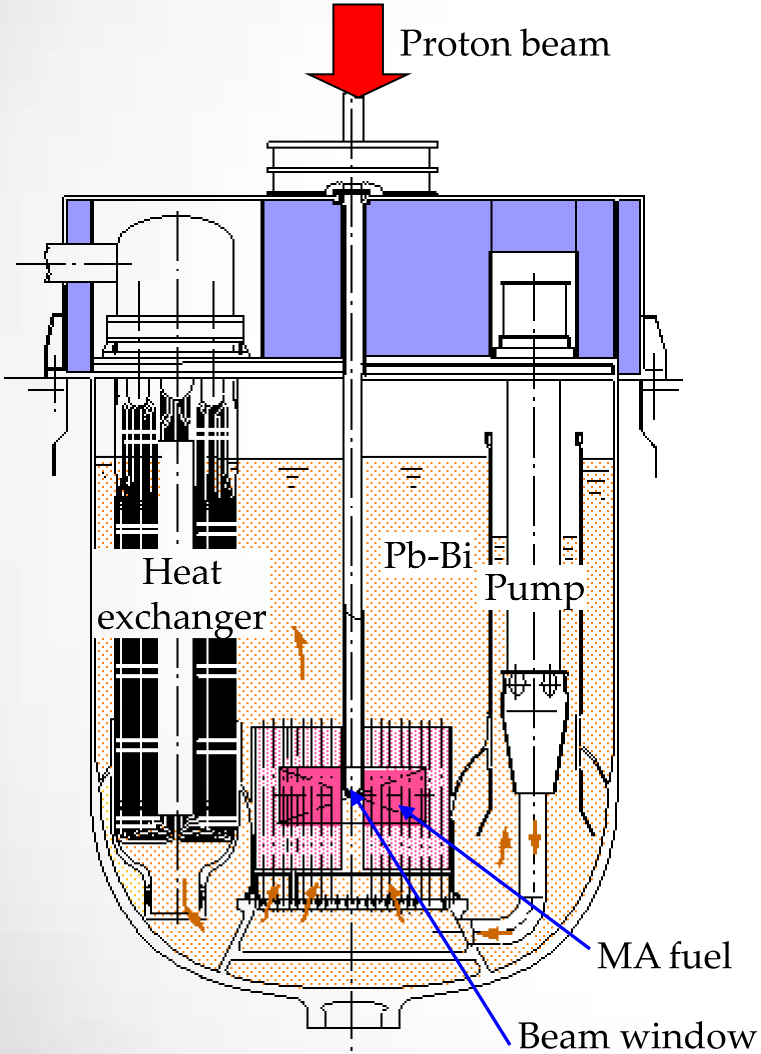
Investigation of Nuclear Data Accuracy for the Accelerator-Driven System with Minor Actinide Fuel

Kenji Nishihara, Takanori Sugawara, Hiroki Iwamoto
JAEA, Japan
Francisco Alvarez Velarde
CIEMAT, Spain
Andrei Rineiski
KIT, Germany

Contents

- IAEA CRP benchmark activity
  - Depletion analysis for 800 MWt ADS (commercial grade)
  - Result by several nuclear libraries
- Uncertainty analysis using covariance matrix
  - Estimation of uncertainty by JENDL-3.3 and JENDL-4.0
  - Reduction of uncertainty by the Transmutation Physics Experimental Facility, TEF-P.

IAEA-CRP benchmark



- LBE cooled tank type reactor
- 800 MW (thermal power)
- 600 day/cycle
- Proton beam: 1.5GeV, 10-20mA
- LBE target with window
- Nitride fuel, (Pu+MA)N+ZrN
  - Pu-N/MA-N/Zr-N=19/31/50 (weight percent)
  - Total actinide amounts 4.2 ton
- Transmutation: 500kg/cycle (12%/cycle)

A. Stanculescu, "The IAEA Coordinated Research Project (CRP) on "Analytical and experimental benchmark analyses of accelerator driven systems," OECD/NEA 11th Information Exchange Meeting, San Francisco, 1-4 Nov. 2010. (2010)

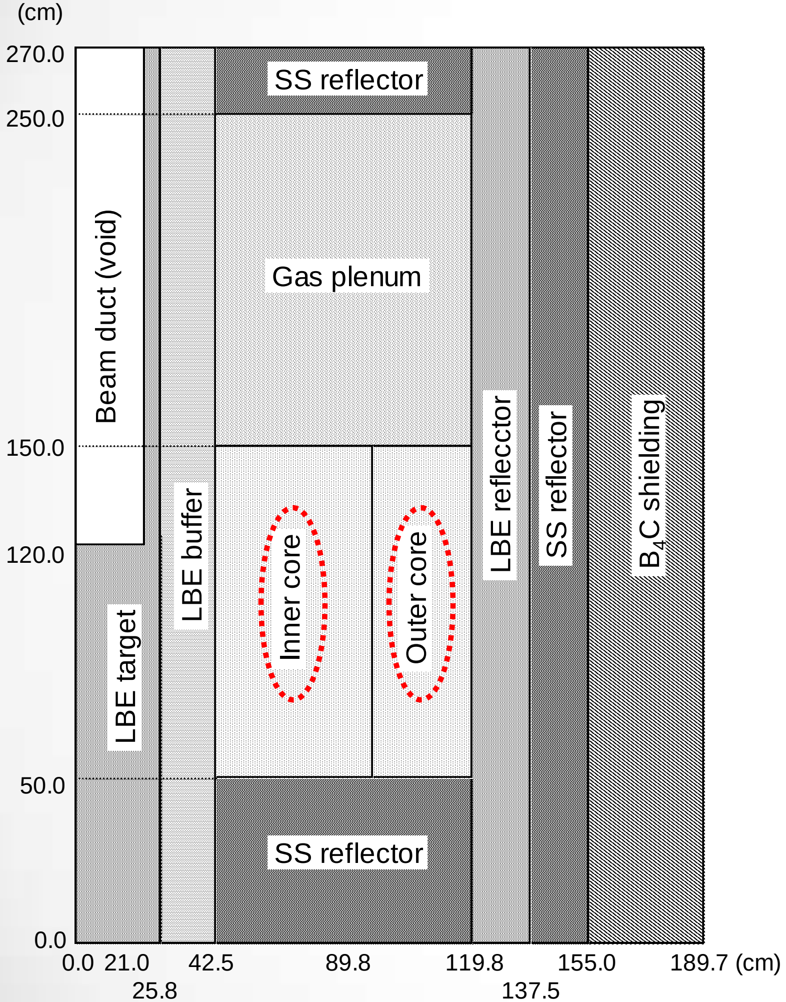
Feature of this benchmark

- Commercial grade ADS
  - High energy proton (1.5GeV)
  - MA-rich fuel
  - Burn-up up to 120 GWd/HMt
- Simplified R-Z geometry

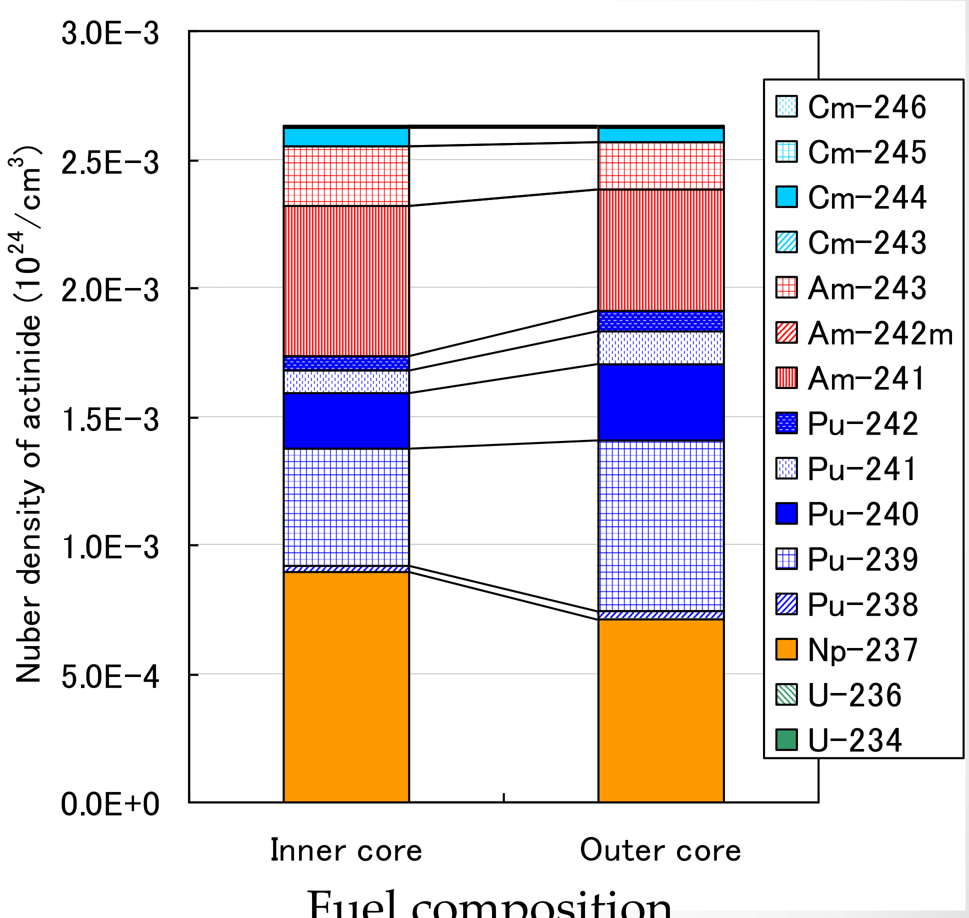
→ Fundamental knowledge of calculation accuracy at the present time for the commercial grade ADS

