JENDL Reactor Constant and its Application

:::::::: The 40th Anniversary of Japan Nuclear Data Committee :::::::

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Abstract

The status of reactor constants about 27 years ago is briefly reviewed from the criticality predictions and nuclear data processing codes. In the second section, status of current users of JENDL-3.3 and/or JENDL-3.2 is consulted with the 2003 Fall Meeting of Atomic Energy Society of Japan. In the third section, the reliabilities of JENDL-3.3 and -3.2 are reviewed mainly from the application to light water reactor (LWR) mockup experiments; MISTRAL and BASALA made on EOLE critical facility of Cadarache Laboratory in France, since an extensive evaluation for nuclear data applicability to LWR have been scarcely performed in relative to for FBR. The results of international benchmark cores and criticality safety analyses are briefly reviewed. In the concluding remarks, overall applicability is shown as a summary with respect to all reactor parameters obtained in the LWR mockup experiment and some remarks are noted.

I. A Glance of the History of Reactor Constants

The activity of the 40 years of JNDC should be faithfully celebrated and highly appreciated, and I would say ”Congratulations!, so long time for nuclear data”. In this period, some files in the world ended and some new ones like CENDL were born. Our JENDL is a highly qualified and big file compared to ENDF/B and JEFF. The contents of these files have been significantly enriched, and currently the energy range is extending so as to dealing with the neutronics of ADS and the other application is for astrophysics.

On the other side, reliability of reactor constants is increasing together with the brushing up of the evaluated nuclear data files. Following two examples show a typical status of reactor constants used for nuclear performance parameters about 30 years ago.

(a): International Intercomparison

International intercomparison of LLMFBR nuclear characteristics[1)] had been conducted by ANL. Sixteen participants, as shown below, had calculated overall nuclear performance parameters of large sodium cooled fast breeder reactor by using their own libraries. The results were intercompared and discussed at ANL(Chicago) (1978).

USA(ANL)[ENDF/B-VI], USA(HEDL)[ENDF/B-VI], Belgium[KEDAK-2], France[CARNAVAL-III,IV], Italy[ENDF/B-IV], Switzerland[ENDF/B-III,-VI], Japan[JENDL,JAERI-FAST-2,GJAERI-FAST-2(25)], Germany[KEDAK-3,KFKINR], Sweden[ENDF/B-III], England(UKAEA)[FGL-5], USSR[BNAB-70]

where Country[data set], and the mean k_{eff}-value, $\overline{k}_{eff}$, and its standard deviation was

$\overline{k}_{eff} = 1.00000 \pm 0.01292$,
with the maximum $(k_{eff} - \overline{k}_{eff}) = 0.151\%$ and minimum $(k_{eff} - \overline{k}_{eff}) = -2.458\%$.

The standard deviation of ± 0.01292 is seemed to be about twice of current deviation even if similar comparison was taken place at present.

(b): MOZART Project

The first LLMFBR Mockup experiment in JAPAN, namely MOZART(**Mo**nju **Z**ebra **A**ssembly **R**eactor **T**est)[2] project had been performed in ZEBRA critical assembly of Winfrith (UK). The MOZART experiment analyses were made by six domestic participants with their libraries; JAERI[JAERI-FAST], Hitachi[HIM-2], Toshiba[NNS-2], Mitsubishi[MICS-V/2], Fuji[FX-2] and Sumitomo[SEAI]. The JAERI-Fast set was based on their own evaluated nuclear data, but the other five domestic libraries were so-called modified Abagyan Set since at that time the text book for this library was available in JAPAN and it had been modified taking into account ENDF/B-I with ETOX nuclear processing code. The final recommended k_{eff} was

$$k_{eff}\ (C/E) = 0.9973 \pm 0.0033 \quad \text{for MZA Core (approximately spherical core),}$$
$$k_{eff}\ (C/E) = 0.9972 \pm 0.0034 \quad \text{for MZB Core (cylindrical core for MONJU physics mockup).}$$

These C/E-values are significantly close to unity within about ± 0.3% which are better than those of current benchmark test and experiment analyses mentioned later. For the MOZART case, consistency between mockup and calculated cores was deeply investigated and twelve corrections were applied.

Comparing the (a)'s k_{eff}-values with the current results shown by Figs. 1 and 3, the reliability of k_{eff} prediction has been improved with time. The trend approaching forward unity, however, seems not to be monotonous, and some fluctuation in the process of updating version is found.

II. JENDL-3.3 Users

1. Users in the 2003 Fall Meeting of the Atomic Energy Society of JAPAN

The users in the reactor physics session of the 2003 Fall Meeting[3] were consulted and the following statistic was obtained as

Table 1 JENDL-3.3 and/or JENDL-3.2 Users

Nuclear Data	Presentations	Fraction (%)	Note
JENDL-3.3 and -3.2	4	29	Mainly MISTRAL, BASALA and BFS analyses.
JENDL-3.2	9	64	PIE data analysis included, see summary.
ENDF/B-VI	1	1	BWR core simulation: 3 group kinetics model.

where SRAC and/or MVP libraries are used except ENDF/B-VI case. JENDL-3.3 reactor constants are not yet widely used since SRAC and MVP libraries have been recently released.

2. Users in FBR project

JNC provides multi-group cross section library based on JENDL-3.3 and -3.2 for FBR Project. They say the 900-group constant is available ,where hyperfine group effect is taken into account by bi-mixture slowing down code PEACO. Besides, they have a unified cross section set ADJ200R based on the cross section adjustment.

