2007 Symposium on Nuclear Data, Nov. 29-30, 2007,
RICOTTI Convention Center, Tokai, Ibaraki, Japan

Integral Test for JENDL-4

Benchmark Results with Preliminary Version of JENDL Actinoid File

29 Nov. 2007

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Background and Objective

- The JENDL Actinoid File (JENDL/AC) is under developing at JAEA.

- Most of the evaluations in JENDL/AC will be taken over to a part of the next general purpose file JENDL-4.

Benchmark calculation for various type of reactors to confirm present performances of JENDL/AC and to polish it more and more.

Goal

Good performance superior to recent other nuclear data files : JENDL-3.3, JEFF-3.1, ENDF/B=VII.0, etc.

Framework of JENDL/AC Benchmark Test

Japan Nuclear Data Committee (JNDC)
Subcommittee on Reactor Constants:
• **Reactor Integral Test WG**
:

Work, Comments, Advices

Once or twice a year

Benchmark results,
Reactor constants

JAEA-Tokai
Nuclear Data Center

Preliminary
nuclear data files

Information
exchange
(once a month)

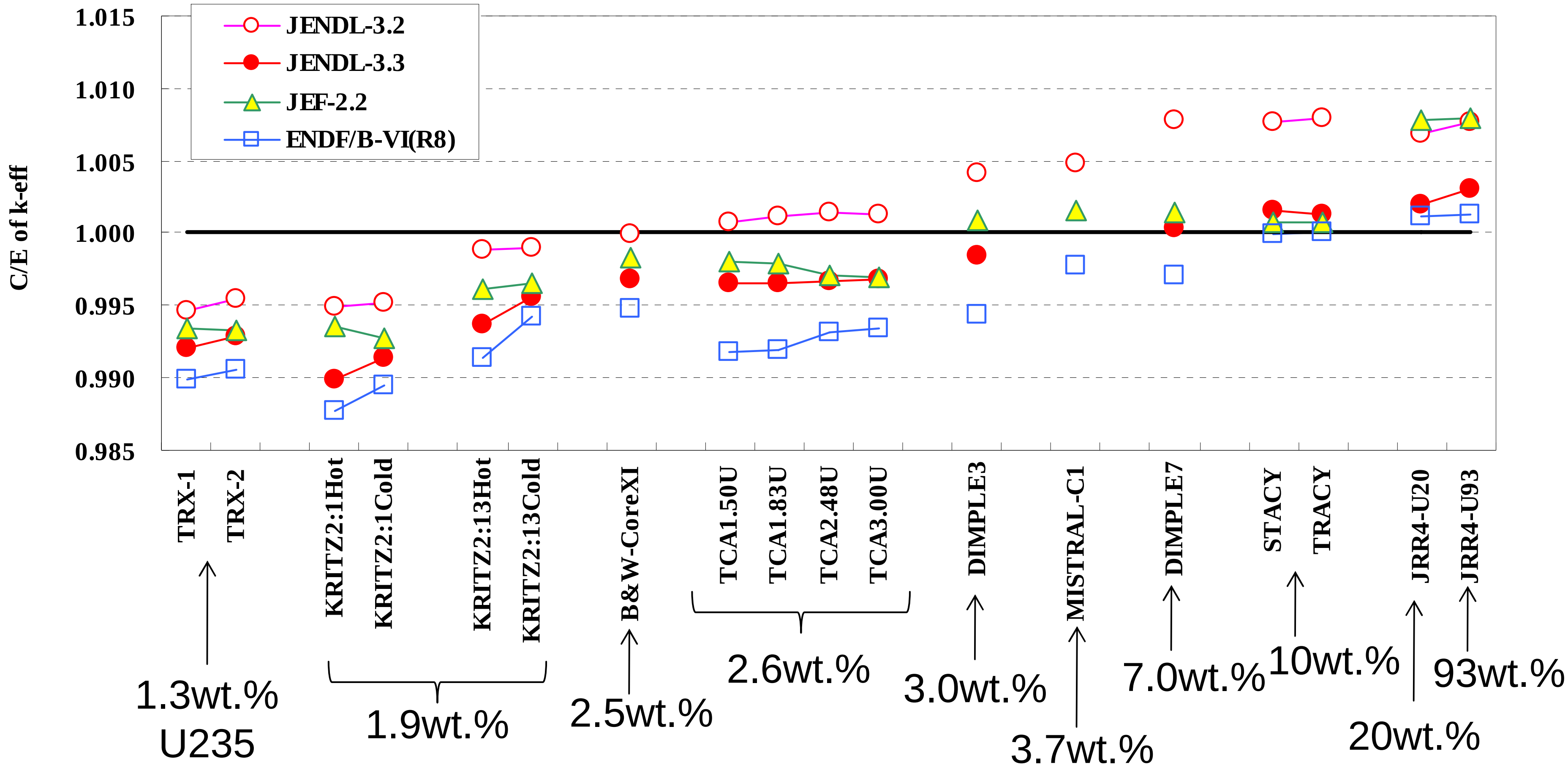
Benchmark results and
sensitivity data

JAEA-Tokai
Reactor Physics Group

data

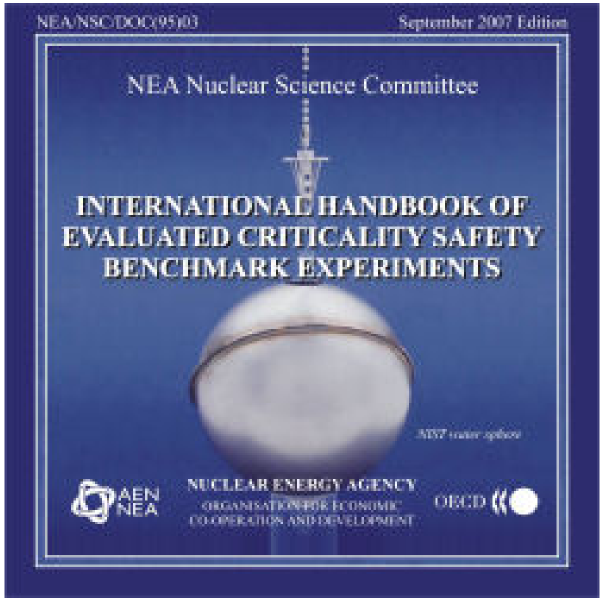
JAEA-Oarai
Reactor Physics Analysis
and Evaluation Group

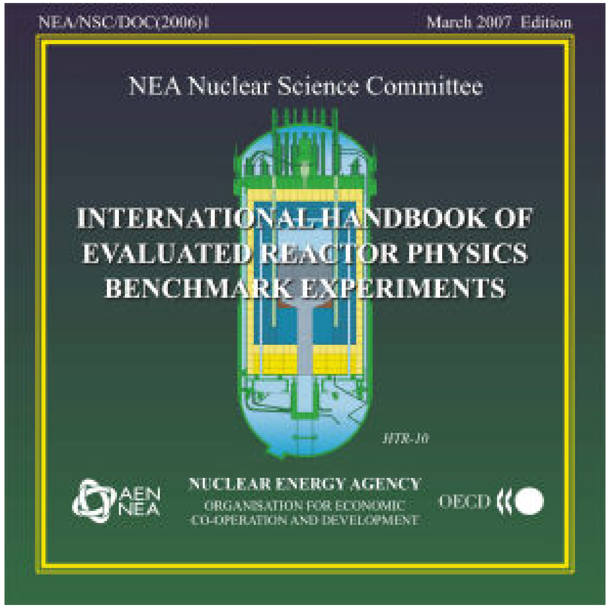
Criticality Benchmark with Former Nuclear Data Files



Proc. of the 2002 Symposium on Nuclear Data, Tokai, Japan, Nov. 21-22, JAERI-Conf 2003-006, pp15-21, (2003).

Benchmark Materials





Handbook (Sep. 2007 Edition) of International Criticality Safety Benchmark Evaluation Project (ICSBEP)

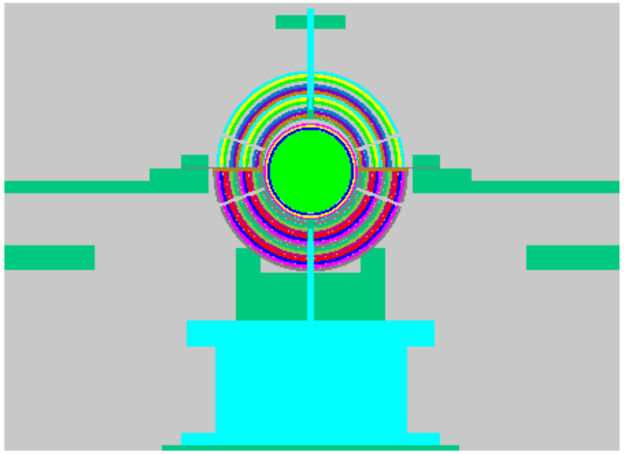
Handbook (Mar. 2007 Edition) of International Reactor Physics Experiment Evaluation Project (IRPhEP)

Benchmark Calculation

- Continuous-Energy Monte Carlo calculation (MVP)

Neutron histories :
20 ~ 60 million
→ 1σ error of keff < 0.0002

- Detailed geometrical model specified in the benchmark handbooks



- Multi-group deterministic calculation for small reactivity analysis or sensitivity study with the codes:
  - SLAROM-UF
  - SN solvers in the CBG system

Classification of Benchmark Problems in ICSBEP Handbook

Fissile Materials

HEU : Highly Enriched Uranium Systems (60%～)
IEU : Intermediate and Mixed Enrichment Uranium Systems (10～60%)
LEU : Low Enriched Uranium Systems (～10%)
MIX : Mixed Plutonium-Uranium Systems (e.g. MOX fuel)
U233 : Uranium-233 Systems
PU : Plutonium Systems
SPEC : Special Isotope Systems

Fuel Forms

SOL : Solution
COMP : Compound (e.g. UO_2,MOX,UF_4)
MET : Metal
MISC : Miscellaneous (e.g. UO_2 rods in fuel solution)

Neutron Spectra

FAST : Fast
INTER : Intermediate
THERM : Thermal
MIXED : Mixed e.g. Multi-region system with different neutron spectra

Example of case index for TCA-UO_2 cores

LEU-COMP-THERM-006
(LCT6.1 ~ LCT6.18)
Different critical configurations
lattice pitch, critical water height, horizontal lattice size (NxN), etc.

-001
-002
:
-018

Selected Benchmark Problems

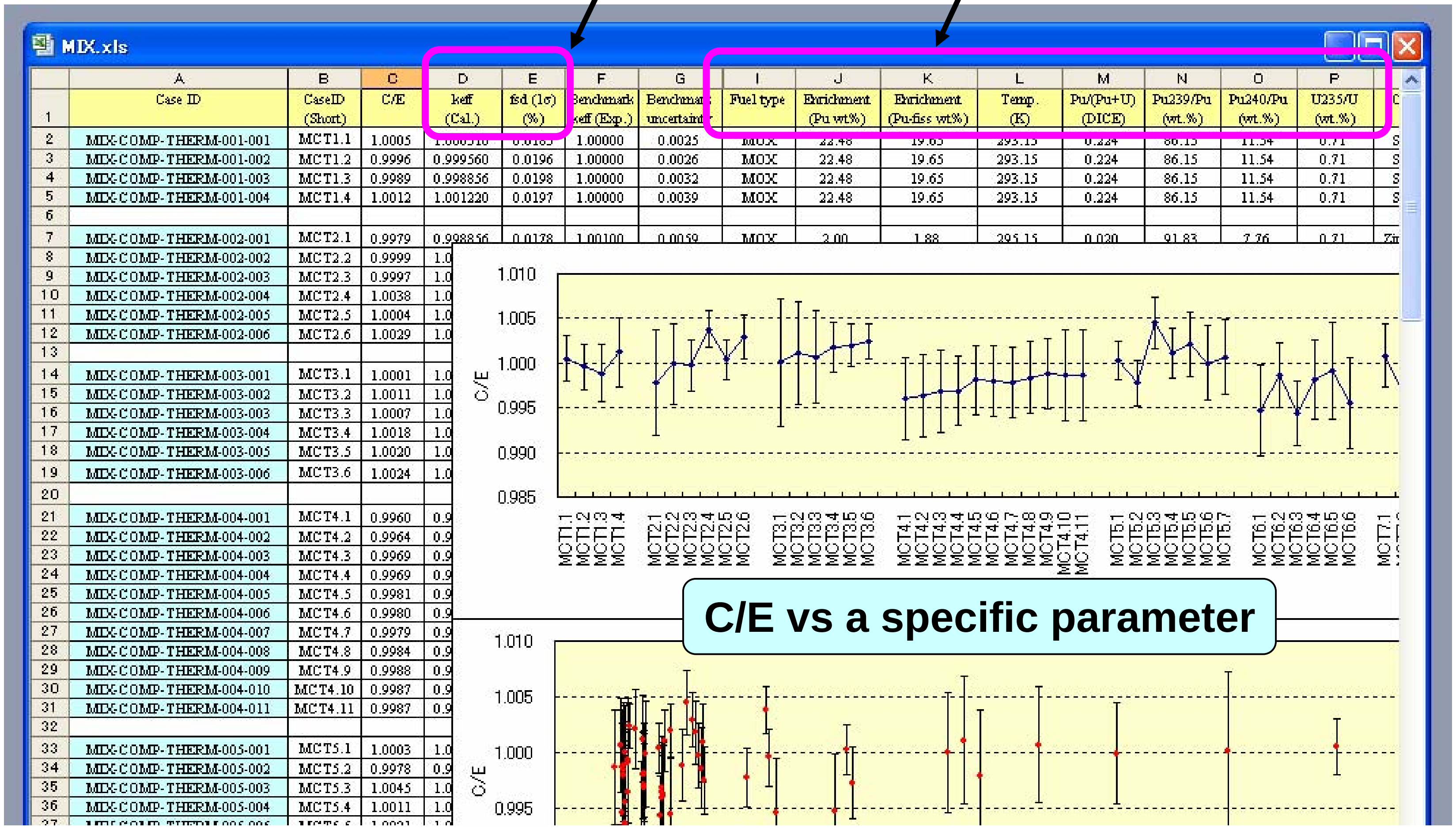
Fuel	Form	Spectra	ICSBEP2006	MVP Cal.
HEU	SOL	INTER	3	2
		THERM	463	50
	COMP	FAST	8	0
		INTER	14	5
		THERM	216	21
		MIXED	45	0
	MET	FAST	304	41
		INTER	14	9
		THERM	127	3
		MIXED	32	8
	MISC	THERM	7	0
IEU	SOL	THERM	5	0
	COMP	FAST	2	1
		INTER	14	2
		THERM	41	1
		MIXED	3	0
	MET	FAST	20	11
LEU	SOL	THERM	104	77
	COMP	THERM	1066	194
	MET	THERM	65	13
	MISC	THERM	11	0
MIX	SOL	THERM	72	9
	COMP	FAST	1	0
		INTER	3	0
		THERM	255	63
		MIXED	17	0
	MET	FAST	45	9
		INTER	2	0
		MIXED	1	0
	MISC	FAST	8	0
		THERM	56	53
		MIXED	8	0
U233	COMP	THERM	5	0
	SOL	INTER	29	29
		THERM	192	44
		MIXED	8	3
	MET	FAST	10	10
		THERM	1	0
PU	SOL	THERM	529	208
	COMP	FAST	6	0
		INTER	1	0
		THERM	21	0
		MIXED	7	0
	MET	FAST	87	37
		INTER	4	4
		THERM	2	2
		MIXED	1	1
SPEC	MET	FAST	20	20
Total			3955	930