- Main issues for the calculation accuracy are,
  - Simulation of high energy reactions
  - Nuclear data for MA

Calculation model



R-Z model



Fuel composition

K.Tsujimoto, *et.al*, *J. Nucl. Sci. Tech.*, **41**(1), 21 (2004).

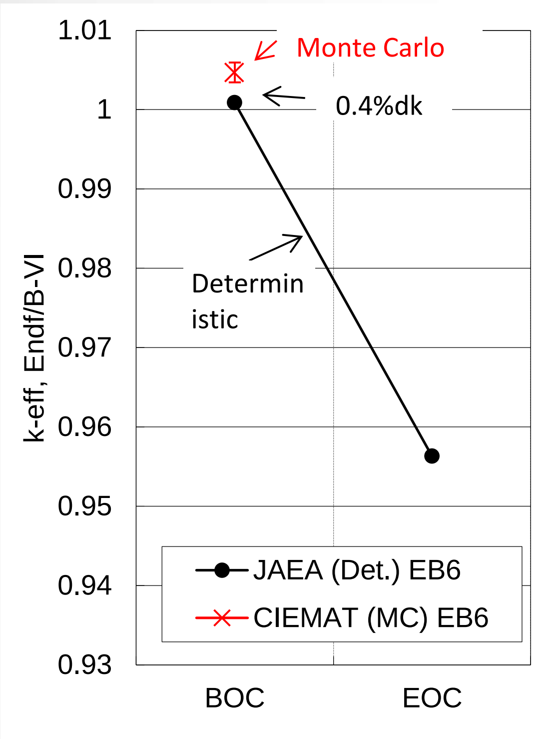
Participants

Participant	Code	Library
JAEA (Japan)	**PHITS, NMTC** or **MCNPX** (MC code for proton and neutron >20MeV) **SLAROM** (Cross section code) **TWODANT** (Deterministic neutron transport code) **ORIGEN** (Burn-up code)	JENDL-3.3 JENDL-4.0 JENDL-3.2 ENDF/B-VI JEFF-3.0
CIEMAT (Spain)	**EVOLCODE2** (MCNPX-based burnup code)	JEFF-3.0 (JEFF-3.1) [a] (ENDF/B-VI) [a]
KIT (Germany)	High energy particles are not analyzed. **C4P, ZMIX** (Cross section code) **DANTSYS** (Deterministic neutron tansport code) **TRAIN** (Burn-up code)	ENDF/B-VII JEFF-3.1

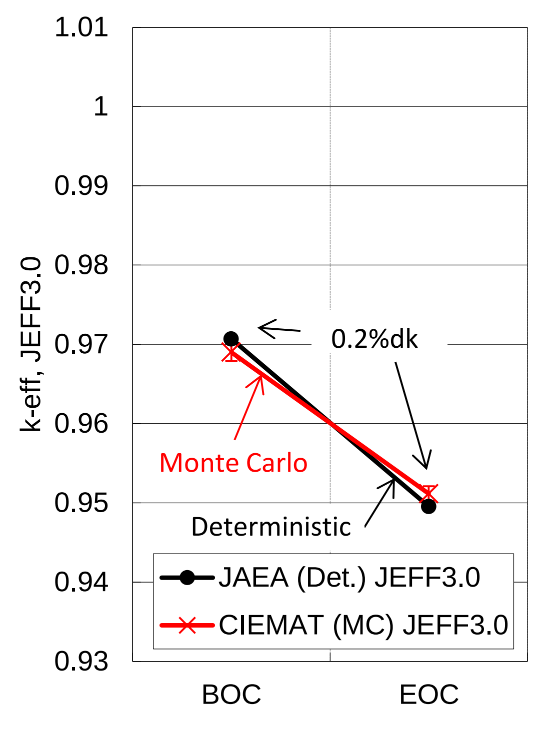
[a] Library in parenthesis is only for the beginning of cycle (BOC).

Deterministic vs. Monte-Carlo
JENDL vs. ENDF vs. JEFF
PHITS vs. MCNPX for high energy

