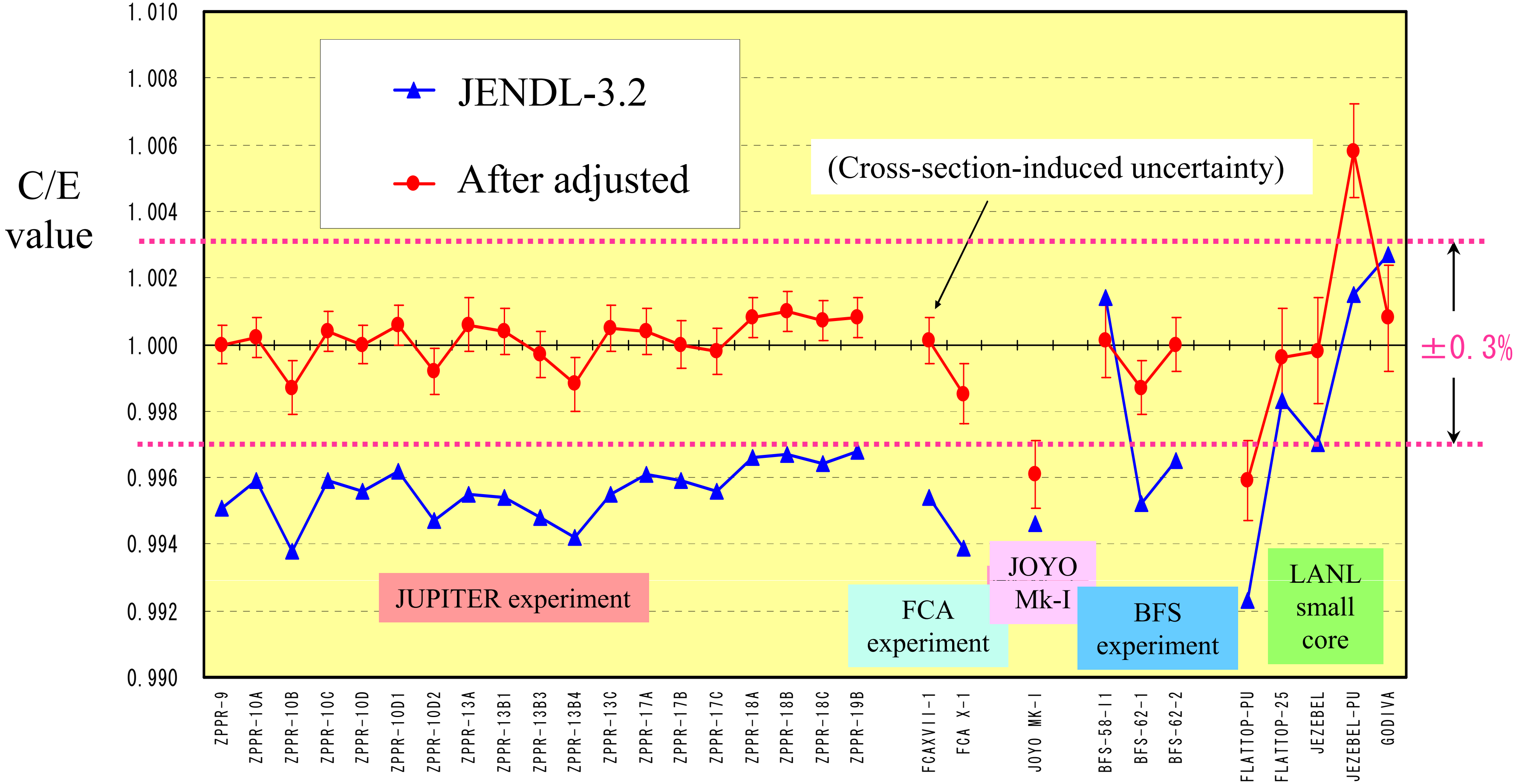
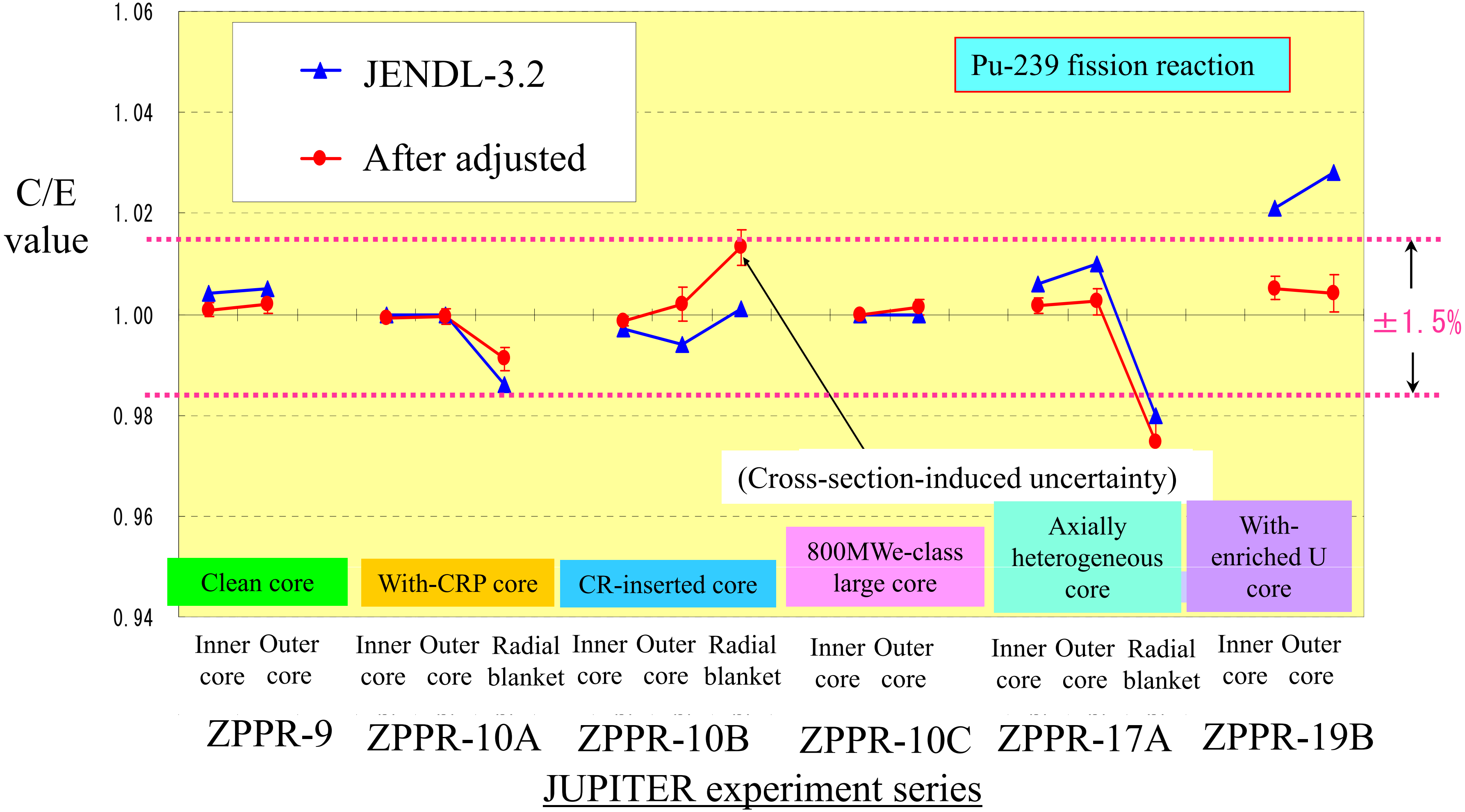
- Change of the fission cross-section by adjustment follows such energy dependency.