We have about 1000 results with MVP and JENDL-3.3

Calculated keff and errors

Core parameters (for trend analysis)

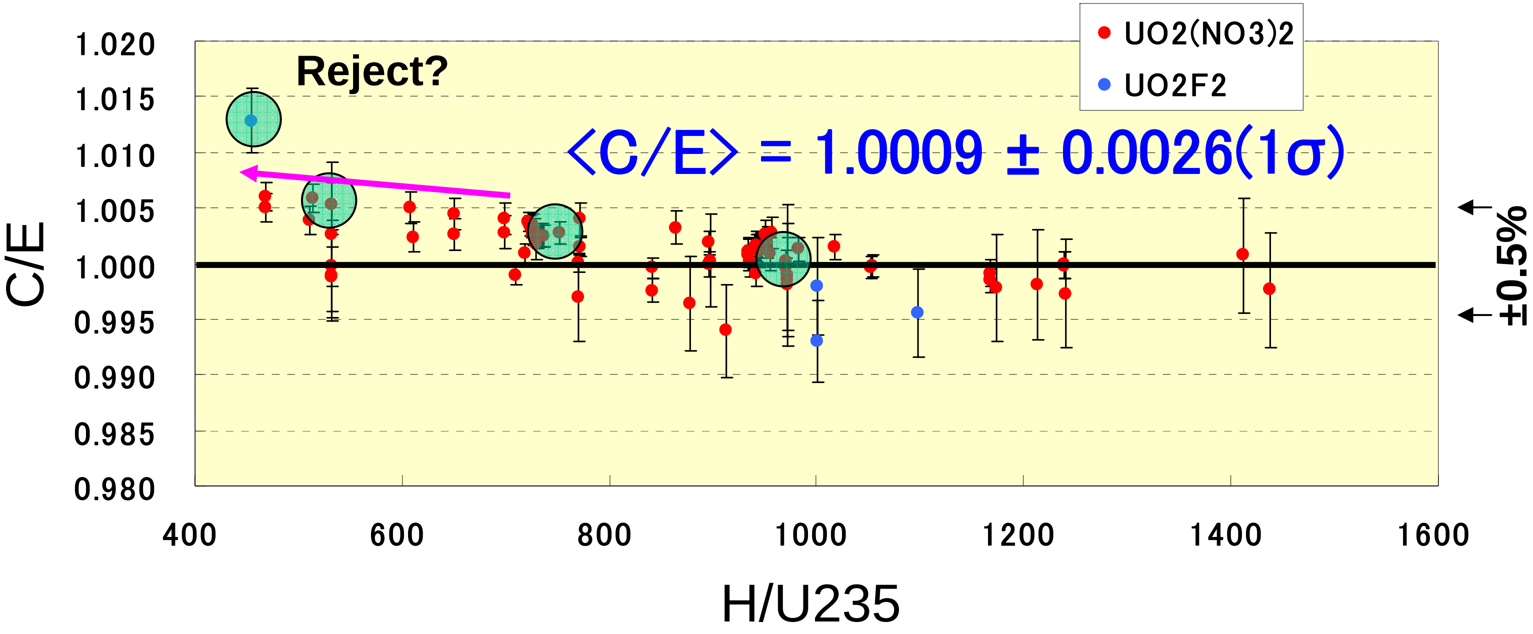
Case index of benchmark problem



JENDL-3.3 Results for LEU/HEU-SOL

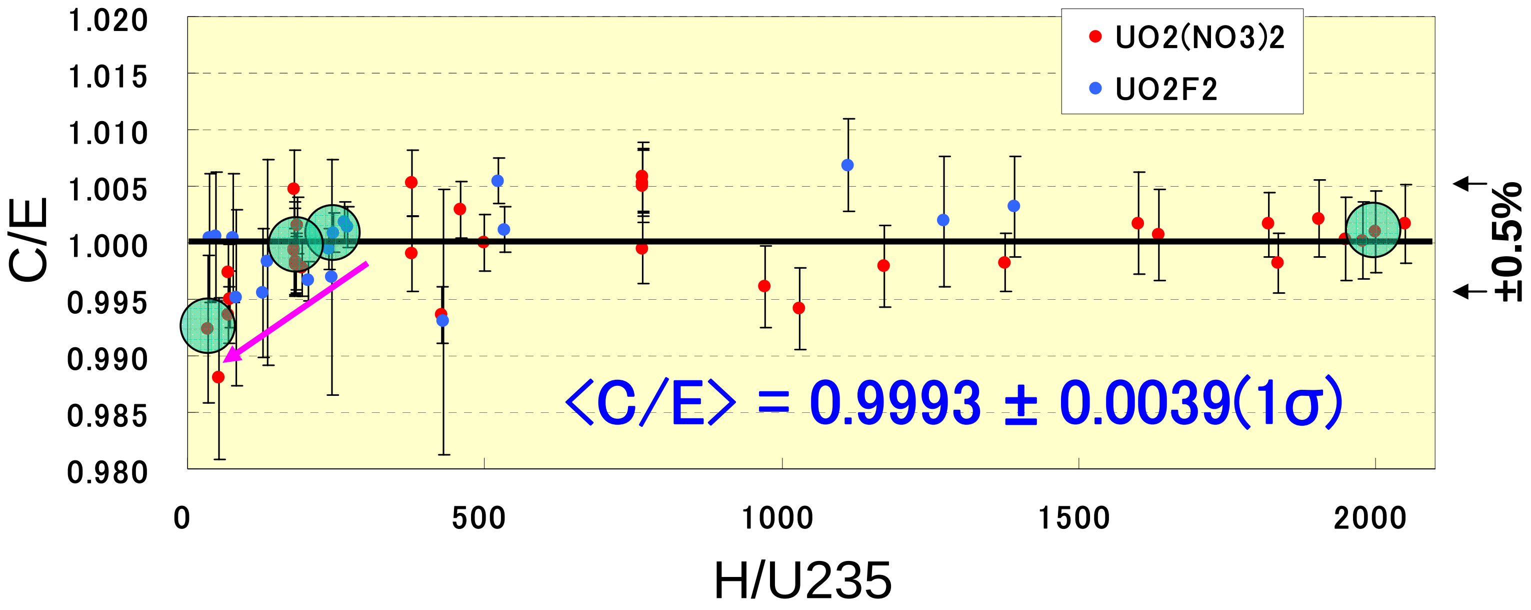
LEU-SOL

5~10%
77 cases



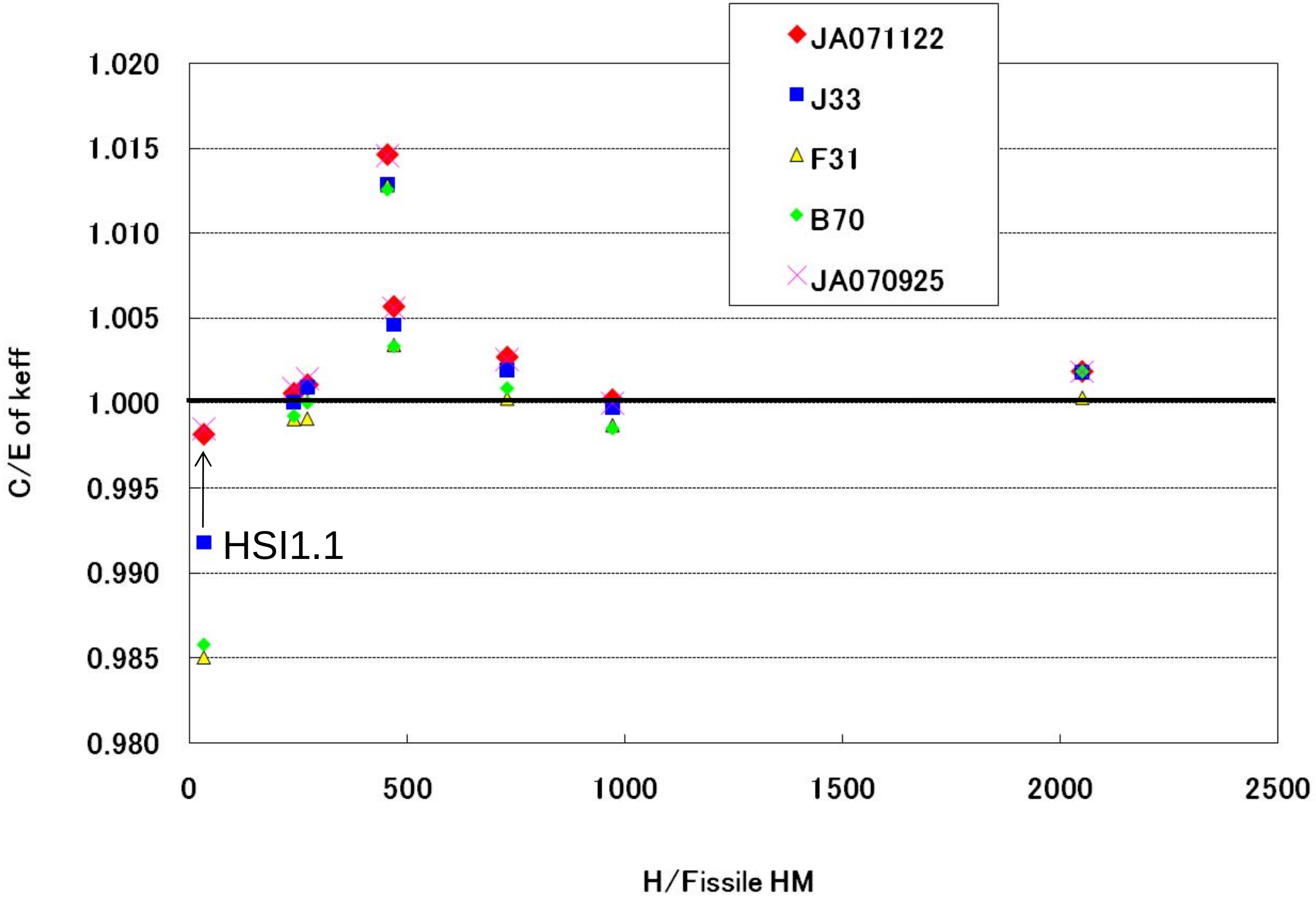
HEU-SOL

89 ~94%
52 cases