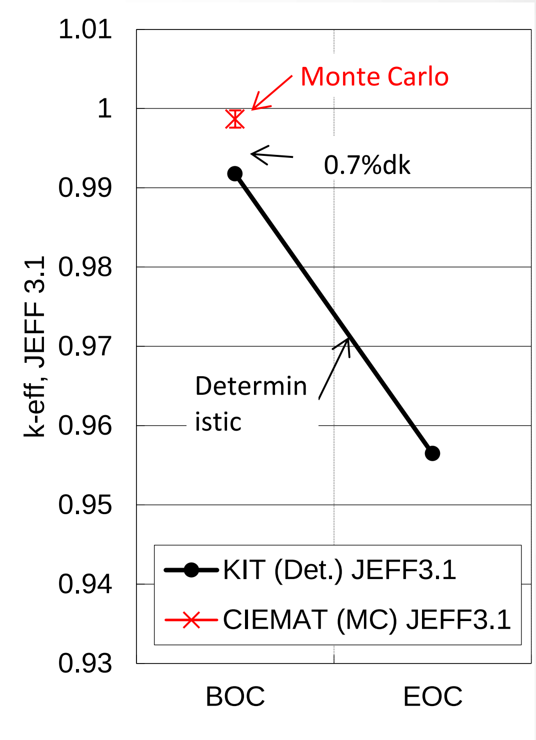
Deterministic vs. Monte-Carlo



ENDF/B-VI



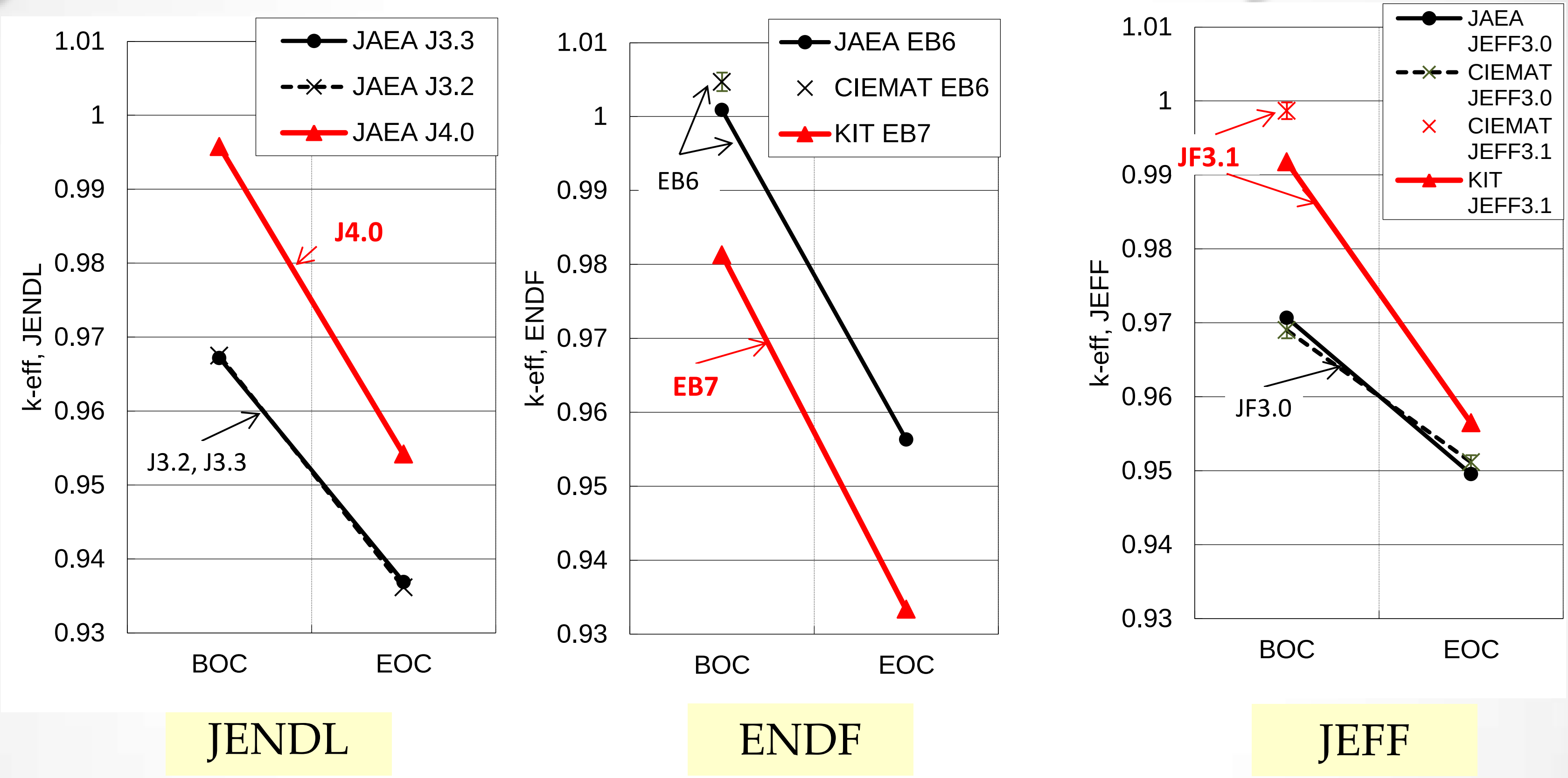
JEFF 3.0



JEFF 3.1

Good agreement is observed (0.2-0.7 %dk).
Influence of calculation methods is small.

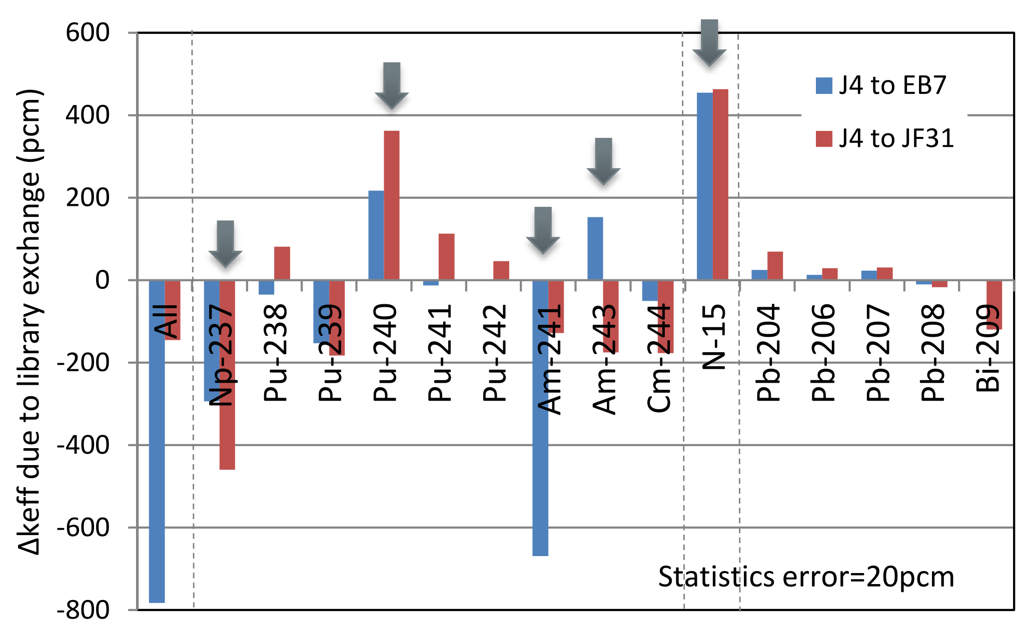
JENDL vs. ENDF vs. JEFF

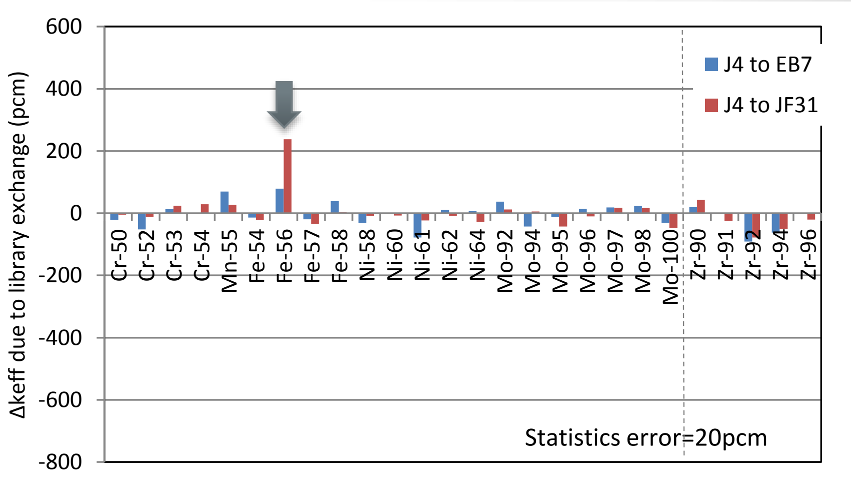


k-effective disperse from 0.98 to 1.0 at BOC and 0.93 to 0.96 at EOC.

Version-up changes k-effective by 2 to 3 %dk for each library.

Which nuclide is responsible at BOC?

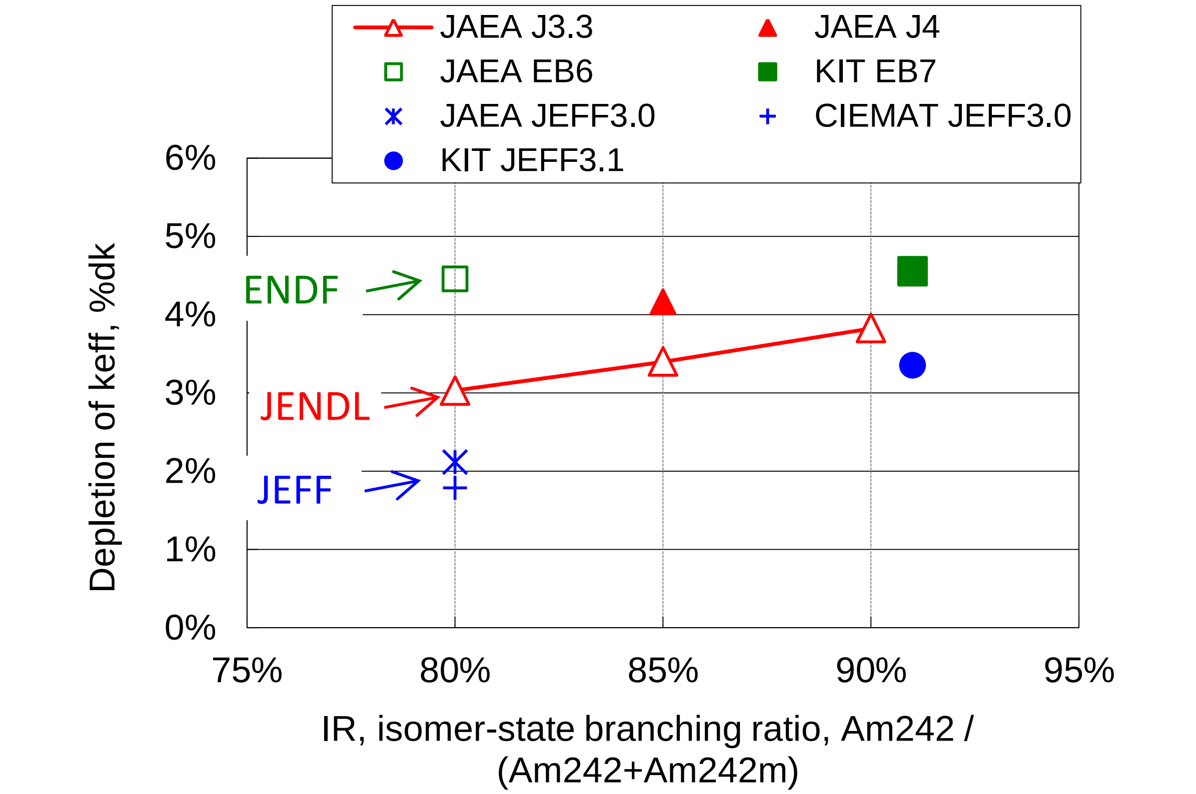




Δkeff due to library exchange of single nuclide from JENDL-4 to ENDF/B-VII or JEFF-3.1 estimated by MCNPX.

