Integral Test of JENDL-3.3 for Thermal Reactors

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Criticality benchmark testing was carried out for 59 experiments in various thermal reactors using a continues-energy Monte Carlo code MVP and its different libraries generated from JENDL-3.2, JENDL-3.3, JEF-2.2 and ENDF/B-VI (R8). From the benchmark results, we can say JENDL-3.3 generally gives better k_{eff} values compared with other nuclear data libraries. However, further modification of JENDL-3.3 is expected to solve the following problems: 1) systematic underestimation of k_{eff} depending on ^{235}U enrichment for the cores with low (less than 3wt.%) enriched uranium fueled cores, 2) dependence of C/E value of k_{eff} on neutron spectrum and plutonium composition for MOX fueled cores. These are common problems for all of the nuclear data libraries used in this study.

1. INTRODUCTION

The latest version of Japanese Evaluated Nuclear Data Library (JENDL-3.3) [1] was released in May, 2002. When it was completed, a criticality benchmark testing [1] of JENDL-3.3 had bee carried out for various types of fast and thermal reactors by Reactor Integral Test Working Group in JNDC. As far as thermal reactors, however, further test is necessary, because the previous benchmark calculation was made only for 13 experiments carried out in JAERI facilities TCA, STACY, TRACY and JRR4. Especially for UO_2 or MOX fueled thermal-neutron cores, the test was done only for TCA experiments. Although the lattice pitches and loading patterns can be varied in TCA, fuels are limited to 2.6wt.% enriched UO_2 and 3.0wt.% MOX. Thus integral test covering more cores with different fuel specifications are required. In the present study, results of the extended integral test of JENDL-3.3 by using a continuous-energy Monte Carlo code are shown for totally 59 experiments in various thermal reactors including the previous benchmark cores. Results with JENDL-3.2[2], JEF-2.2 [3], and ENDF/B-VI (R8) [4] are also shown for comparison.

2. BENCHMARK CORES

Table 1 shows the benchmark cores selected in this study. Most of them are light water moderated UO_2 or MOX fueled uniform lattices at room temperature. Exceptions are as follows:

- Fuel material of TRX is metal uranium (Al clad cylindrical rod),
- Temperature of KRITTZ-2 in hot condition is about 245°C,

- Fuels of STACY and TRACY are homogeneous uranyl nitrate solutions,
- JRR4 is a light water moderated research reactor which uses MTR type fuel elements. In 1998, the fuel material of JRR4 was changed from 93% enriched U-Al alloy to 20% enriched U_3Si_3-Al dispersed alloy. Two minimum critical cores (JRR4-U93 and JRR4-U20) at room temperature with the above different fuels are selected in this study,

Except for JRR4, TRACY and MISTRAL, the benchmark problems are taken from public international benchmark literatures, in which details of experimental procedures, material compositions, calculation modeling and evaluated benchmark uncertainties are described. See the literatures given in Table 1 for details.