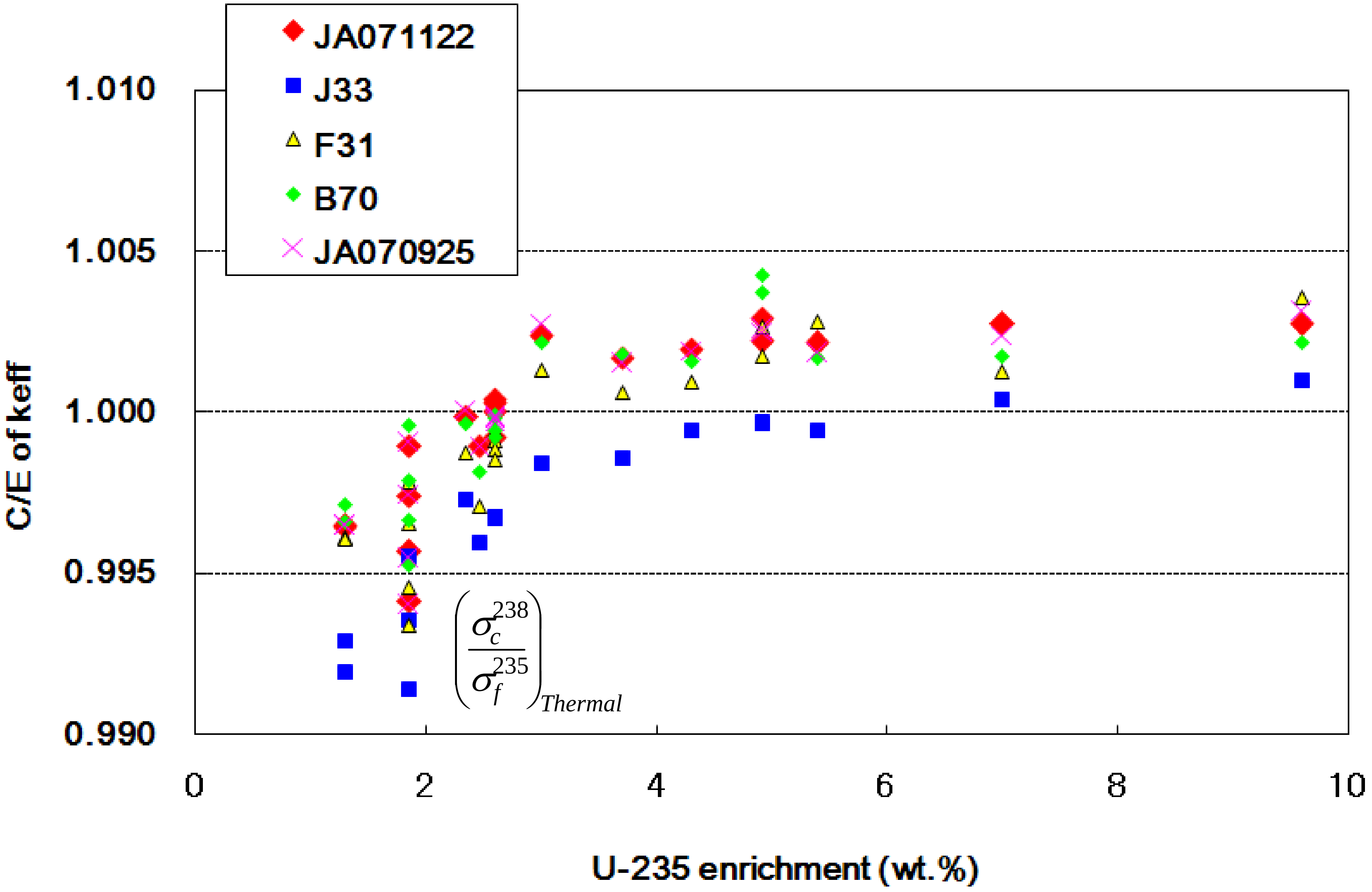


JAEA

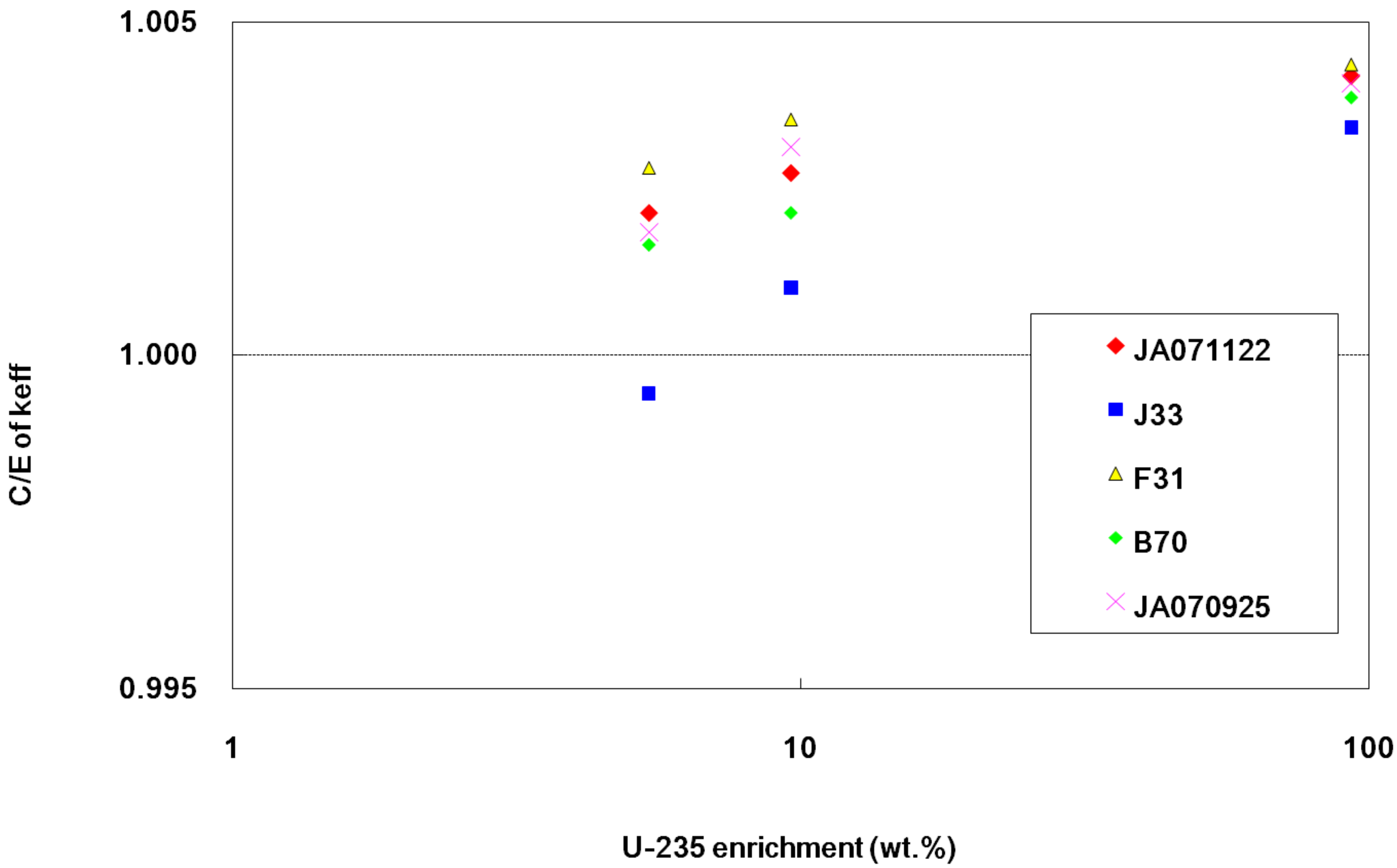
Criticality of U235 Solution Fueled System (LST & HST)



Criticality of Low Enriched U235 Fueled System (LCT)

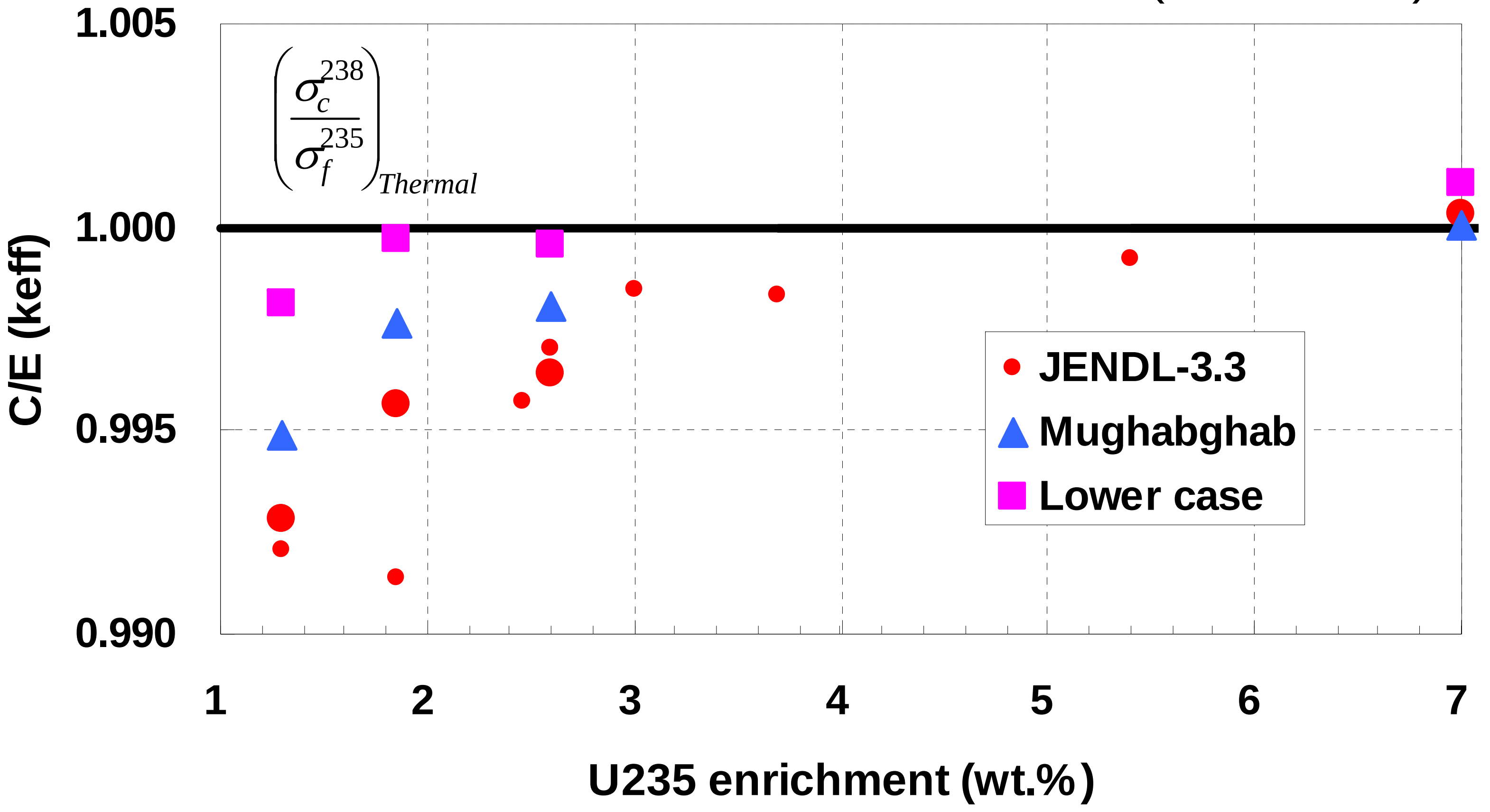


Criticality of Enriched U235 Fueled System (KUCA)



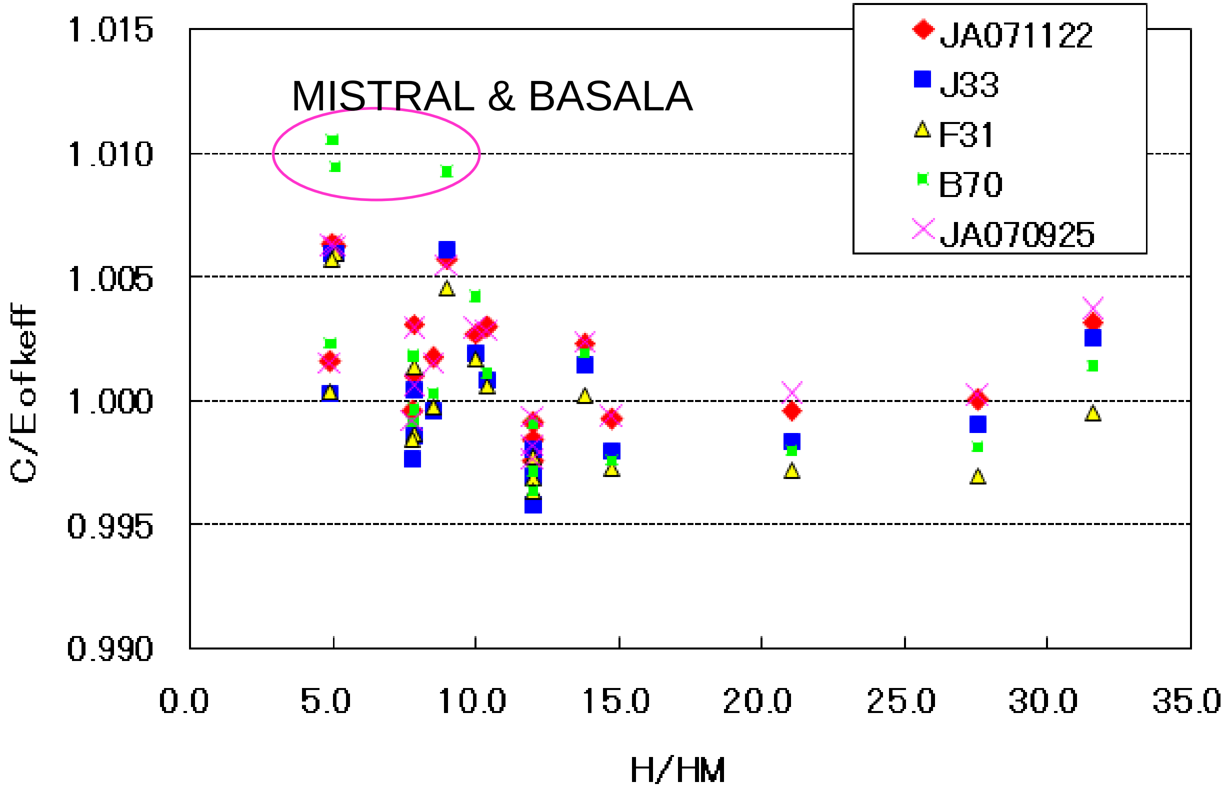
Effect of Thermal Capture Cross Section of U238