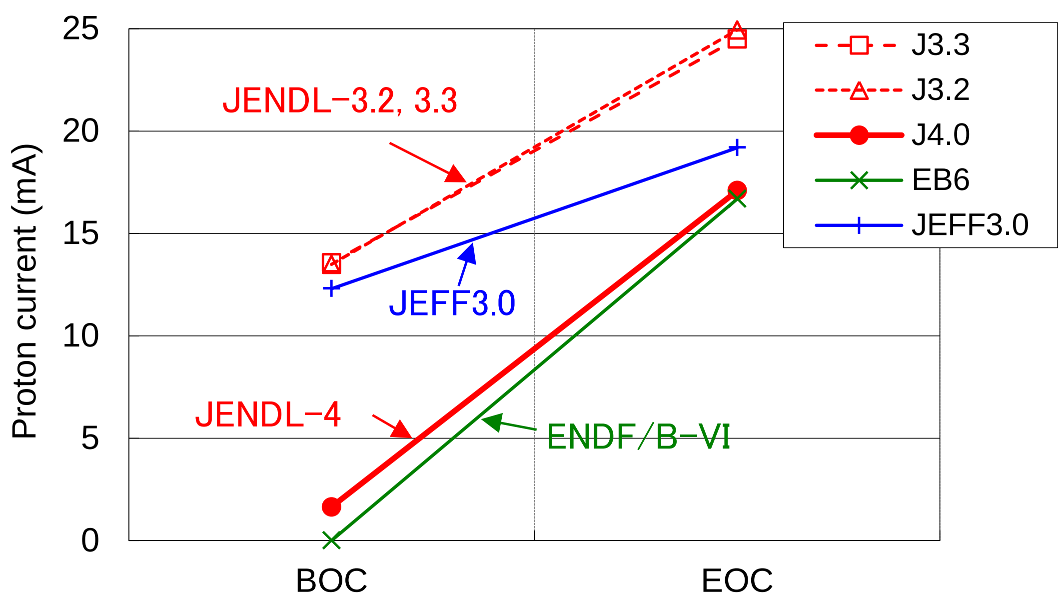
Large changes are observed at Np-237, Pu-240, Am-241, Am-243, N-15 and Fe-56. Sensitivity analysis is needed in the future.

Effect of isomer-state branching ratio (IR) of Am-241 capture reaction



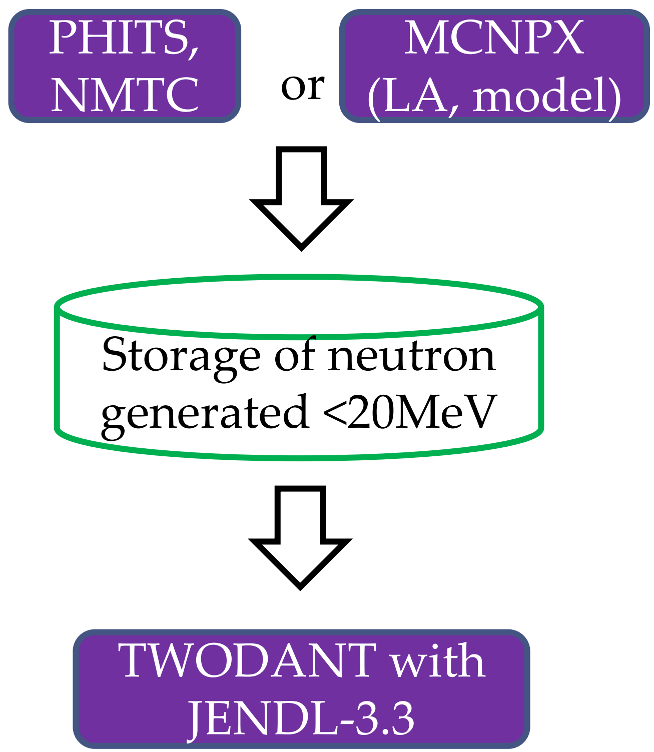
- The depletion of keff becomes larger by 0.8%dk if the IR changes from 80% to 90%.
- JEFF-3.1 library gives smaller depletions than JENDL-4 and ENDF/B-VII.

Proton beam current



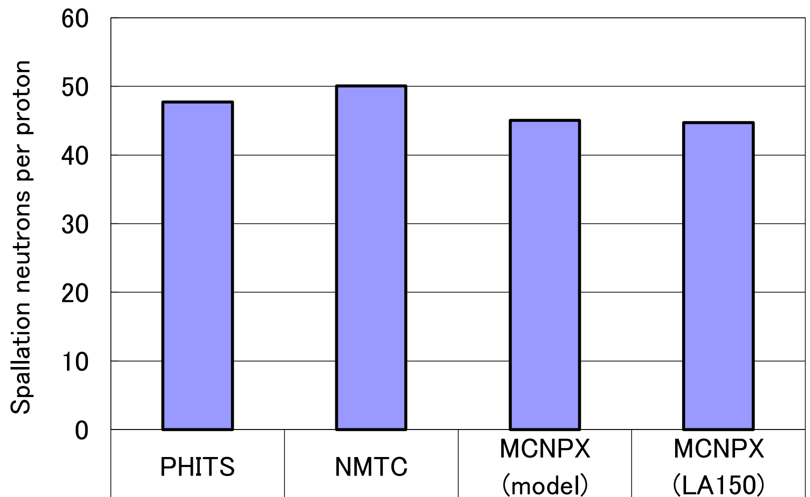
Criticality affects too much on proton beam current to make comparison.

Effect of high-energy particle calculation on current



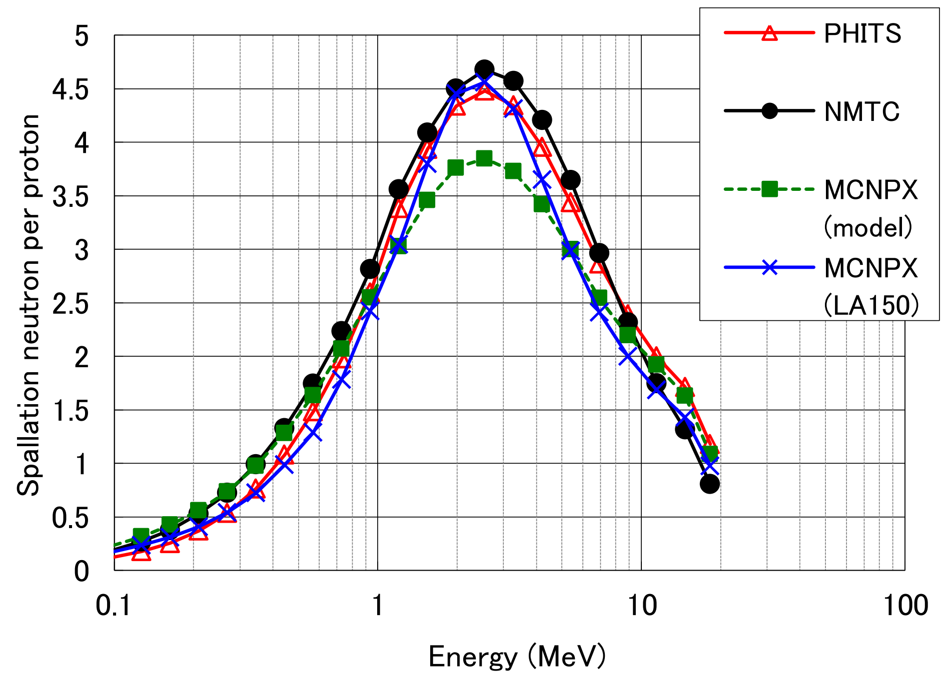
Comparison between high-energy codes was done with common low-energy code and library.