Analytical Results (1/4) - Criticality -



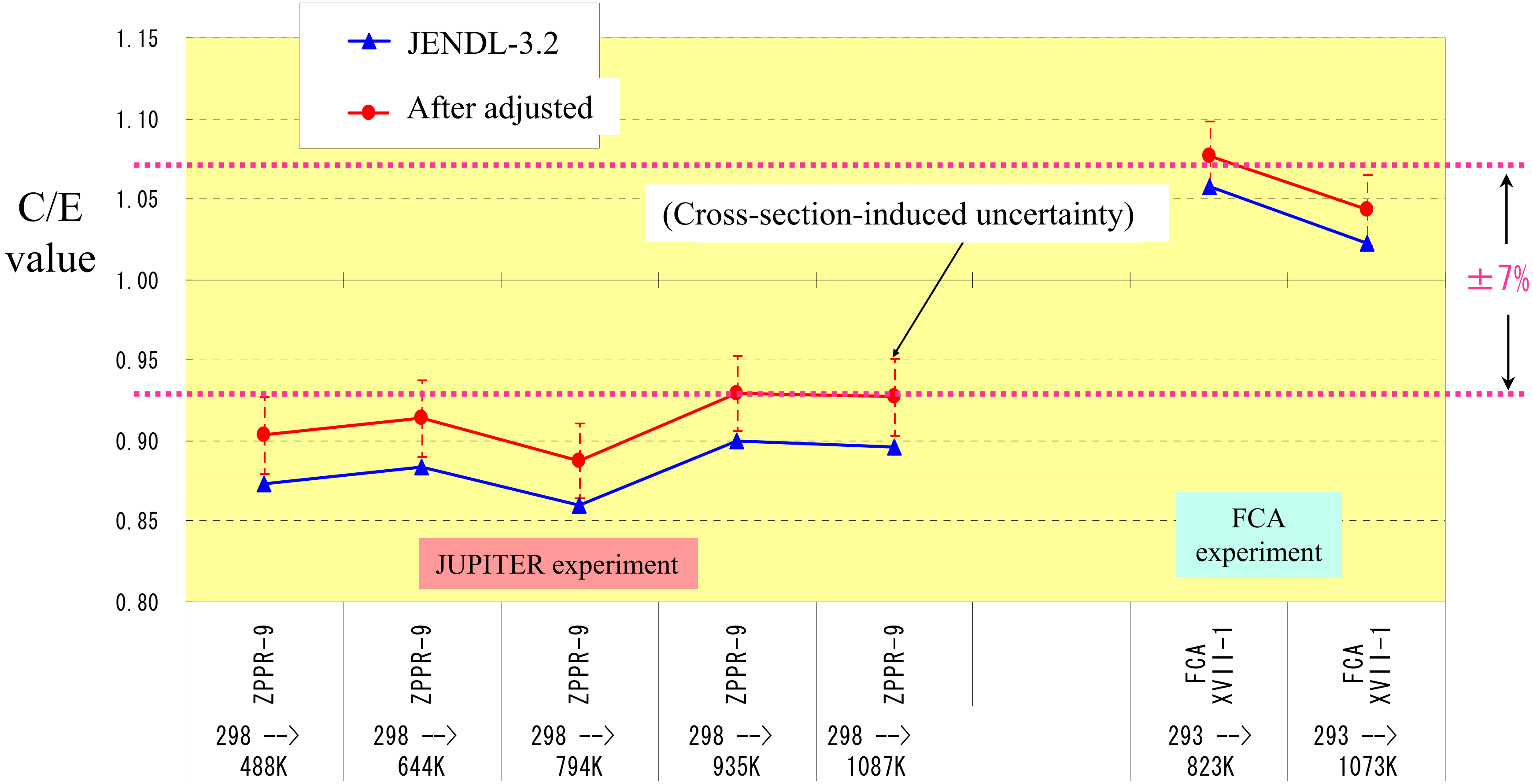
- The C/E values of criticality after adjusted are within ±0.3% Δk, except several small cores.
- The good performance is not only for Pu-fuel cores, but enriched-U fuel cores.