2.64 barn (Lower case),
2.68 barn (Mughabghab)
2.717 barn (JENDL-3.3)



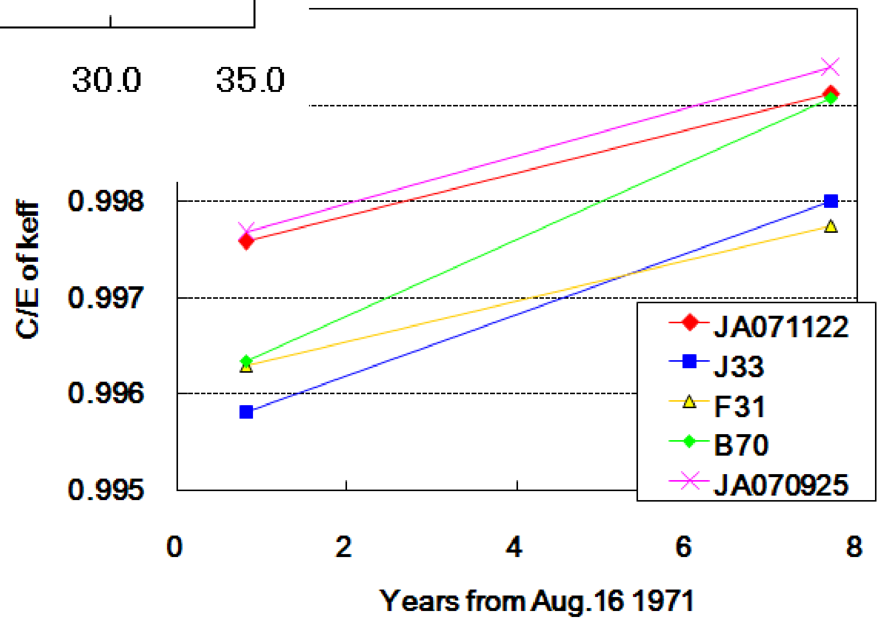
Proc. of the 2004 Symposium on Nuclear Data, Nov. 11-12, 2004,
JAERI, Tokai, Japan, pp.56-63, JAERI-Conf 2005-003, (2005).

Light Water Moderated MOX Fueled System



B70: 0.27%Δk/kk'
J33: 0.22%Δk/kk'
F31: 0.15%Δk/kk'
JA071122: 0.17%Δk/

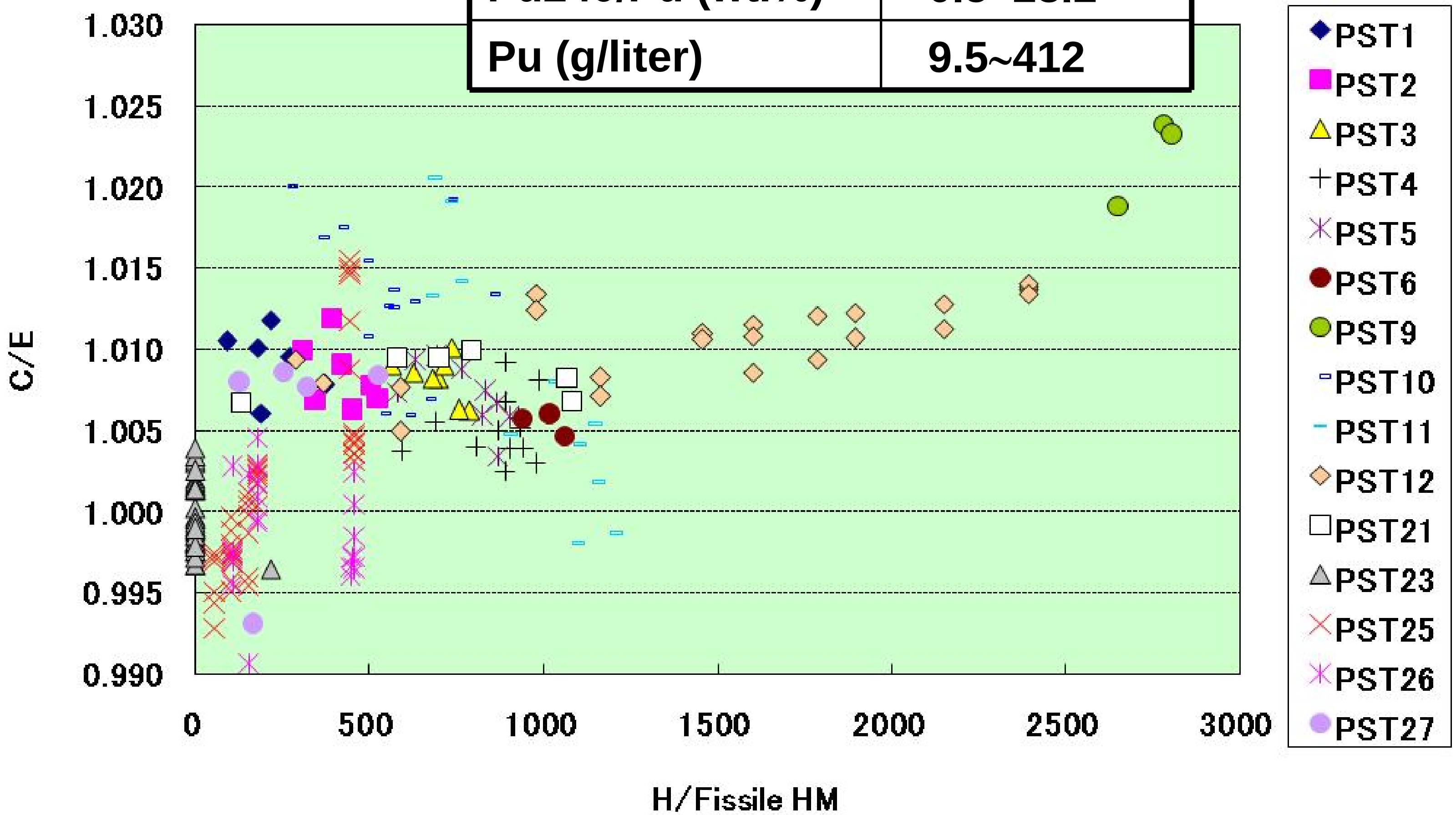
Time-change of C/E for criticality of MOX fueled TCA core due to β-decay of Pu-241 to Am-241



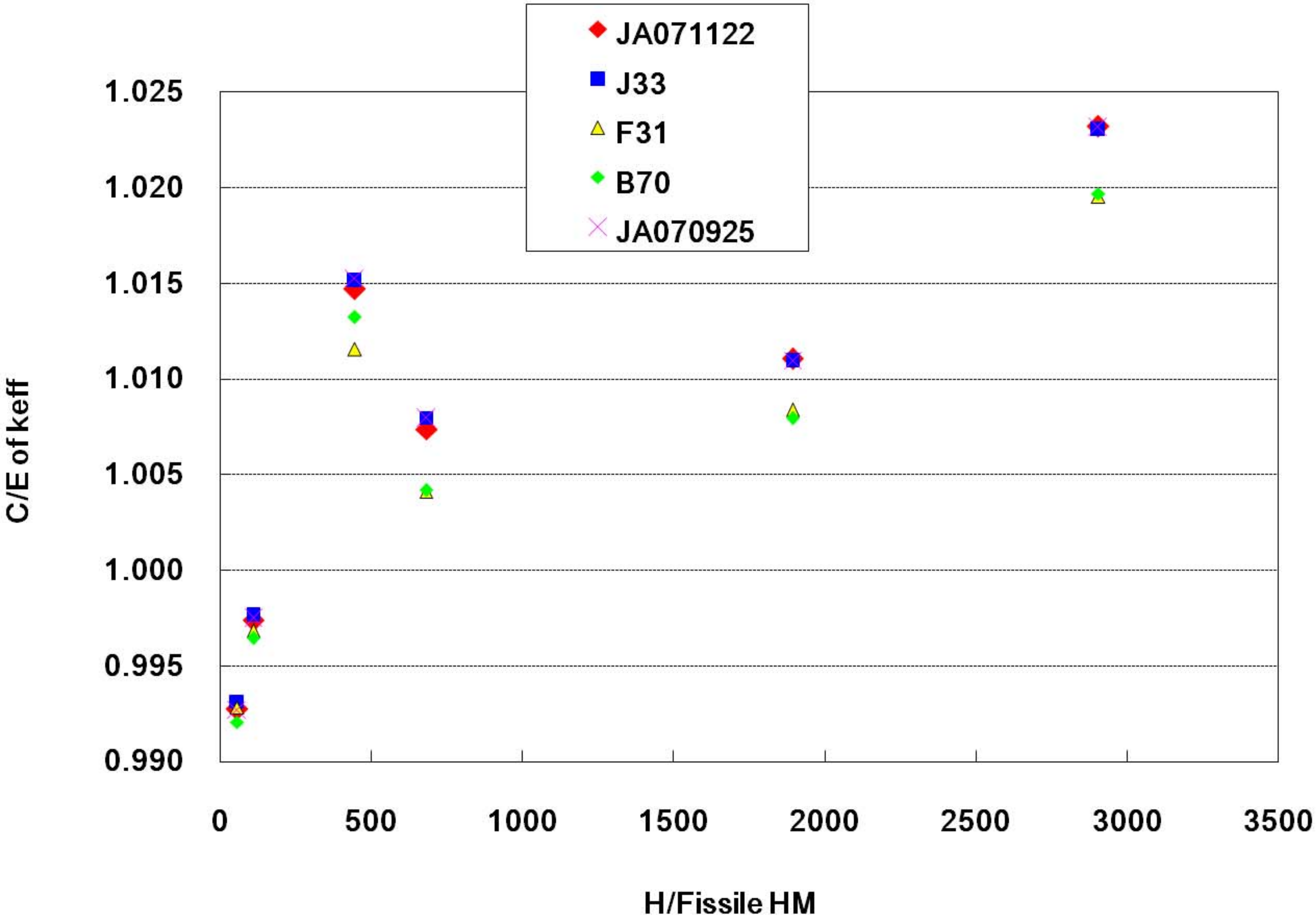
PU-SOL-THERM System (J33)

$\langle C/E \rangle = 1.005 \pm 0.012\ (2\sigma)$

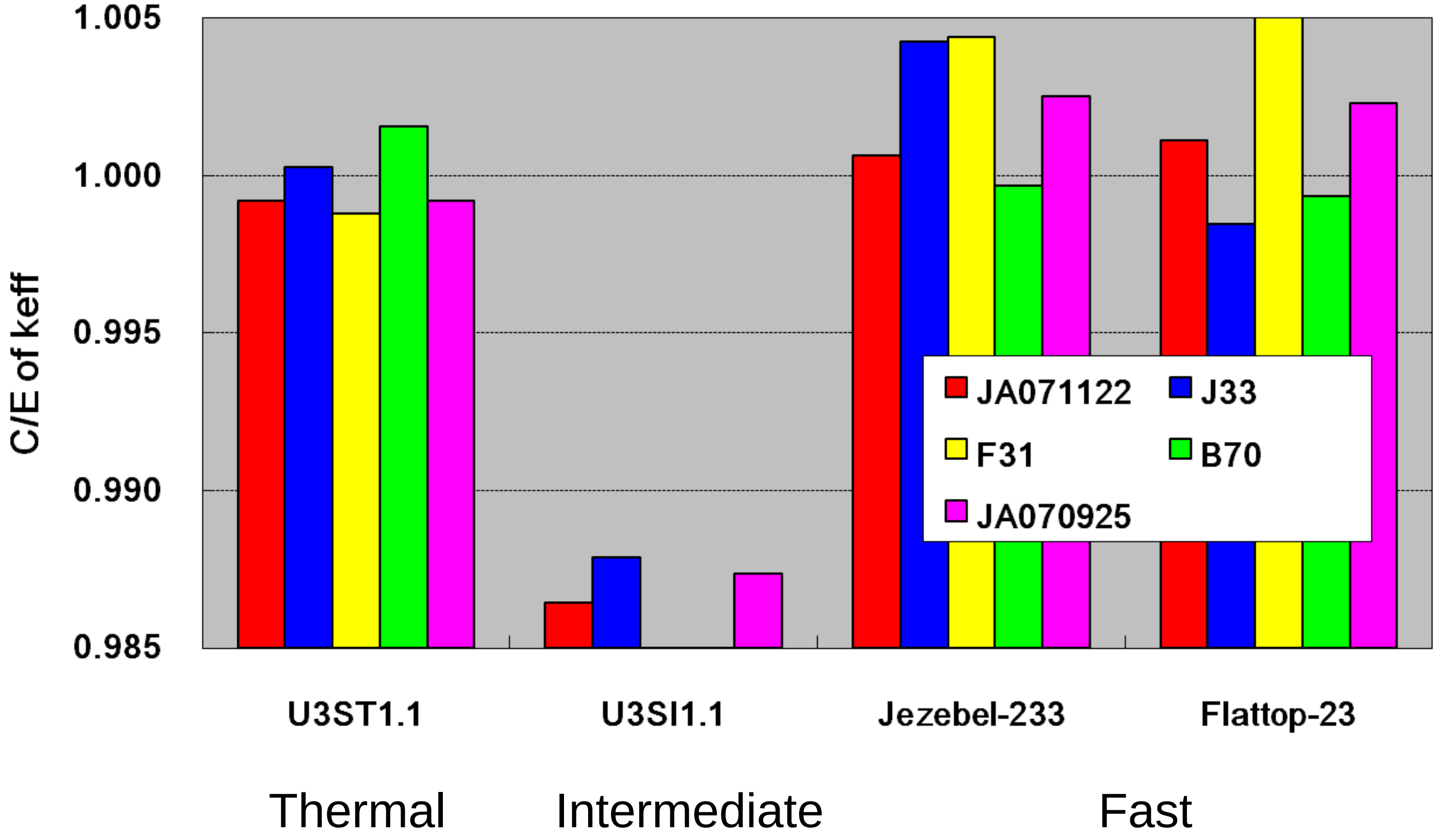
Number of cases	208
Pu/HM (wt.%)	99.4~100
Pu239/Pu (wt.%)	71.8~99.4
Pu240/Pu (wt.%)	0.5~23.2
Pu (g/liter)	9.5~412