Neutron produced at energy below 20 MeV



	PHITS	NMTC	MCNPX (model)	MCNPX (LA150)
N/P	47.72	50.06	45.04	44.71
MeV	4.10	3.70	3.95	3.88

Number of neutron and average energy（MeV)

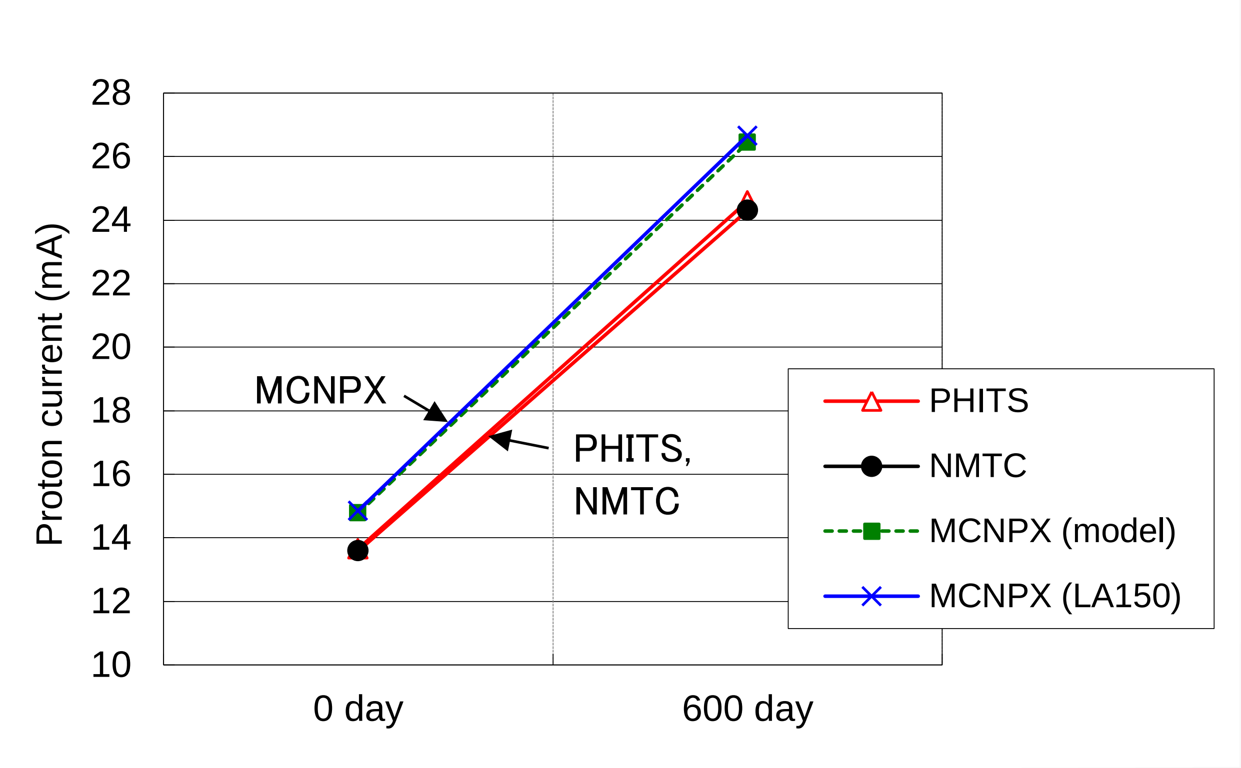


Spectrum of spallation neutron

- Number of neutron : PHITS > MCNPX
- Average energy of neutron : PHITS > MCNPX

→ PHITS results in 8% larger efficiency of source proton.

Result of beam current



Proton beam current based on JENDL-3.3

The difference of current is 8% between MCNPX and others.

Summary of IAEA-CRP benchmark

- Small difference between participants (codes) with same library.
- k-effective disperse from 0.98 to 1.0 at BOC and 0.93 to 0.96 at EOC.
  - At BOC, Np-237, Pu-240, Am-241, Am-243, N-15 and Fe-56 affect.
- Beam current is not comparable due to different k-effective.
  - Based on JENDL-3.3, current differs by 8% between MCNPX and PHITS.

To design an ADS is very difficult, if estimated k-effective changes 2 to 3 %dk.

→ Uncertainty and its reduction by ctritical experiments should be evaluated.

Uncertainty evaluation

- The error caused by the nuclear data is calculated as square root of **GMG^t**.
- Sensitivity coefficient **G**
  - **SAGEP code** gives sensitivity of cross sections on: criticality, coolant void reactivity and Doppler coefficient
- Covariance data **M**
  - Covariance data are presented in JENDL-3.3 and JENDL-4.0.
  - Provisional values were employed for nuclides and reactions which were not prepared in JENDL.

Covariance data in JENDL-3.3

Nuclide	Capture	Fission	ν	Elastic	Inelastic	χ	μ-bar	Nuclide	Capture	Fission	ν	Elastic	Inelastic	χ	μ-bar
Pu-238	○	○	△	△	△			N-15	□	–	–	○	□	–	
Pu-239	○	○	○	○	○	○	○	Fe	○	–	–	○	○	–	○
Pu-240	○	○	○	○	○	○	○	Cr	○	–	–	○	○	–	○
Pu-241	○	○	○	○	○		○	Ni	○	–	–	○	○	–	○
Pu-242	○	○	△	△	△			Zr-90	○	–	–	□	○	–	
Np-237	○	○	○	△	△			Zr-91	□	–	–	□	□	–	
Am-241	○	○	○	△	△			Zr-92	□	–	–	□	□	–	
Am-242m	○	○	△	△	△			Zr-94	□	–	–	□	□	–	
Am-243	○	○	○	△	△			Zr-96	□	–	–	□	□	–	
Cm-242	△	△	△	△	△			Pb-204	□	–	–	□	□	–	
Cm-243	△	△	△	△	△			Pb-206	□	–	–	□	○	–	
Cm-244	○	○	△	△	△			Pb-207	□	–	–	□	○	–	
Cm-245	△	△	△	△	△			Pb-208	□	–	–	□	○	–	
Cm-246	△	△	△	△	△			Bi-209	□	–	–	□	○	–	

Most of reactions for MA, Zr and Pb are missing in the JENDL-3.3.
They are supplemented with provisional estimations.

○: Formally included
△ : Provisional estimation, Nakagawa (2007)
□ : Provisional estimation, Shibata (2007)

Covariance data in JENDL-4.0

Nuclide	Capture	Fission	ν	Elastic	Inelastic	χ	μ-bar	Nuclide	Capture	Fission	ν	Elastic	Inelastic	χ	μ-bar
Pu-238	○	○	○	○	○	○	○	N-15		-	-	○		-	
Pu-239	○	○	○	○	○	○	○	Fe-56	○	-	-	○	○	-	○
Pu-240	○	○	○	○	○	○	○	Cr-52	○	-	-	○	○	-	○
Pu-241	○	○	○	○	○	○	○	Ni-58	○	-	-	○	○	-	○
Pu-242	○	○	○	○	○	○	○	Zr-90		-	-			-	
Np-237	○	○	○	○	○	○	○	Zr-91		-	-			-	
Am-241	○	○	○	○	○	○	○	Zr-92		-	-			-	
Am-242m	○	○	○	○	○	○	○	Zr-94		-	-			-	
Am-243	○	○	○	○	○	○	○	Zr-96		-	-			-	
Cm-242	○	○	○	○	○	○	○	Pb-204		-	-			-	
Cm-243	○	○	○	○	○	○	○	Pb-206		-	-			-	
Cm-244	○	○	○	○	○	○	○	Pb-207		-	-			-	
Cm-245	○	○	○	○	○	○	○	Pb-208		-	-			-	
Cm-246	○	○	○	○	○	○	○	Bi-209		-	-		○	-	

Lack

All of the actinide reactions are covered in the JENDL-4.0!

But, data for N-15, Zr and Pb are needed. (Provisional data in the JENDL-3.3 are used in the present study.)