Table 1 Benchmark cores

Lattice names	H/HM	Fuel (Enrichment)	Remarks	Ref.
TRX-1	3.3	Metal-U (1.3 % ^{235}U) , Al cladding	Triangular pitch=1.806cm, 764 rods	[5]
TRX-2	5.6		Triangular pitch=2.174cm, 578 rods	
KRITZ2:1 Cold	3.4	UO_2 (1.86% ^{235}U) Zry-2 cladding	Pitch=1.485cm, 44x44 rods, Temp.=19.7°C, Boron: 218ppm, Hc=65.28cm	[6] [7]
KRITZ2:1 Hot	2.8		Pitch=1.485cm, 44x44 rods, Temp.=248.5°C, Boron: 26ppm, Hc=105.5cm	
KRITZ2:13 Cold	5.0		Pitch=1.635cm, 40x40 rods, 22.1°C, Boron: 452ppm, Hc=96.17cm	
KRITZ2:13 Hot	4.1		Pitch=1.635cm, 40x40 rods, 243.0°C, Boron: 280ppm, Hc=110.9cm	
B&W-Core XI	5.4	UO_2 (2.5% ^{235}U) Al cladding	LEU-COMP-THERM-008-001* 4961 rods, Boron: 1511ppm	[8]
TCA-1.50U (3 cases)	4.3	UO_2 (2.6% ^{235}U) Al cladding	LEU-COMP-THERM-006-001～003* Pitch=1.849cm, 19x19,20x20,21x21 rods	[8]
TCA-1.83U (5 cases)	5.3		LEU-COMP-THERM-006-004～008* Pitch=1.956cm, 17x17,18x18,19x19,20x20,21x21	
TCA-2.48U (5 cases)	7.2		LEU-COMP-THERM-006-009～013* Pitch=2.150cm, 16x16,17x17,18x18,19x,19,20x20	
TCA-3.00U (5 cases)	8.6		LEU-COMP-THERM-006-014～018* Pitch=2.293cm, 15x15,16x16,17x17,18x18,19x19	
DIMPLE3	3.0	UO_2 (3.0% ^{235}U)	LEU-COMP-THERM-048-001*	[8]
MISTRAL Core 1	5.1	UO_2 (3.7% ^{235}U)	Boron: 300ppm	[9] [10]
DIMPLE7	8.4	UO_2 (7.0% ^{235}U)	LEU-COMP-THERM-018* Pitch=1.32cm, 376 rods, Hc=53.9cm	[8]
STACY (7 cases)	from 73 to 103	uranyl nitrate solution (10% ^{235}U)	LEU-SOL-THERM-005-001～007* 7 cases of solution fueled cores with different uranium concentrations from 225 to 310 gU/liter in a water reflected 60cmϕ cylindrical tank	[8]
TRACY	52	uranyl nitrate solution (10% ^{235}U)	Solution fuele in a 50cmϕ cylindrical tank with a channel for a transient rod (out of core) Run-64, Temp.=27.8C, 430gU/l, Hc=45.3cm	[11]
JRR4-U20	-	U_3Si_3-Al dispersed alloy (20% ^{235}U)	Minimum critical core with 12 MTR type fuel elements	[12]
JRR4-U93	-	U-Al alloy (93% ^{235}U)	Minimum critical core with 12 MTR type fuel elements	
KRITZ2:19 Cold	10.4	MOX (1.5% Pu-t) $^{239}Pu/Pu$=0.914	Pitch=1.80cm, 25x24 rods, Temp.=21.1°C, Boron: 4.8ppm, Hc=66.56cm	[6] [7]
KRITZ2:19 Hot	8.5		Pitch=1.80cm, 25x24 rods, Temp.=235.9°C, Boron: 5.2ppm, Hc=104.2cm	

TCA-2.42PU (3 cases)	12.0	MOX (3.0% Pu-t) Pu-fiss./Pu～0.75	MIX-COMP-THERM-004-001～003* Pitch=1.825cm, 23x23 rods, Date:1972-1974	[8]
TCA-2.98PU (3 cases)	14.8		MIX-COMP-THERM-004-004～006* Pitch=1.956cm, 21x21 rods, Date:1972-1975	
TCA-4.28PU (3 cases)	21.1		MIX-COMP-THERM-004-007～009* Pitch=2.225cm, 20x20 rods, Date:1972-1974	
TCA-5.50PU (2 cases)	27.6		MIX-COMP-THERM-004-010～011* Pitch=2.474cm, 21x21 rods, Date:1972-1973	
CRX-Case 1	4.9	MOX (6.6% Pu-t) ^{239}Pu/Pu=0.906	MIX-COMP-THERM-003-001* Pitch=1.3208cm, 23x22 rods, Hc=82.90cm	[8]
CRX-Case 2	6.4		MIX-COMP-THERM-003-002* Pitch=1.4224cm, 19x19 rods, Hc=81.295cm	
CRX-Case 3	6.4		MIX-COMP-THERM-003-003* (Boron:337ppm) Pitch=1.4224cm, 21x21 rods, Hc=88.06cm	
CRX-Case 4	13.8		MIX-COMP-THERM-003-004* Pitch=1.8679cm, 13x13 rods, Hc=68.41cm	
CRX-Case 5	16.7		MIX-COMP-THERM-003-005* Pitch=2.01158cm, 12x12 rods, Hc=76.76cm	
CRX-Case 6	31.6		MIX-COMP-THERM-003-006* Pitch=2.6416cm, 11x11 rods, Hc=79.50cm	
MISTRAL Core 2	5.1	MOX (7.0% Pu-t)	Pitch=1.32cm, Boron: 0ppm	[9]
MISTRAL Core 3	6.0		Pitch=1.39cm, Boron: 230ppm	[10]