Analytical Results (2/4) - Power Distribution -



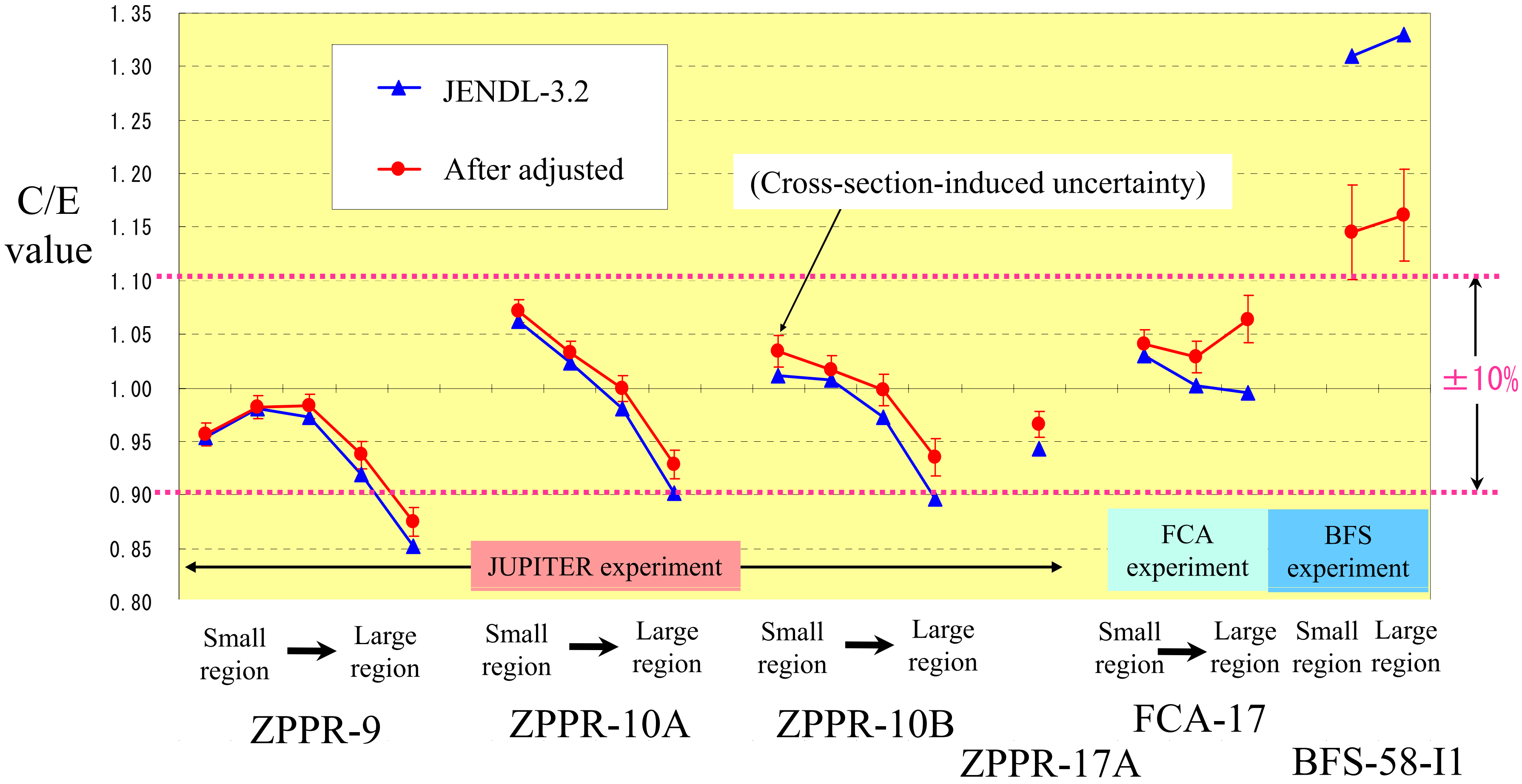
- The C/E values of reaction rate distribution after adjusted are sufficiently smaller than ±1.5% in the core fuel region.
- There is room for improvement for the blanket region.

Analytical Results (3/4) - Doppler Reactivity -



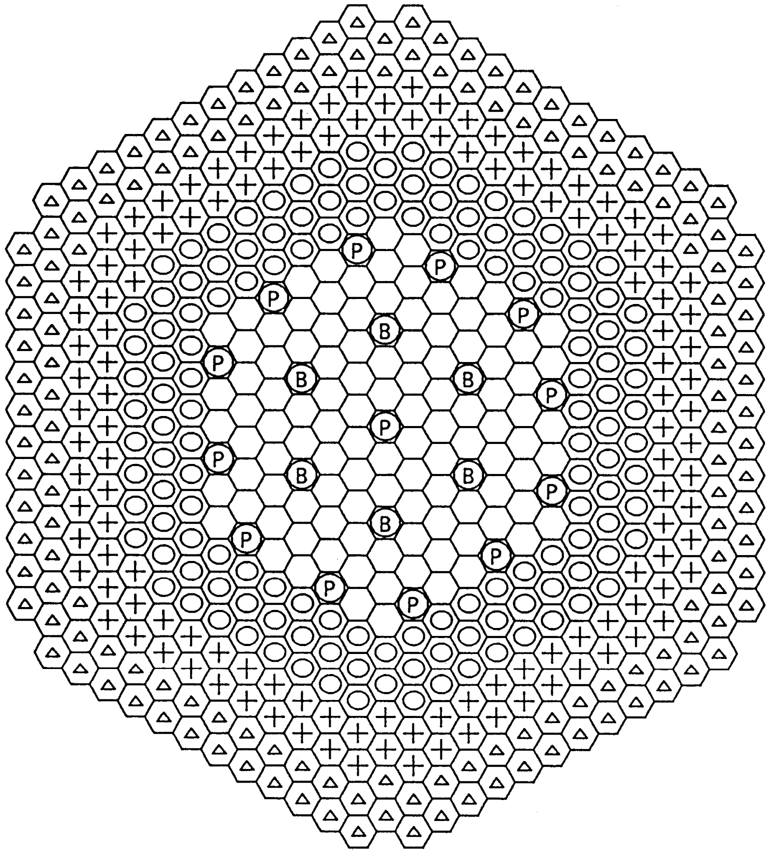
■ It seems Re-investigation is needed for the accuracy of sample Doppler reactivity measurements.