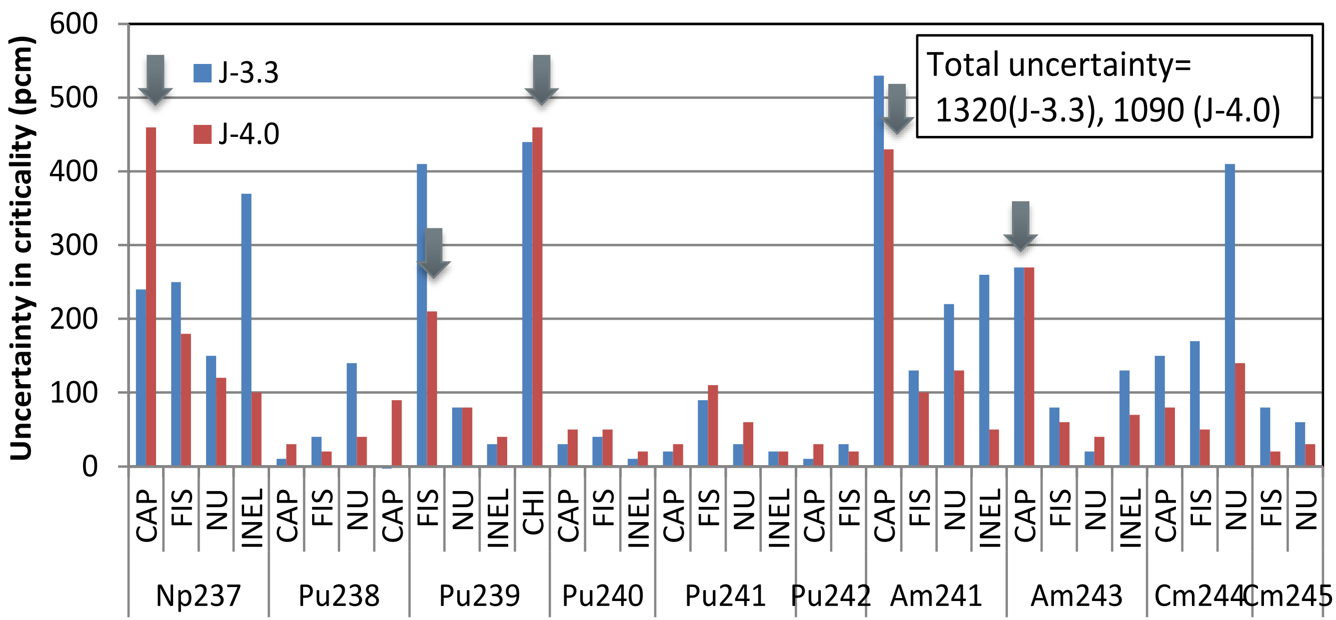
Uncertainty of JENDL-3.3 and 4.0

	JENDL-3.3		JENDL-4.0	
	Value	$(GMG^t)^{0.5}$	Value	$(GMG^t)^{0.5}$
Criticality (k-eff)	0.971	**1320 pcm** (1.4%)	0.999	**1090 pcm** (1.1%)
Coolant void reactivity [Δk/k]	5300 pcm	350 pcm (6.6%)	3500 pcm	280 pcm (8.1%)
Doppler reactivity coefficient [TΔk/dT]	-3.5e-4	7.0%	-3.3e-4	6.3%

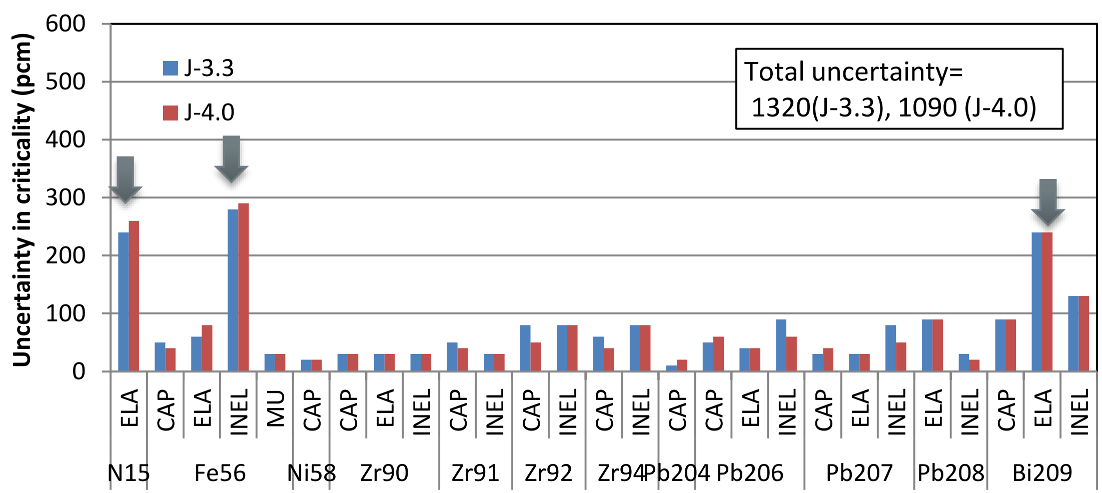
Uncertainties evaluated by JENDL-4.0 are equal or less than those by JENDL-3.3.

The uncertainty of criticality, 1090 pcm, is smaller than the result of IAEA benchmark activity (2000-3000pcm).

Breakdown of uncertainty (criticality)

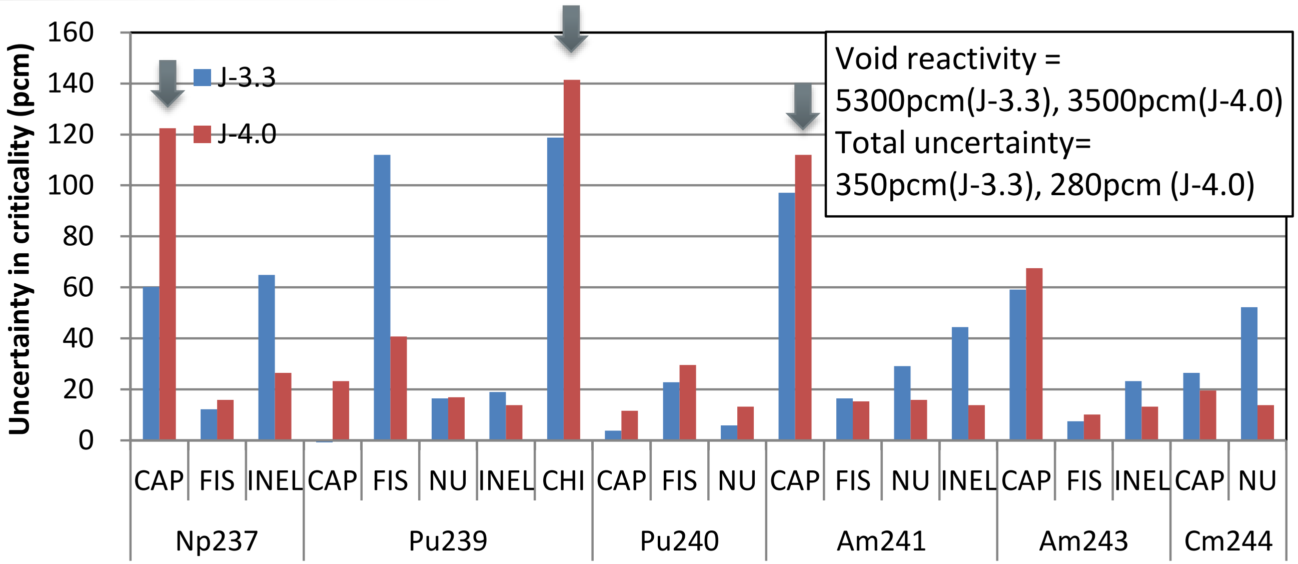


Np-237 capture
Pu-239 fission and chi
Am-241 capture
Am-243 capture



N-15 elastic
Fe-56 inelastic
Bi-209 elastic

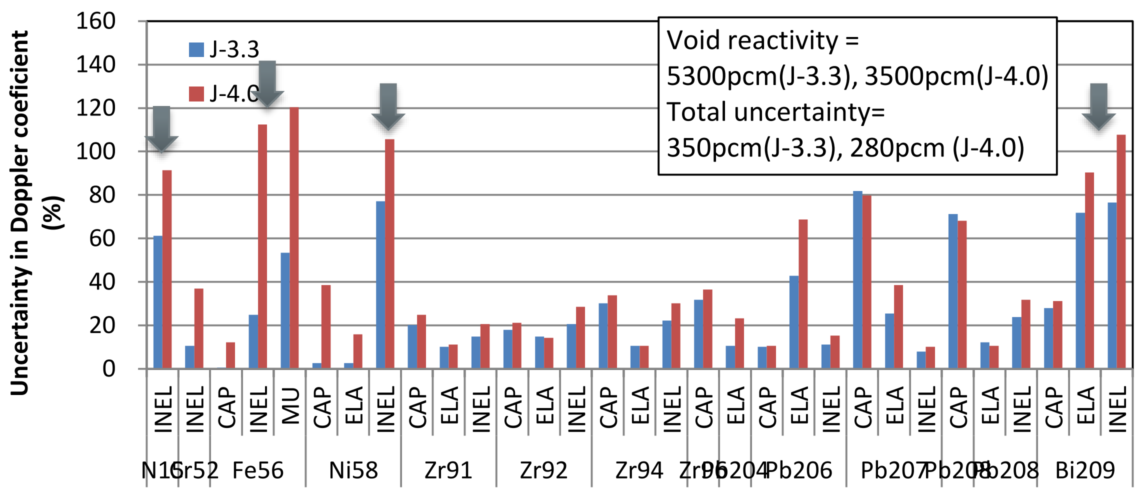
Breakdown of uncertainty (void reactivity)