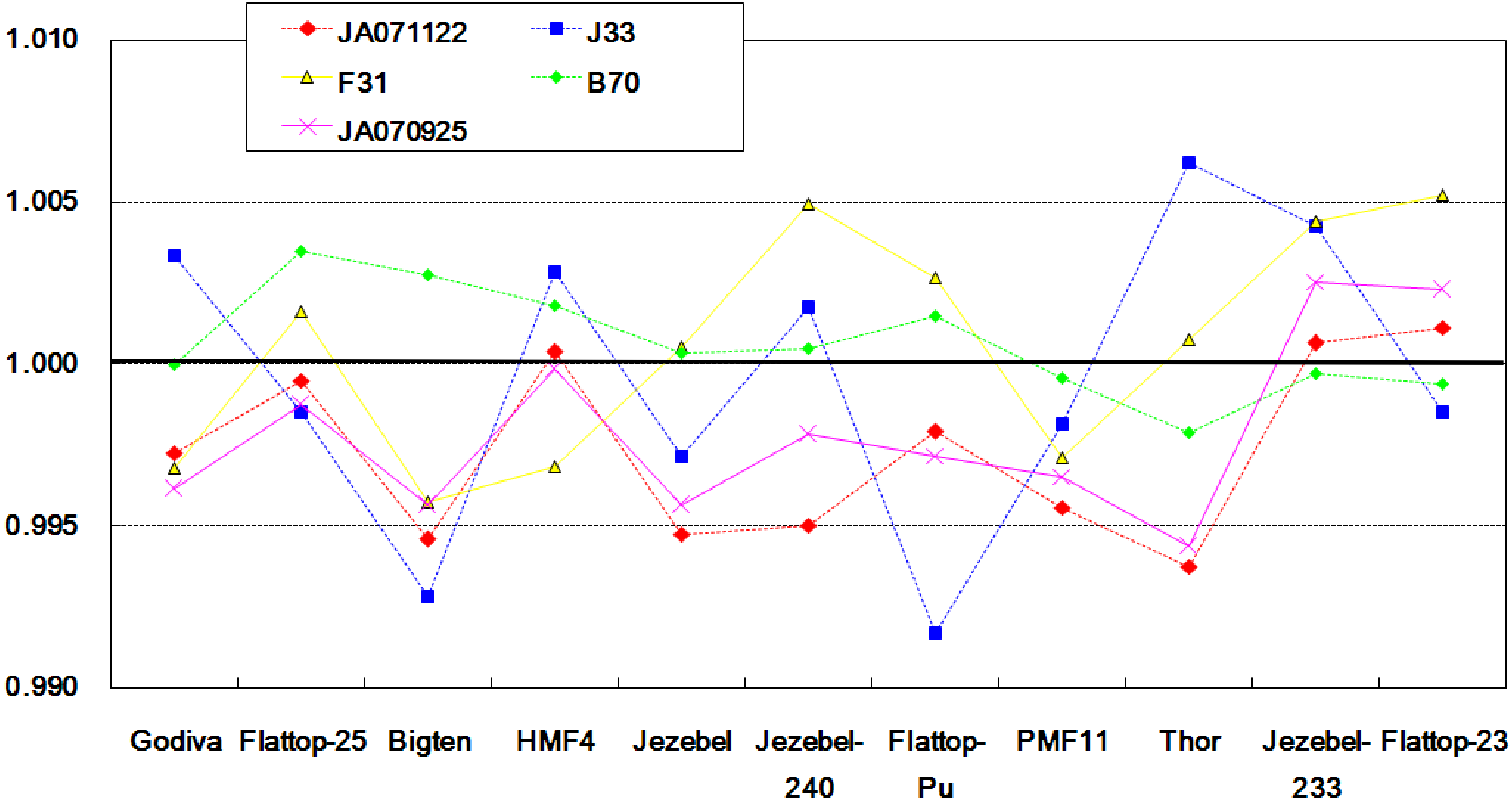
PU-SOL-THERM System (Different Nuclear Data)