* Benchmark identification numbers in the "Handbook of International Criticality Safety Benchmark Evaluation Project (ICSBEP)"[8],
H/HM: ratio of atomic number densities of hydrogen and all heavy metal nuclides in fuel, Pitch: rectangular lattice pitch, NxN: loading pattern of fuel rods in rectangular lattice, Hc: critical water height, Date: measurement date in Pu core, Temp: system temperature, Boron: boron concentration in water moderator if any

3. MVP CALCULATION

A series of benchmark calculations was performed by using a continuous-energy Monte Carlo code MVP [13, 14] and its four different nuclear data libraries generated from JENDL-3.2, JENDL-3.3, JEF-2.2 and ENDF/B-VI(R8). The library generation was performed with the LICEM code system [15, 16]. The thermal scattering law data S(α,β) for the JENDL-3.3 and ENDF/B-VI(R8) libraries were taken from ENDF/B-VI, while the data for the other libraries were taken from ENDF/B-III. The resonance shielding effects in unresolved resonance region was treated by the probability table.

In each of the MVP calculations, the first 30 cycles were skipped, followed by 1,000 active cycles, each with 10,000 particles per cycle. Statistical errors (1σ) of k_{eff} values are within the range from 0.00015 to 0.00025.

As far as the MISTRAL cores are concerned, MVP results are referred from the literatures [9, 10] published by NUPEC members, because detailed information to construct 3D modeling of the MISTRAL cores are not opened. The JENDL-3.3 result for the MISTRAL cores has not been reported yet.

4. RESULTS AND DISSCUSSIONS

(1) Uranium fueled cores

Figure 1 shows the C/E values of k_{eff} for the uranium fueled benchmark cores. In this figure, the values for the TCA cores (TCA1.50U, TCA-1.83U, TCA-2.48U and TCA3.00U) are the averages for the three or five experiments in which fuel loading patterns (See in Table 1) are different but the lattice pitches are the same. The averaging was done because meaningful differences or systematic tendency were not observed for the C/E values among the experiments. The value for the STACY core is also averaged one for the seven experiments in which uranium concentration are different in the range specified in Table 1. In Fig.1, benchmark cores are lined up in the order of ^{235}U enrichment and in the order of H/HM values for the cores with the same enrichment.

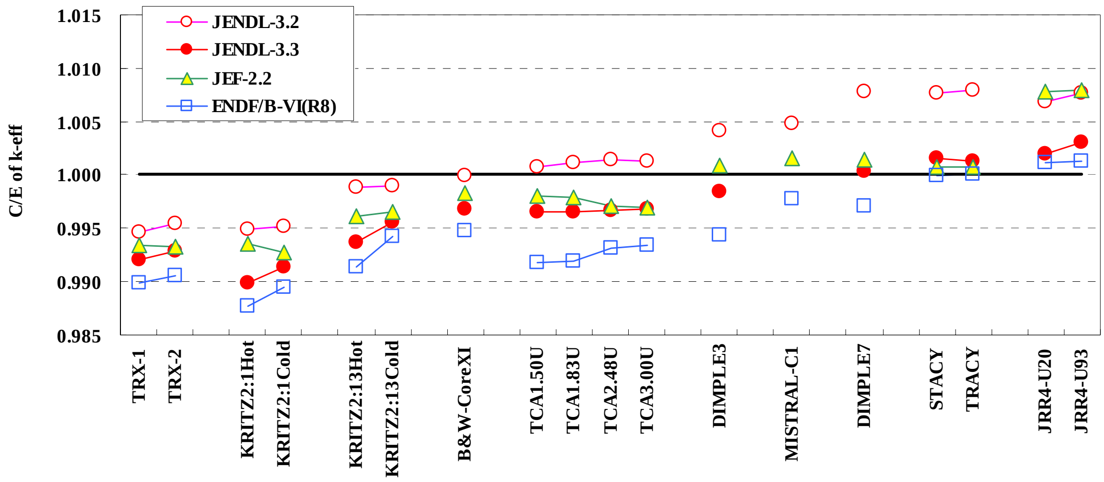

Fig.1 C/E values of k_{eff} for uranium fueled benchmark cores

From Fig.1, the followings are observed:

- The calculated k_{eff} values are larger in the order of the JENDL-3.2, JEF-2.2, JENDL-3.3 and ENDF/B-VI (R8) results.
- JENDL-3.2 overestimates criticality by about 0.5%Δk or more for the core in which ^{235}U enrichment is higher than 3.0wt.% (e.g. MISTRAL-C1, DIMPLE-3/-7, STACY, TRACY, JRR4), whereas it underestimates criticality of the core in which the enrichment is lower than 2.0 wt.% (e.g. TRX-1/-2, KRITZ2:1, KRITZ2:13).
- In the JENDL-3.3 results, the overestimations observed in the JENDL-3.2 results are improved. This is mainly due to the modification [1] of thermal cross section data of ^{235}U. For the STACY and TRACY results, modification of thermal cross section of ^{14}N(n,p) is also contributing to the improvement by about 0.2%Δk. [17]
- For the cores with relatively lower ^{235}U enriched fuels, all nuclear data libraries give underestimated results. It is significant especially for ENDF/B-VI (R8) which gives lower k_{eff}, compared with other libraries. The underestimation depends on ^{235}U enrichment systematically.
- The KRITZ2 benchmark gives information on the prediction accuracy for total temperature coefficient. From the difference of the C/E values among hot and cold conditions in KRITZ2:1 and KRITZ2:13, JENDL-3.2 gives the most accuracy result.

(2) MOX fueled cores

Table 2 shows the plutonium compositions of the MOX fueled benchmark cores. For the MISTRAL cores, it is reported that the plutonium fuel has regular plutonium composition. The KRITZ2:19 and CRX benchmarks are useful to test cross section data of ^{239}Pu, because ^{239}Pu contents in these cores are more than 90wt.% and the reactivity contribution of higher-order plutonium isotopes and ^{241}Am is small. In addition, the two experiments in KRITZ2:19 and six experiments in CRX were performed within four and three months, respectively. Therefore, plutonium aging effects, which is reactivity loss due to decay from ^{241}Pu to ^{241}Am with half-life of 14.4 years, can be neglected in these benchmarks. On the other hand, the TCA-MOX and MISTRAL benchmarks are important because their fuel compositions are similar to those of MOX fueled LWRs. In these cores, plutonium aging effects should be taken into account.

Table 2 Composition of Pu composition (wt.% including Am241)

Lattice name	Pu238	Pu239	Pu240	Pu241	Pu242	Am241
KRITZ2:19	-	91.4	7.9	0.4	0.03	0.3
CRX	-	90.5	8.6	0.8	0.04	0.1
TCA-4.24PU*	0.5	68.1	22.0	7.1	2.0	0.3
TCA-2.98PU**	0.5	68.1	22.0	6.1	2.0	1.3

*oldest (13, Apr., 1972) and ** latest (21, May, 1975) experiments in the TCA-MOX benchmark

At first, time dependence of the C/E values was investigated for eleven experiments in the TCA-MOX benchmark. The MVP results with JENDL-3.2 and JENDL-3.3 are shown in Fig. 2. In the JENDL-3.2 results, the C/E values have a tendency to increase slightly as time passes. It was improved in the JENDL-3.3 results due to the modification [1] of ^{241}Am capture cross section in JENDL-3.3.

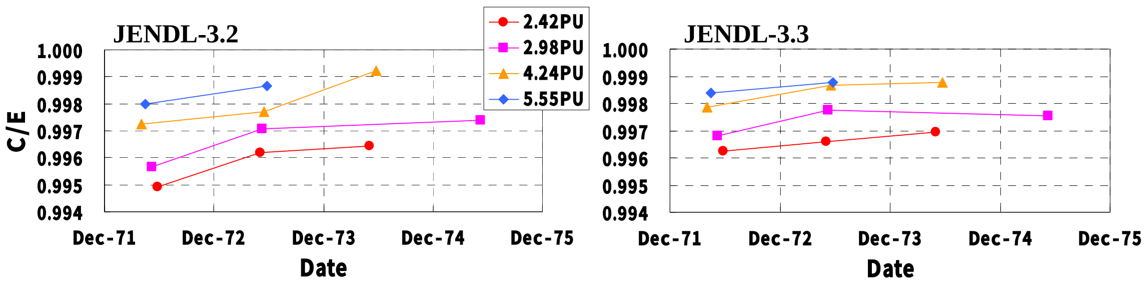

Fig.2 Time dependence of C/E values (k_{eff}) for MOX fueled TCA cores

Figure 3 shows the C/E values of k_{eff} for all of the MOX fueled benchmark cores. In this figure, the values for the TCA cores (2.42PU, 2.98PU, 4.28PU, 5.50PU) are not averaged ones but values for the 4 experiments measured within 24 days. This is for excluding the plutonium aging effect. From Fig.3, the followings are observed:

- Remarkable differences are not observed between the k_{eff} values of JENDL-3.2 and JENDL-3.3 results. The k_{eff} values obtained with JEF-2.2 and ENDF/B-VI(rev.8) are smaller than those with JENDL-3.2 and JENDL-3.3.

- From the results for the TCA and CRX cores, where H/HM values varied from about 5.0 to 30.0, it is said that the C/E values depend on neutron spectra in MOX fueled cores. The dependency seems to be attributed to ^{239}Pu cross section data, because the dependency is observed in CRX, where contributions of higher-order plutonium isotopes and ^{241}Am are small in CRX.
- From the comparison between the KRITZ2:19 and CRX results, significant dependency of the C/E values on plutonium enrichment is not observed in the MOX cores with high ^{239}Pu content. On the other hand, the C/E values are quite different between the results for TCA and MISTRAL, in which MOX fuels have more contents of higher-order plutonium and ^{241}Am, compared with KRITZ2 and CRX cores.
- In the KRITZ2 results, difference of the C/E values between hot and cold conditions is almost equivalent to those in the UO_2 fueled KRITZ2 cores. The differences are almost same among the results with the different nuclear data libraries.

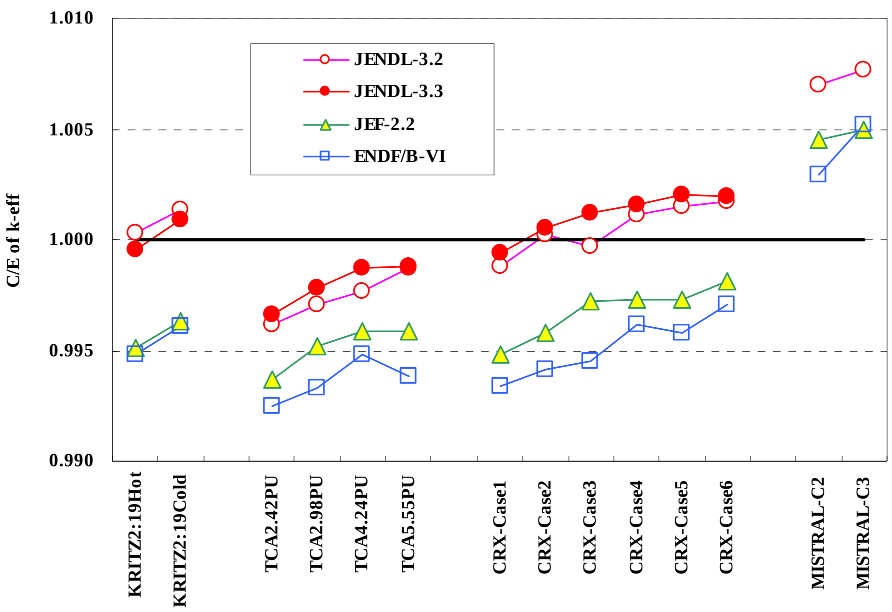

Fig.3 C/E values of k_{eff} for MOX fueled benchmark cores

5. CONCLUSIONS

Criticality benchmark testing of JENDL-3.3 was performed for 59 experiments in various thermal reactors. From the benchmark results, we can say JENDL-3.3 generally gives better k_{eff} values, compared with JENDL-3.2, JEF-2.2 and ENDF/B-VI (R8). However, further modification of JENDL-3.3 is expected to solve the following problems: 1) systematic underestimation of k_{eff} depending on ^{235}U enrichment for the cores with low (less than 3wt.%) enriched uranium fueled cores, 2) dependence of C/E value for k_{eff} on neutron spectrum and plutonium composition for MOX fueled cores. These are common problems for all of the nuclear data libraries used in this study.

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