Np-237 capture

Pu-239 chi

Am-241 capture



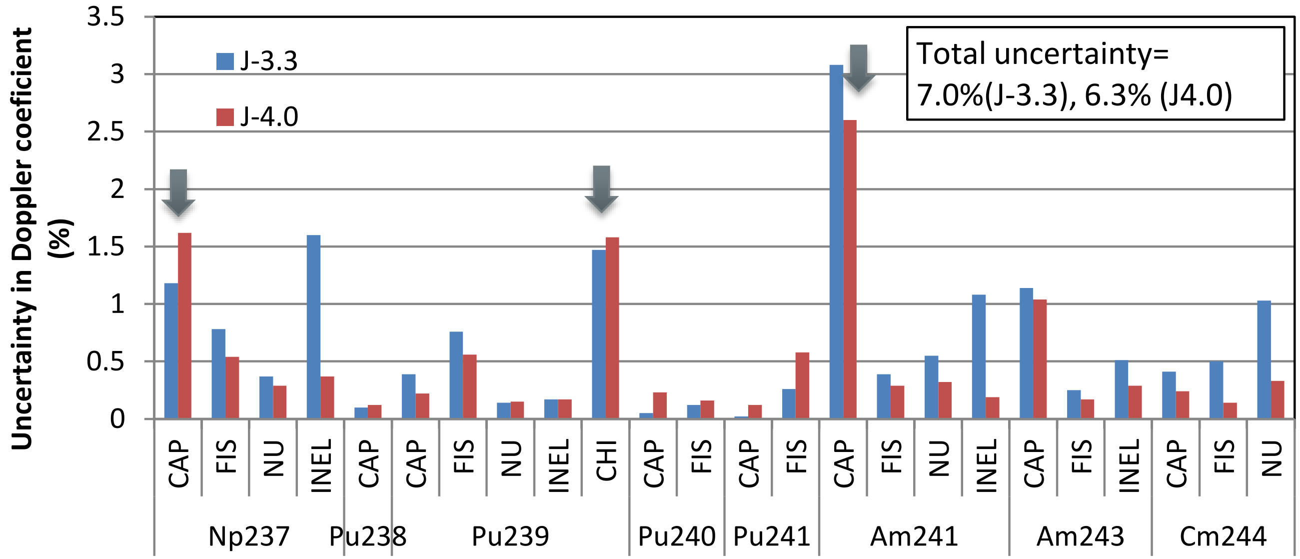
N-15 elastic

Fe-56 inelastic, mu

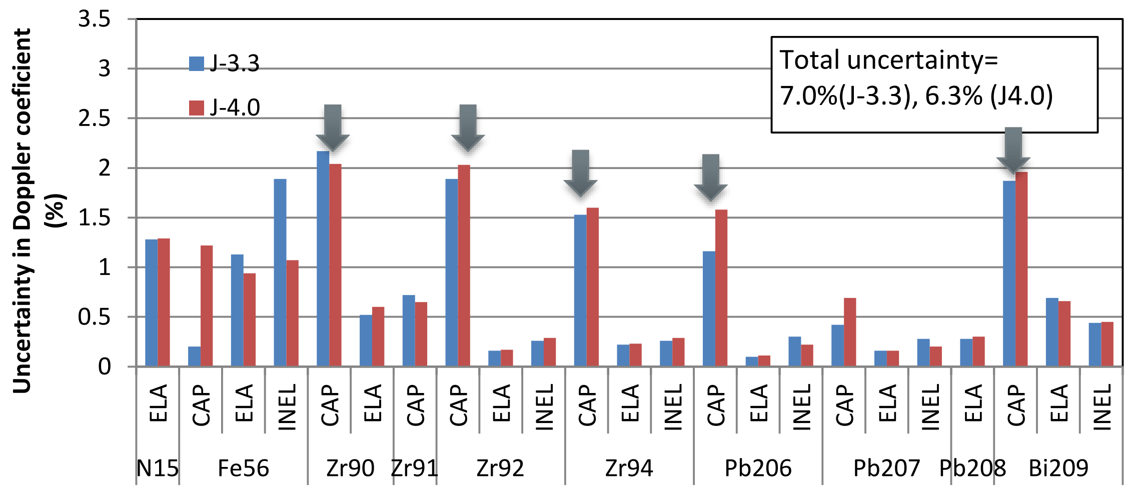
Ni-58 inelastic

Bi-209 elastic and inelastic

Breakdown of uncertainty (Doppler)

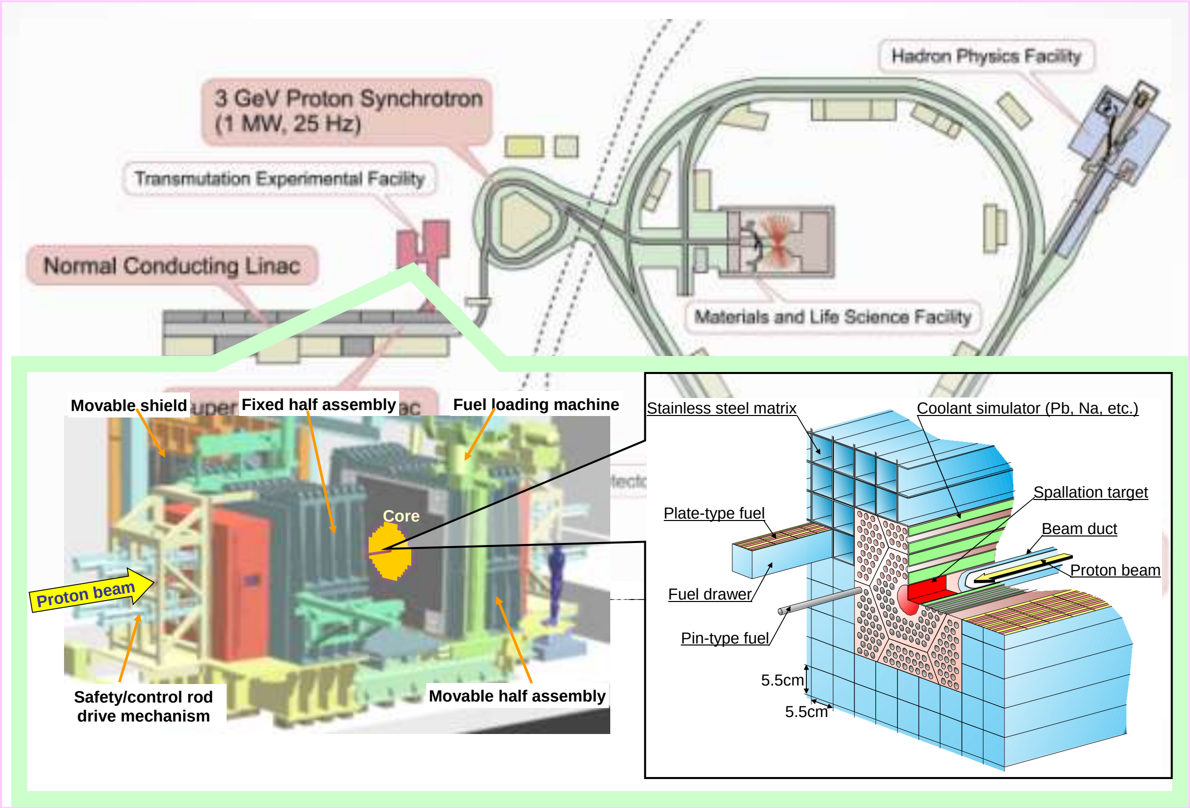


Np-237 capture
Pu-239 chi
Am-241 capture



Zr-90 capture
Zr-92 capture
Zr-94 capture
Pb-206 capture
Bi-209 capture

Reduction of uncertainty by TEF-P



Three calculation cases were performed to confirm effect of MA amount on the uncertainty reduction :

CASE-mg: Measurement of the **reaction rate ratio** by using mg-order MA sample

CASE-g: Measurement of the MA **sample worth** by using g-order MA sample

CASE-kg: Various experiments by using kg-order MA to **simulate spectrum** dependences.