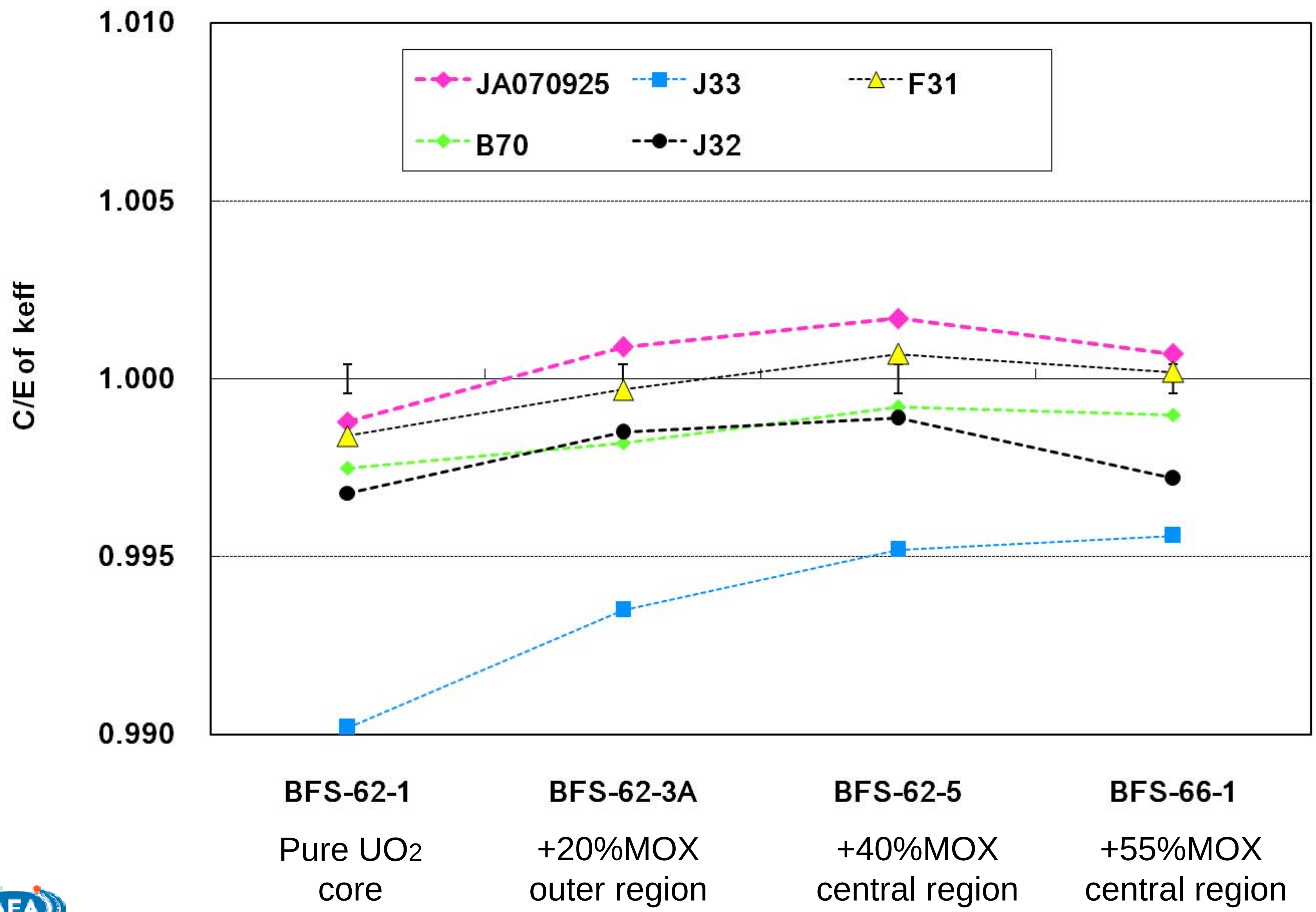
U233 Fueled System



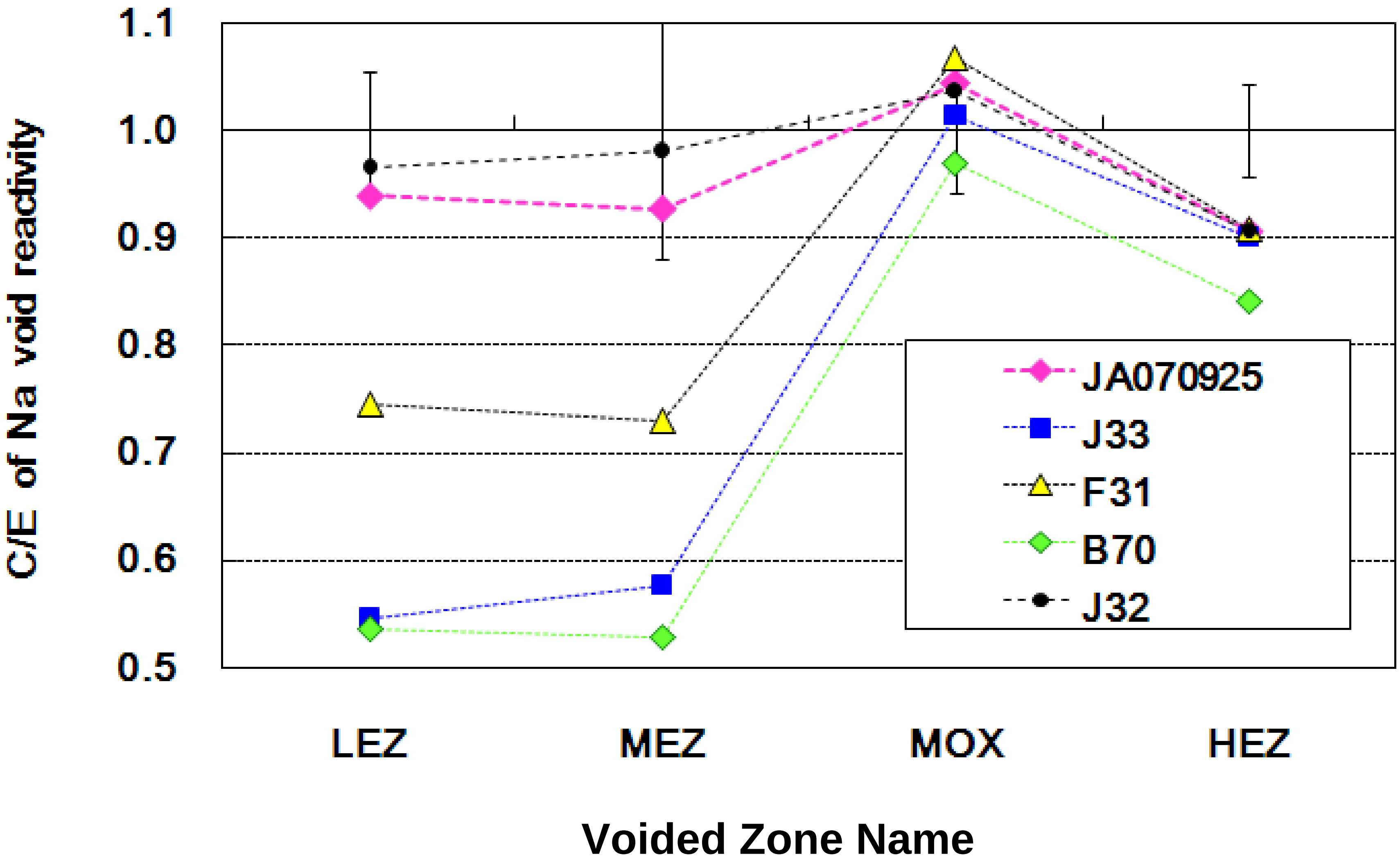
Small Reactor Benchmark



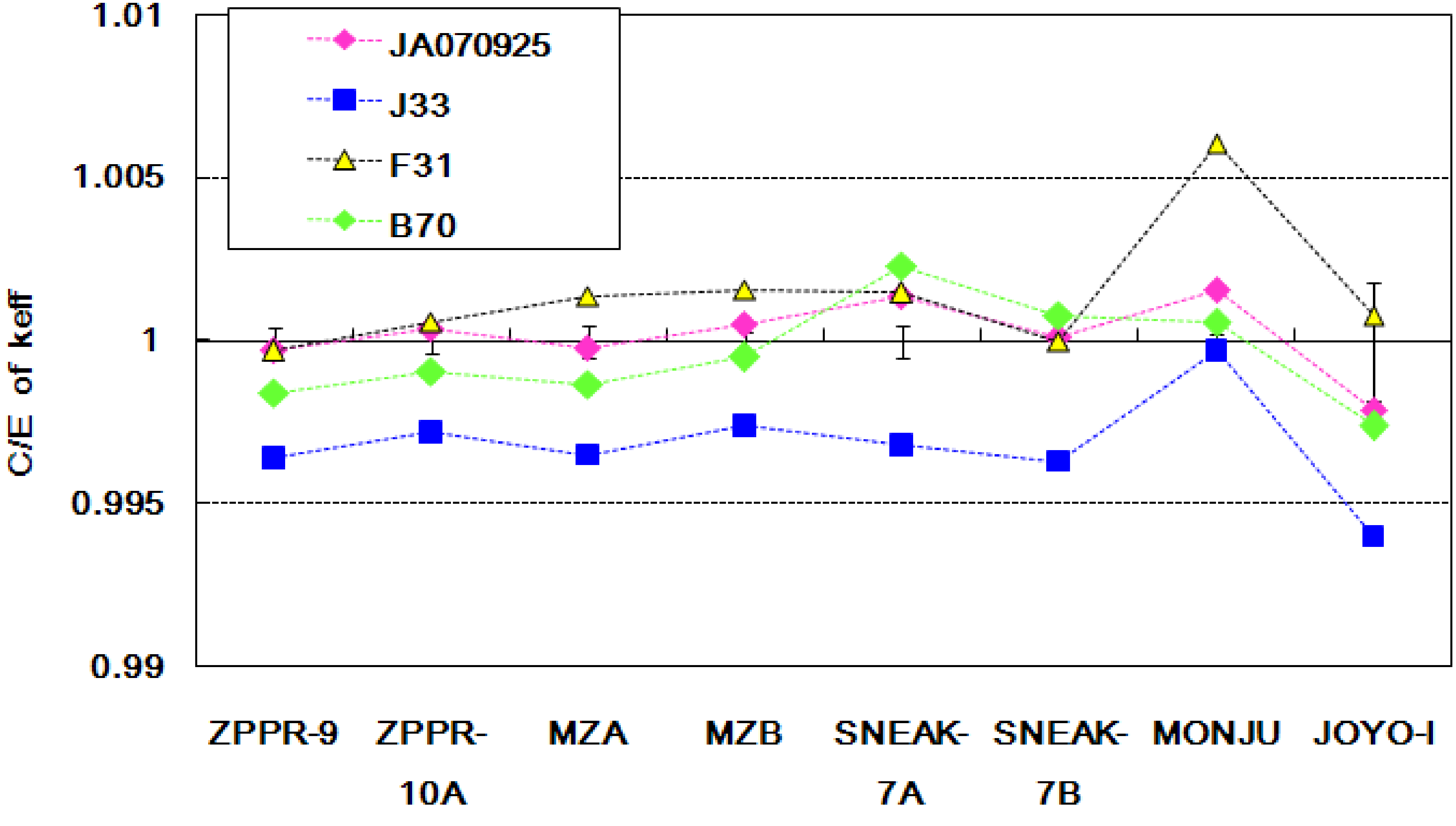
Criticality of Uranium Fueled Fast Reactors (BFS-2)



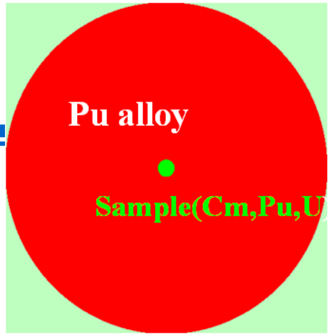
Na Void Reactivity of U Fueled Fast Reactor (BFS-62-3A)



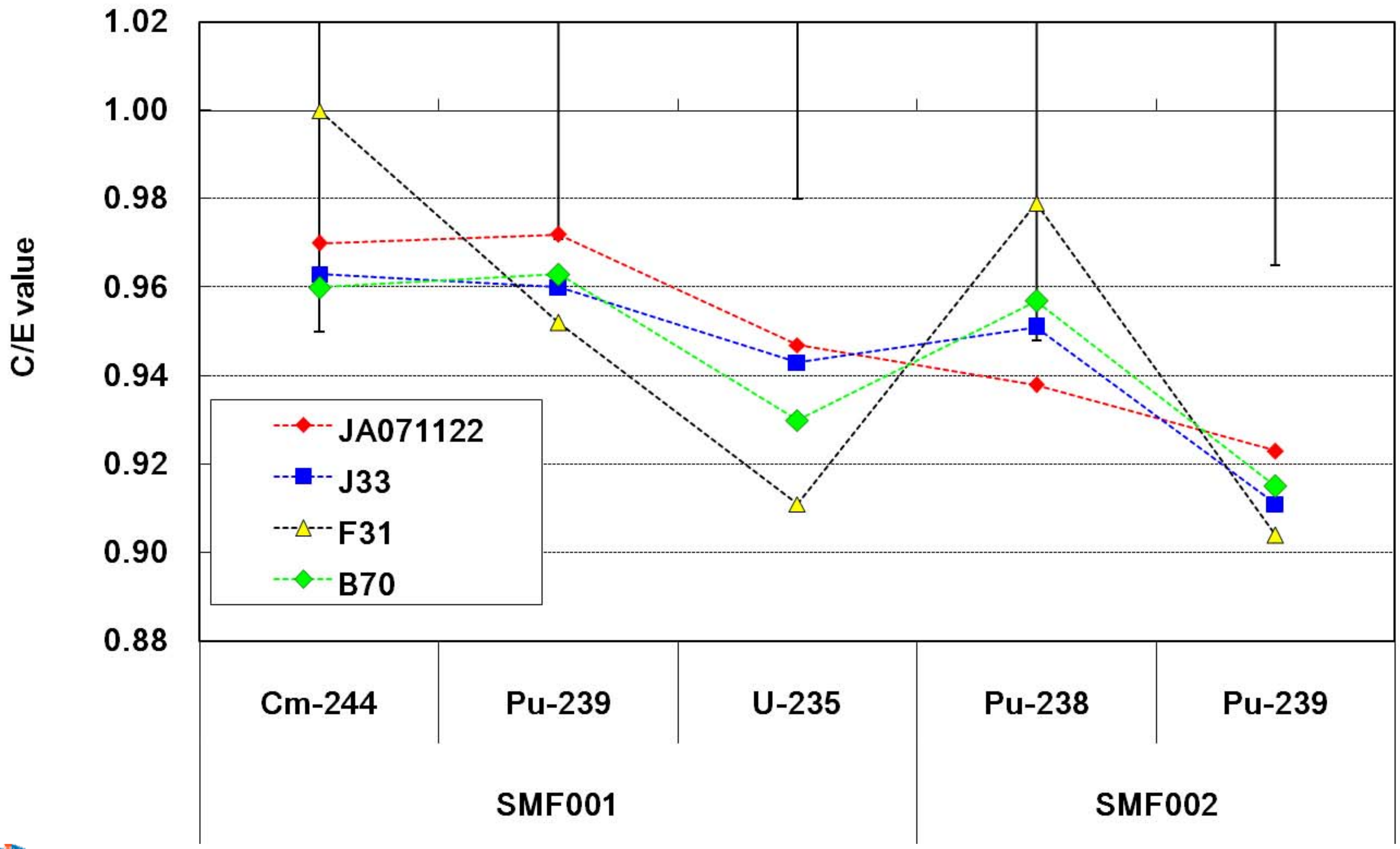
Criticality of MOX Fueled Fast Reactors



SPEC-MET-FAST Benchmark (Cm244, Pu238)