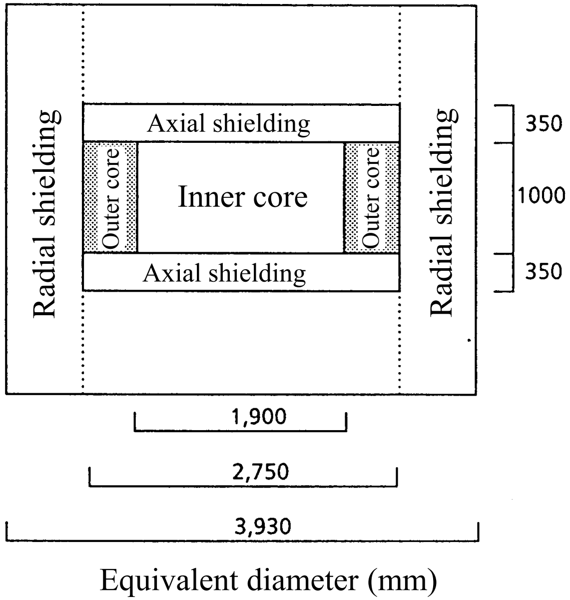
Analytical Results (4/4) - Sodium Void Reactivity -



- The effect of adjustment is small for JUPITER and FCA experiment, and the C/E values are within app. ±10%.
- The discrepancy of C/E values from 1.0 may be caused by something, besides nuclear data.

A 600 MWe-class FBR Core





Inner core	108
Outer core	138
Stainless steel	126
B_4C shield	150
Primary control rod	13
Backup control rod	6
Total	541

Core structure of a 600 MWe-class FBR

Nuclear Design Accuracy of a 600 MWe-class FBR

*1σ value (Non-diagonal terms show correlation factors.)

Design method / Nuclear parameter	No use of integral information Total : % unit. (( ) : contribution from nuclear data covariances.)					
	Criticali-ty	Breeding ratio	Power distribution	Doppler reactivity	Na void Reactivity	Burnup reactivity loss
Criticality (End of equilibrium cycle)	0.81 (0.78) %					
Breeding ratio (C28/F49 reaction rate ratio)	-0.54	1.5 (1.2) %			(Symmetry matrix)	
Power distribution (Outer core region)	-0.46	0.49	1.4 (1.1) %			
Doppler reactivity (Whole core region)	-0.54	0.36	0.34	4.0 (3.4) %		
Na void Reactivity (Whole core region)	-0.02	0.05	0.06	0.12	6.1 (3.7) %	
Burnup reactivity loss	0.24	-0.56	-0.32	0.05	-0.11	5.6 (5.4) %

Design Accuracy Improvement of a 600 MWe-class FBR

(*1σ value)

Design method / Nuclear parameter	No use of integral information	E/C-bias method	Cross-section adjustment method
Criticality (End of equilibrium cycle)	0.81	0.41	0.23
Breeding ratio (C28/F49 reaction rate ratio)	1.5	2.3	1.0
Power distribution (Outer core region)	1.4	1.6	1.0
Doppler reactivity (Whole core region)	4.0	4.9	2.6
Na void Reactivity (Whole core region)	6.1	6.5	4.9
Burnup reactivity loss	5.6	7.5	3.6

Concluding Remarks

- JAEA has performed the nuclear data adjustment study for 20 years, continuously. As a result, a unified cross-section set, **ADJ2000**, was developed, the features of which are, JENDL-3.2 base, adjustment of self-shielding factors, application of the latest cross-section covariance, experiments of wide-variety cores, adoption of burnup characteristics and Doppler reactivity.
- ADJ2000 can predict **wide-variety cores** with high accuracy, from large to small size cores, and from critical experiments to power reactor, for **various core parameters**, being used in the future FBR development project.

The next adjusted set, **ADJ2010**, is quite soon expected with **JENDL-4** and associated **covariance data**.