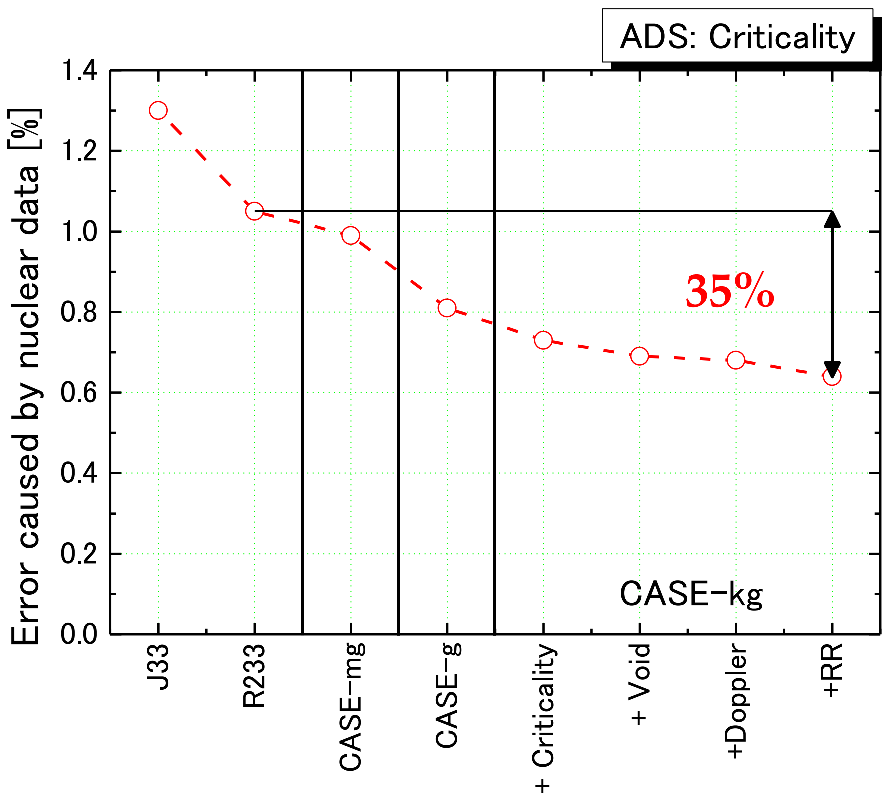
Experiments in CASE-kg

- Experiment descriptions of case-kg
  - Criticality: 1 case
  - Coolant void reactivity : 3 cases (3 different void fractions)
  - Doppler reactivity : 3 cases (300℃, 550℃, 800℃)
  - Measurement of reaction rate ratio : 8 cases

    (Denominator: Pu-239 fission,

    Numerator: Fission and Capture for 4 MA nuclides, Np-237, Am-241, Am-243 and Cm-244)
  - **Total 15 cases**

Calculation Results (1/2)



J33: Result by using JENDL-3.3

R233: Adjusted by existing 233 integral data*1

(→ It is able to consider R233 result as the current best value)

'+Criticality': Adjusted by CASE-g results and criticality experiment in CASE-kg.

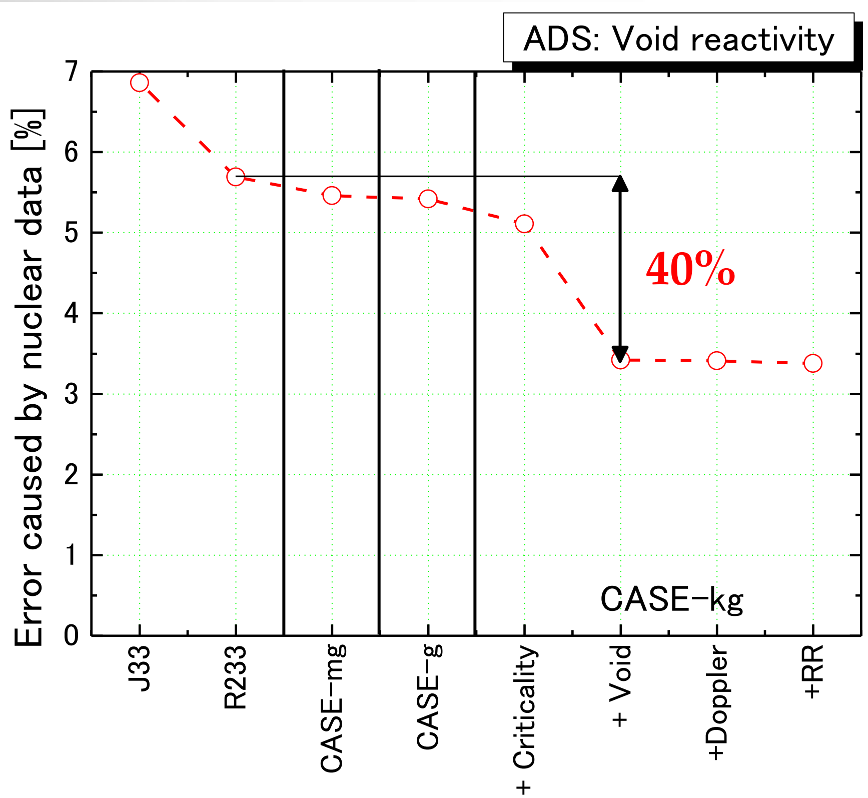
…

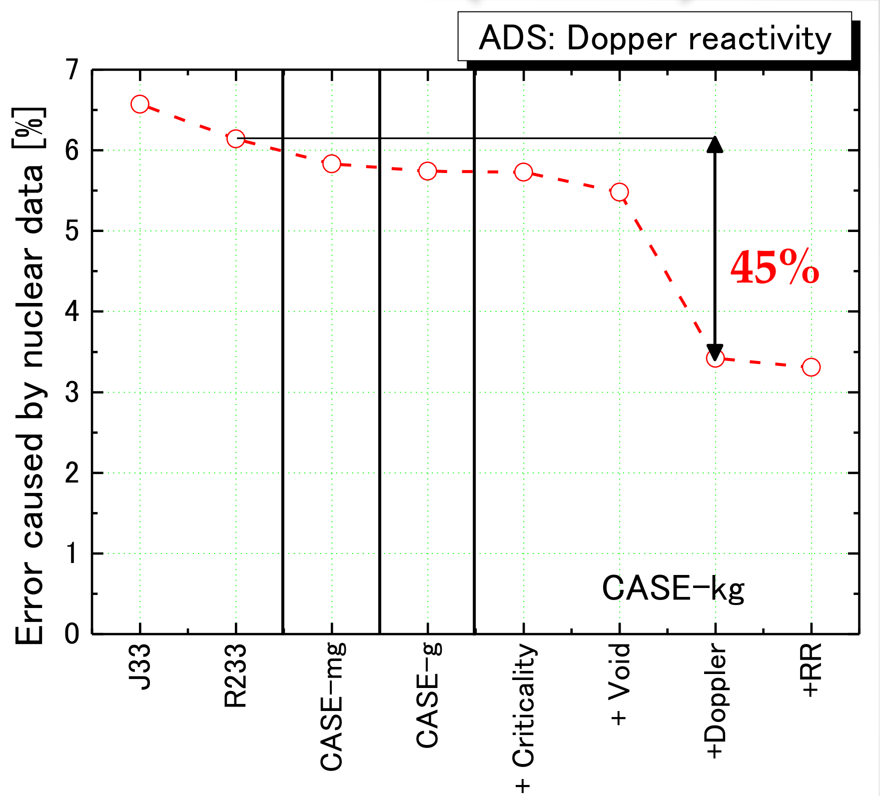
'+RR': Adjusted by '+Doppler' result and measurement of reaction rate ratio in CASE-kg.

- The error was reduced significantly in **CASE-g** although the result in CASE-mg was hardly reduced.
- Experiments in CASE-kg was also effective to improve the error. The value was about **35%** smaller than that in R233.

*1: T. Hazama, et al., JNC TN9400 2002-064, (2002)

Calculation Results (2/2)





- For both coolant void and Doppler reactivity, the errors were hardly improved in CASE-mg and CASE-g.
- The measurement of coolant void reactivity or Doppler reactivity **with kg-order MA** was effective for the error reduction.

Summary for uncertainty analysis

- To estimate the uncertainty caused by the nuclear data
  - The uncertainty was **1090 pcm** for the criticality, **280 pcm** for the void reactivity and **6.3%** for the Doppler reactivity based on the JENDL-4.0 covariance data.
  - However, the uncertainty of criticality, 1090 pcm, looks smaller than the result of IAEA benchmark activity (2000-3000pcm).
- To estimate the impact of the TEF-P
  - The uncertainty would be reduced up to **45%** by the TEF-P.

Thank you for your attention!