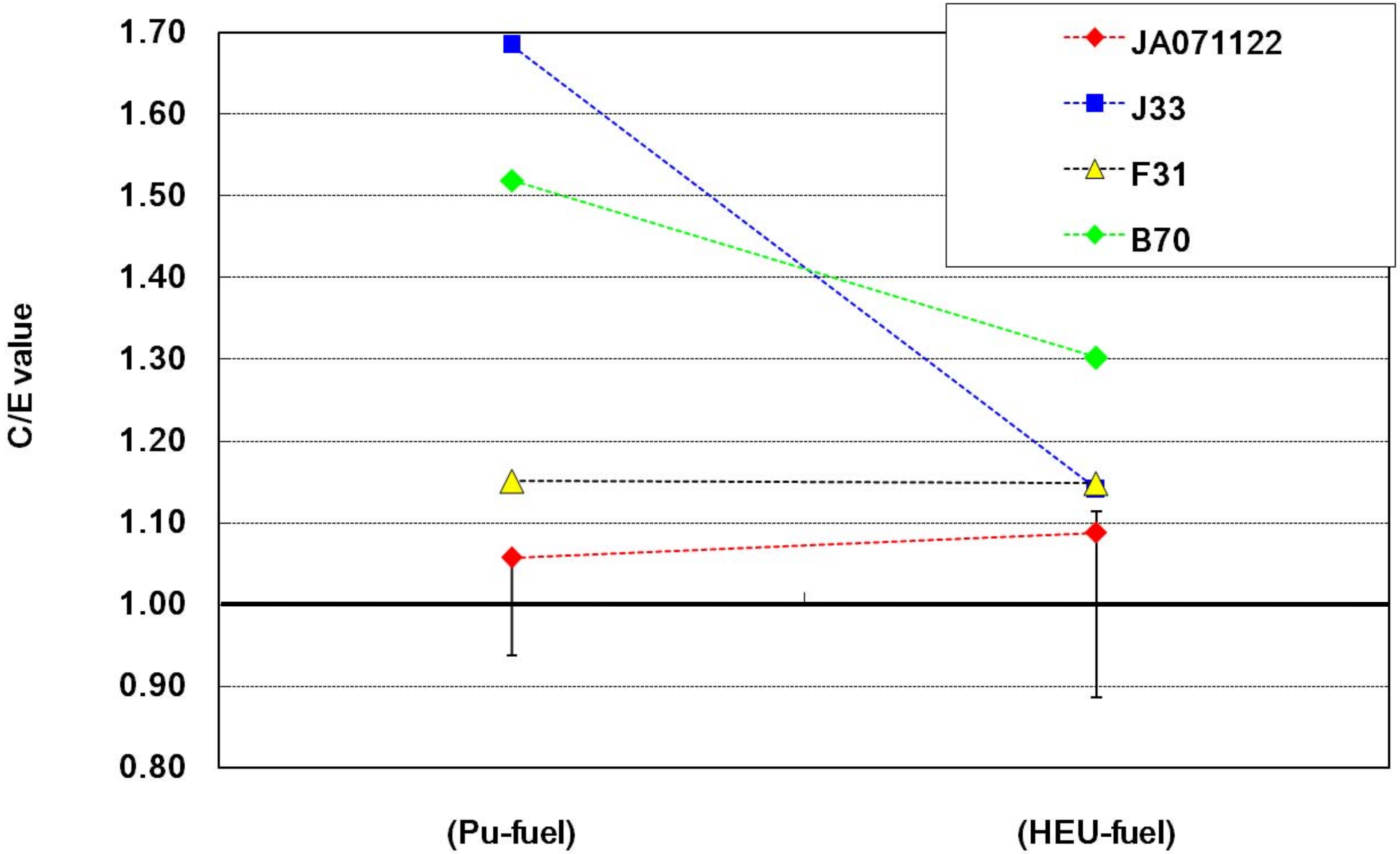


Sample reactivity in Pu alloy

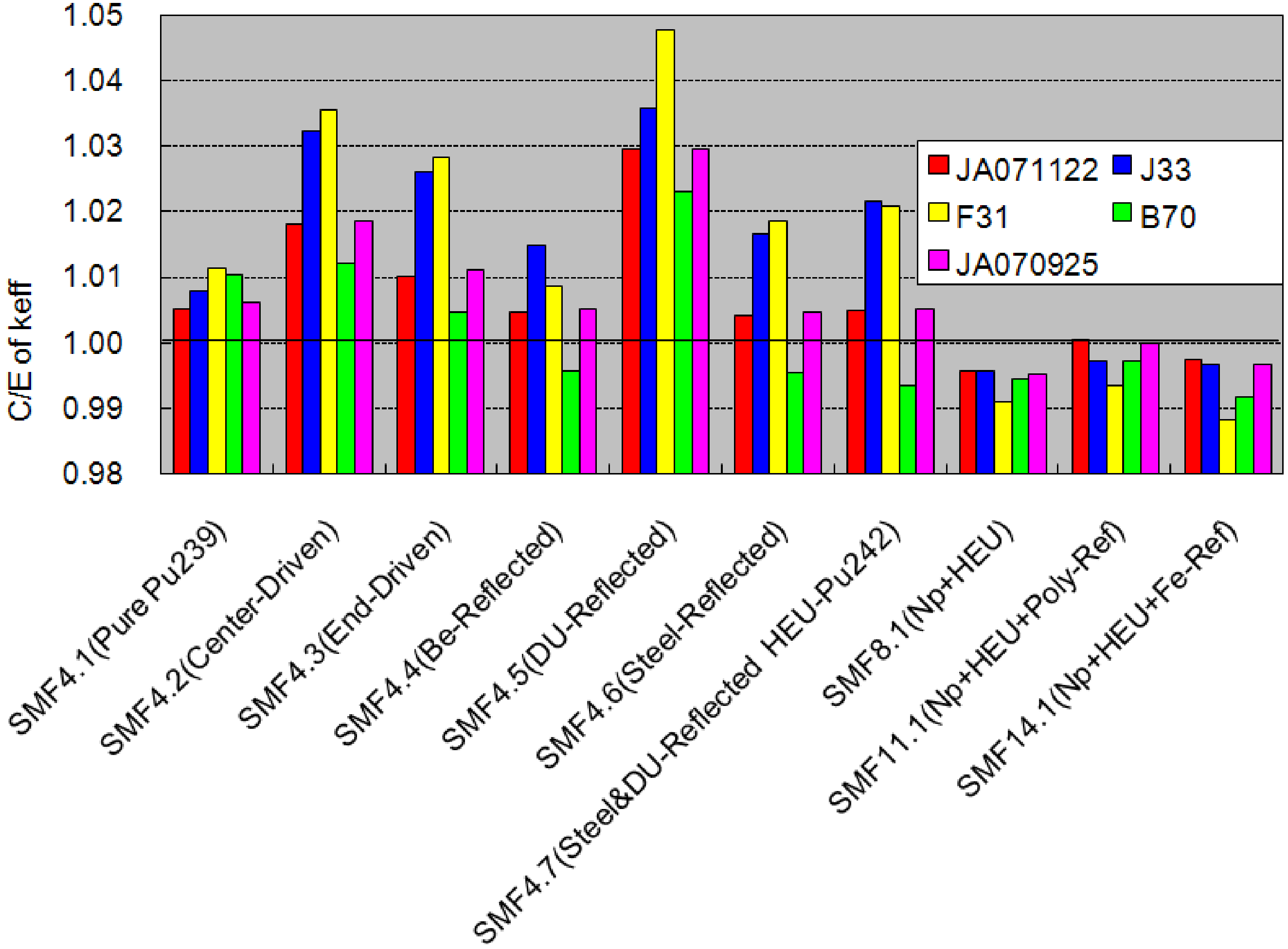


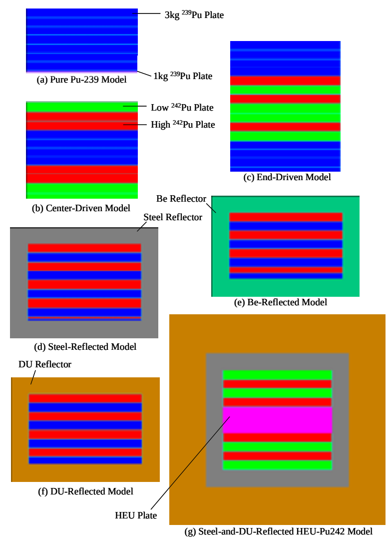
SPEC-MET-FAST Benchmark (Np237)

Np237 sample reactivity in Pu or HEU fuel (SMF3)



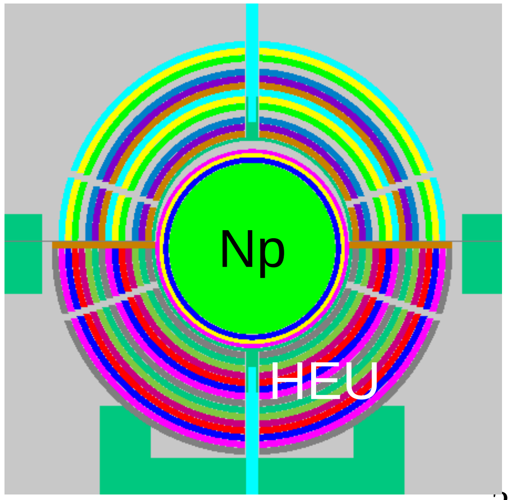
SPEC-MET-FAST Benchmark (Pu242, Np237)



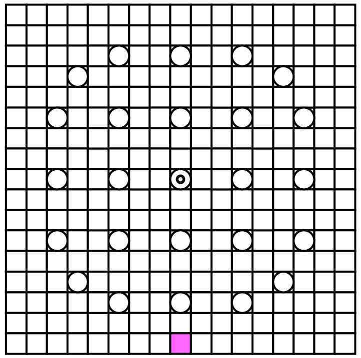


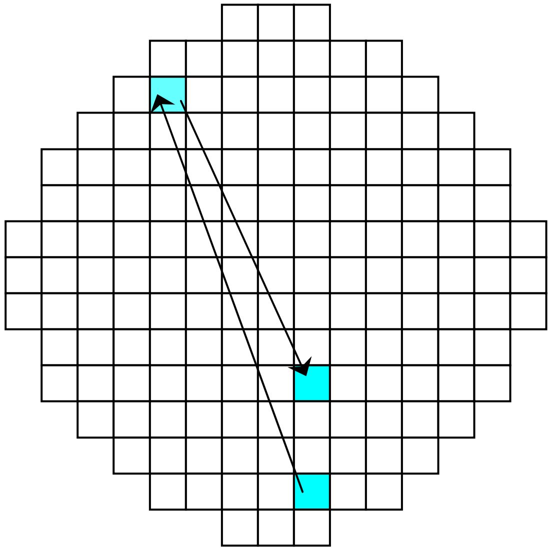
SMF4

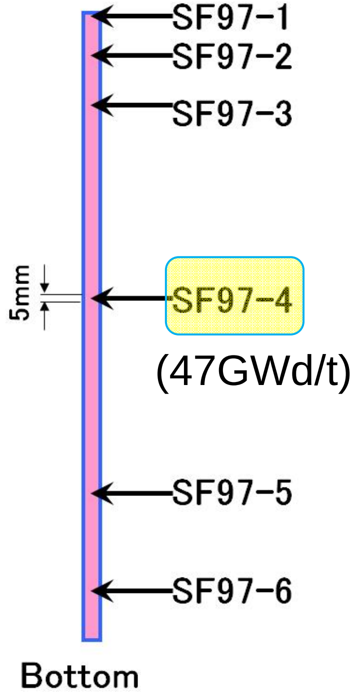
SMF8

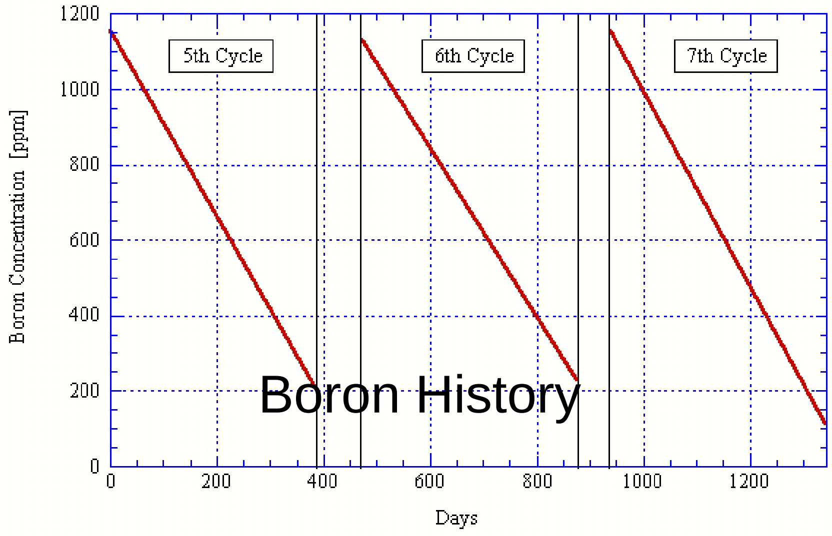


PIE Analysis by MVP-BURN for PWR Spent Fuel

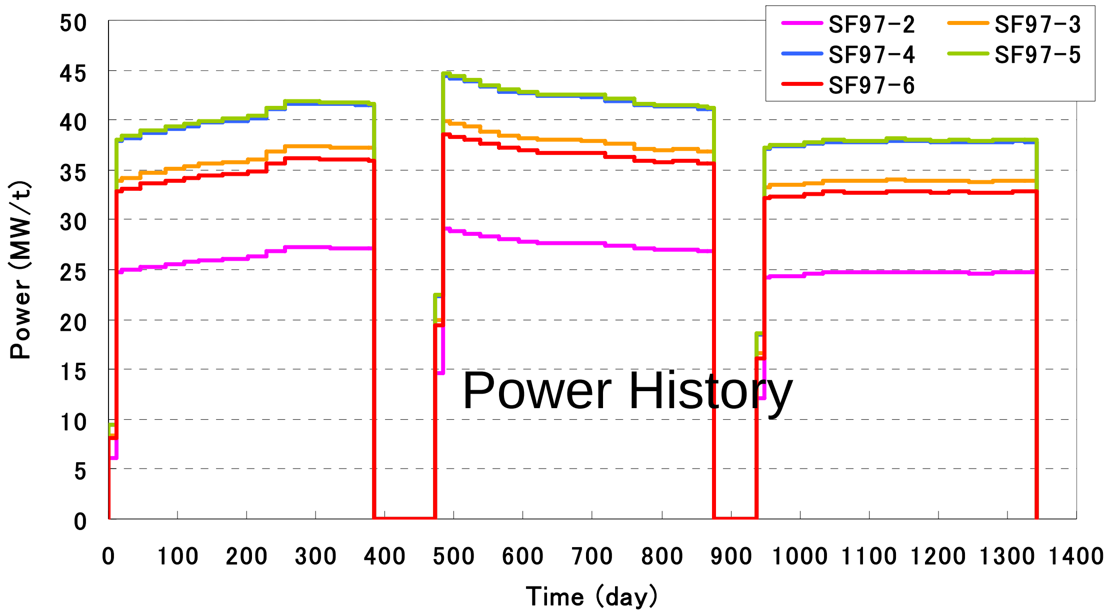






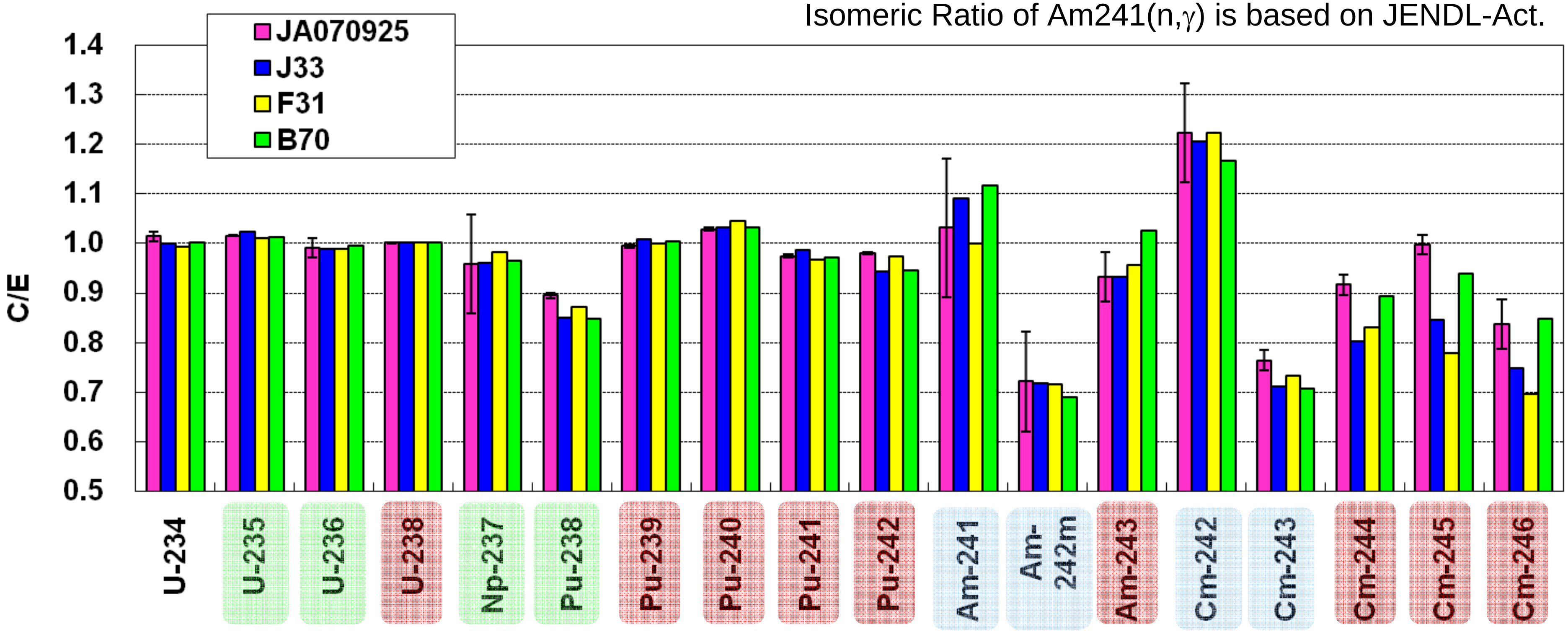


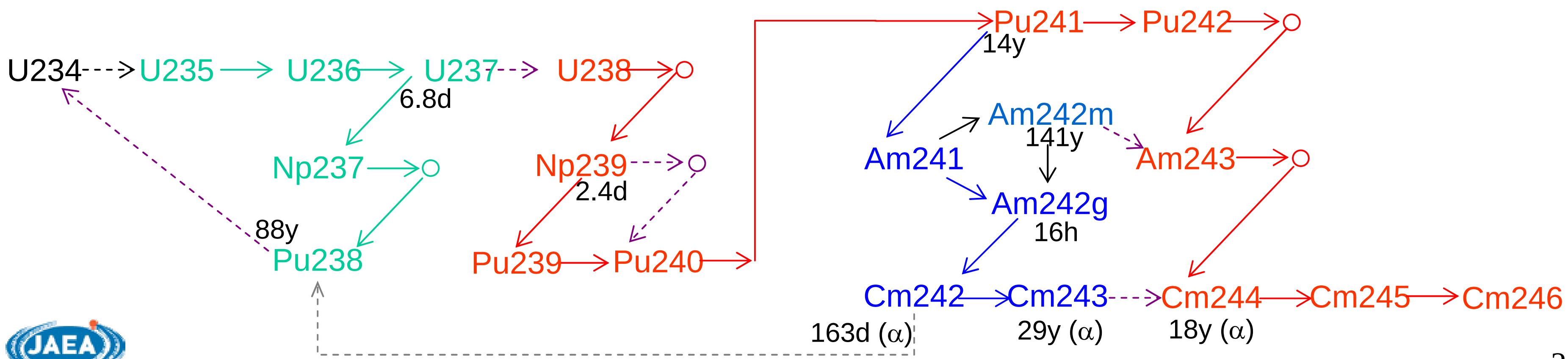
Boron History of Takahama-3 Reactor



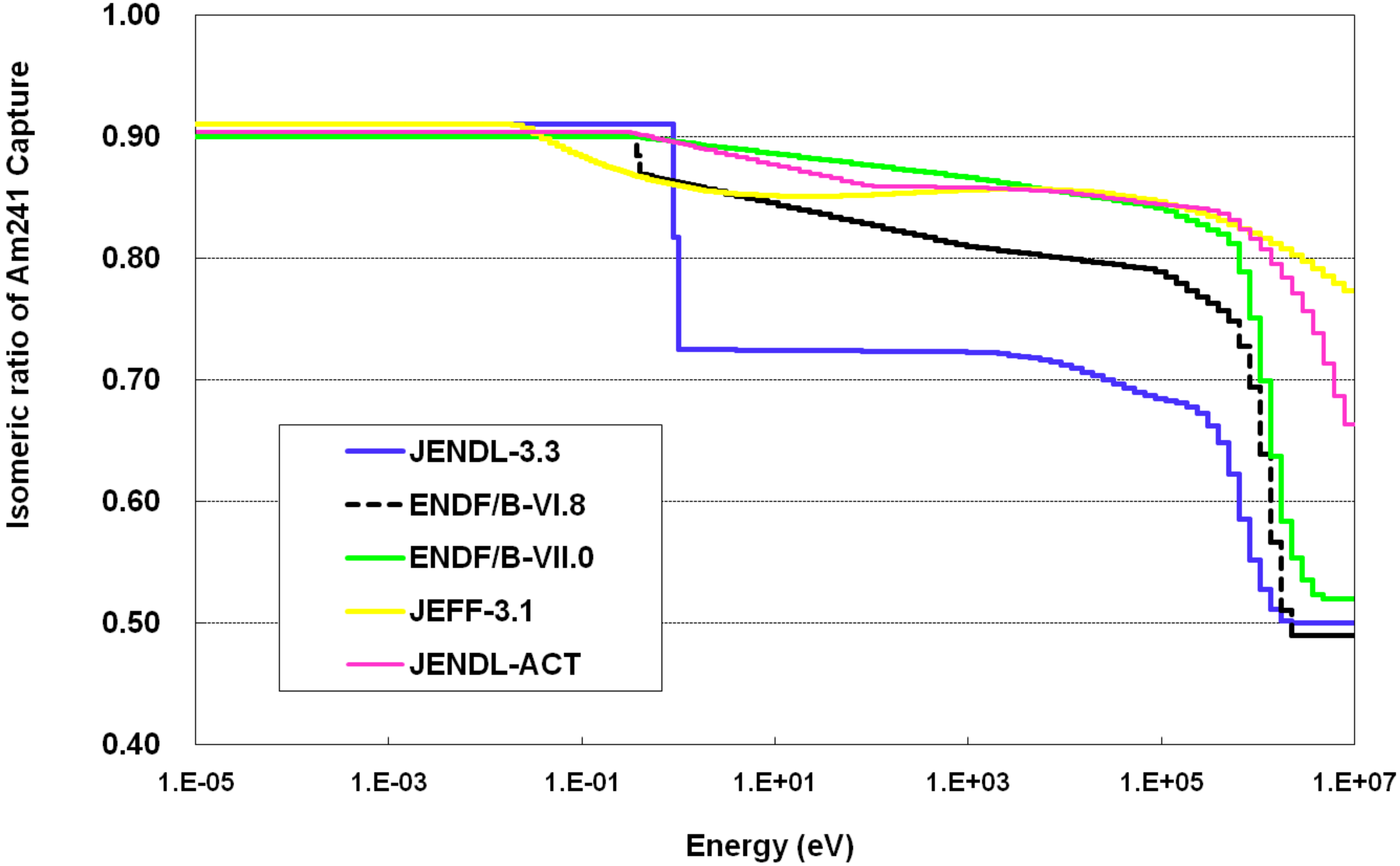
PIE Analysis by MVP-BURN for PWR Spent Fuel

Isomeric Ratio of Am241(n,γ) is based on JENDL-Act.

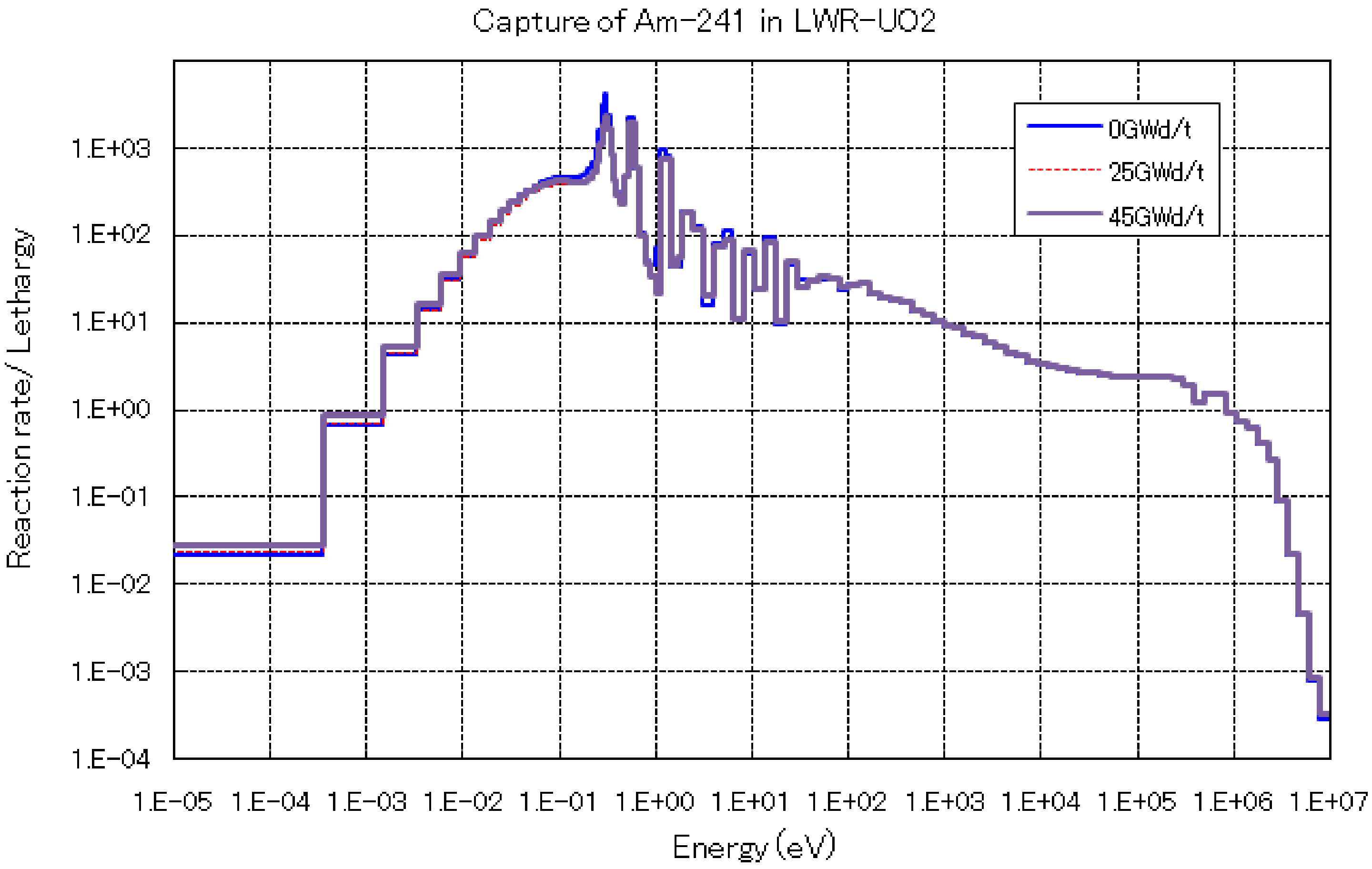




Isomeric Ratio of Am241 Capture



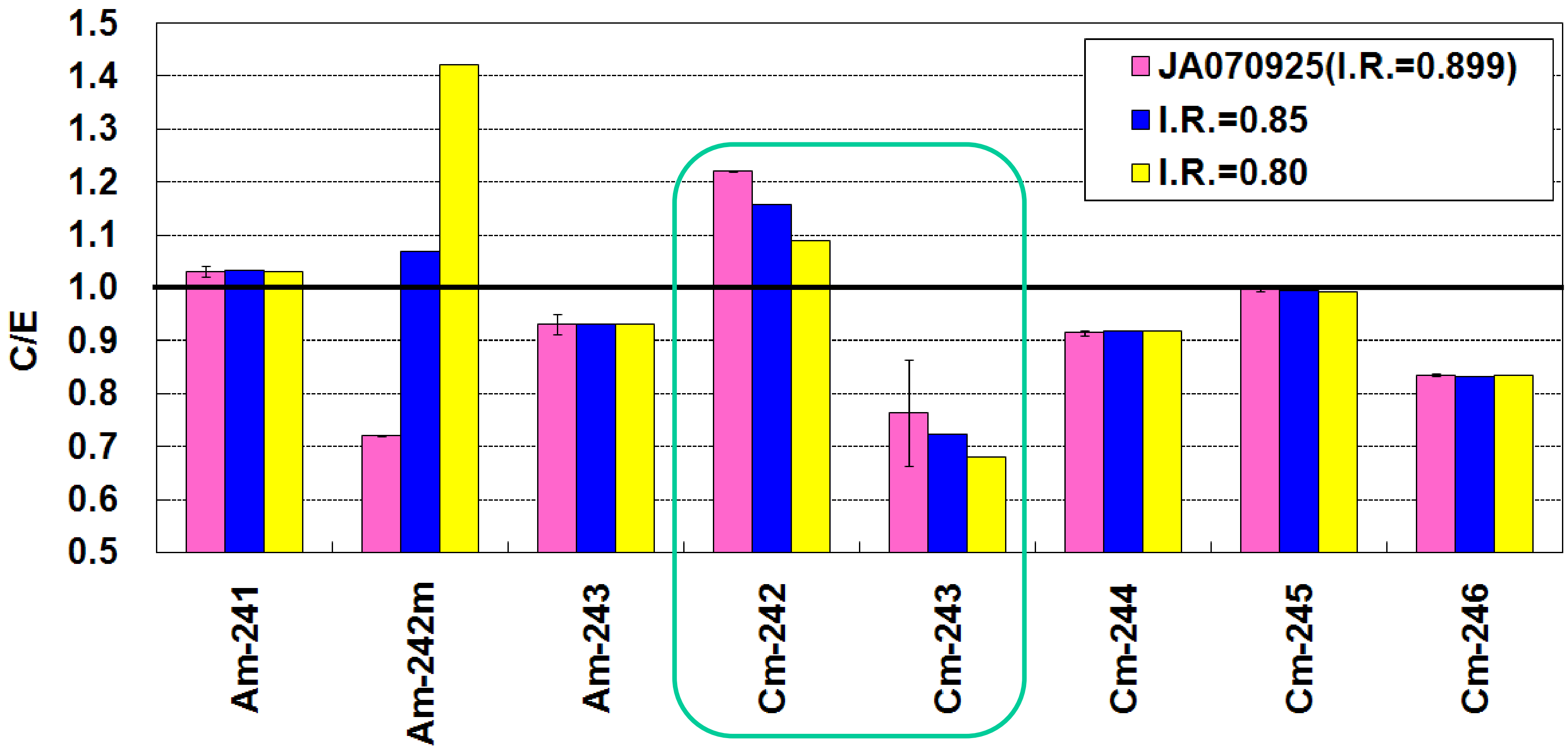
Capture Reaction Rate of Am241 in Typical LWR



One-group Isomeric Ratio of Am241(n,γ)

J33	B68	B70	F31	JA070925
0.877	0.885	0.898	0.873	0.899

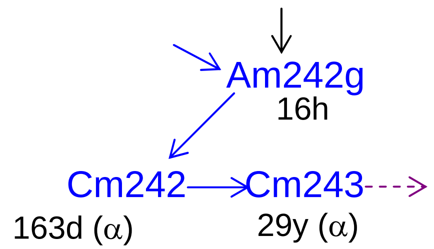
Sensitivity on Isomeric Ratio of Am241 Capture



I.R.=0.86～0.87

(about 0.88 in JA071122)

σ_a



Conclusion

Good performance of JENDL/AC for various types of reactors was confirmed by comparison with the results of other recent nuclear data files, JENDL-3.3, JEFF-3.1, and ENDF/B-VII.0.

However, further investigation is recommended for:

- Criticality of PU-SOL-THERM system,
- Criticality of U233-SOL-INTER system,
- Generation of Cm-242 and Cm-243 in the LWR spent fuel.

PC Cluster of
Reactor Physics Group

Altix3700Bx2/2048CPU