Therefore, as long as FBR project, only one reactor constant set given by JNC is commonly used. The newest version of JENDL-3.3 will be cited for their future library although the current adjustment was applied for JENDL-3.2.

3. Users in LWR project

No standard or common used reactor constant, at present, is available for LWR reactor, but as shown in section III.2 SRAC and MVP with their cross section libraries have been commonly used like a standard library in many universities and organizations such as NUPEC (Nuclear Power Engineering Corporation). Analyses of LWR mockup experiments MISTRAL and BASALA[4,5] of Section III are typical examples for SRAC and MVP.

Individual cross section library used for BWR, PWR core designs and fuel management seems to be produced from JENDL-3.3 or -3.2 file but detailed information isn't opened.

III. Applicability of JENDL Reactor Constants

1. Benchmark Test

Criticality benchmark test for well known fourteen cores were made by Takano et al,[6] where continuous energy Mote Carlo code MVP with JENDL-3.3 nuclear data was used. Resultant k$_{eff}$'s are shown in Fig.1.

The calculated to experiment ratio (C/E) and their standard deviation is approximately

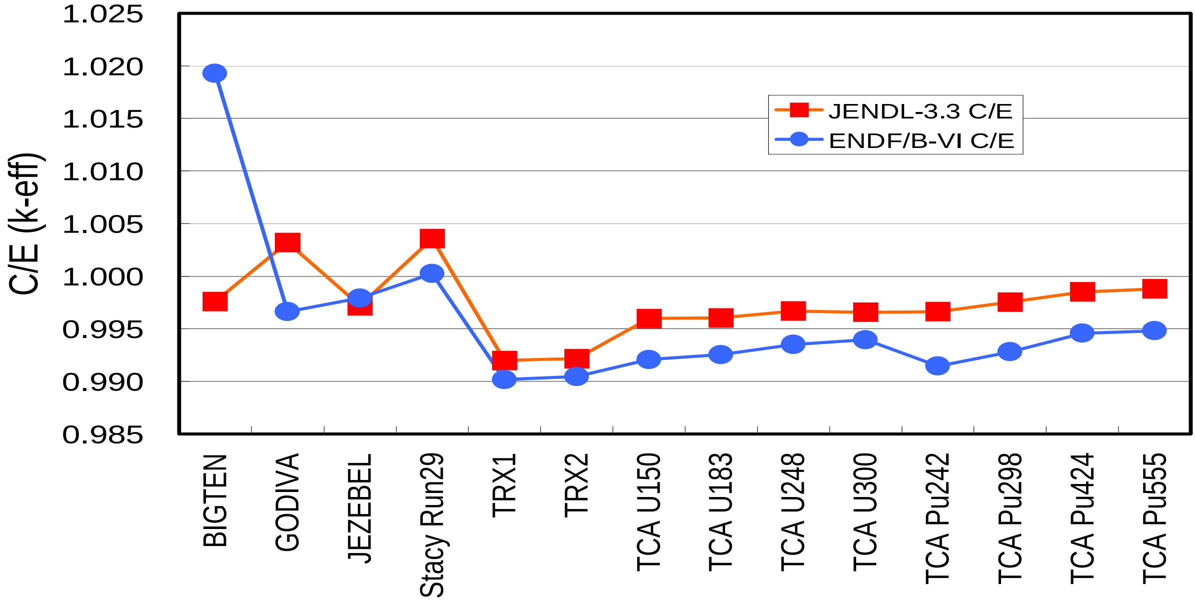

Fig. 1 The C/E values for thermal cores calculated with JENDL-3.3 and ENDF/B-VI.5

$$k_{eff} \quad C/E - value \simeq 1.000 \pm 0.005$$

where the TRX-1 and -2 cores are excluded since their k$_{eff}$'s may be improved by more careful treatment of ^{238}U cross sections by taking into account hyperfine group effect. Therefore, the current reliability of JENDL-3.3 for criticality may be within $\pm$0.005.

2. MISTRAL and BASALA MOX Physics Mock-up Experiment

(1) Core Specifications in MISTRAL and BASALA MOX Physics Mock-up Experiments

Some mockup experiments focusing on the fast reactor such as MONJU had been carried out for overall core performance parameters so far. For light water reactors, however, such mockup experiment had never been extensively performed, although well known benchmark cores like those used by H. Takano et al[6] as well as experiments for a few parameters in TCA or FCA of JAERI are available

Recently, MOX core physics mock-up experiments MISTRAL and BASALA[4,5] had been doned in the EOLE Light Water Critical Facility at Cadarache Lab. of CEA, France, where overall nuclear performance parameters as shown in Table 2 are measured. For instance, reactivity worth of a cruciform

BWR control blade shown in Fig.2 as well as effective delayed neutron fraction β_{eff} were obtained.

The analyses have been made by NUPEC with four organizations, and serious evaluation of the applicability of nuclear data and codes to LWRs fueled with partial or full MOX has been made. The detailed C/E-values of overall nuclear performance parameters are shown in detail in references 4) and 5), and thus the results are shortly summarized in Table 3 where results of criticality safety[7)] and PIE data analysis[3)] are shown together.

Table 2 Nuclear Performance Parameters measured in MISTRAL and BASALA Cores[4, 5)]

Item	MISTRAL Core				BASALA Core	
	Core 1	Core 2	Core 3	Core 4	Core 1	Core 2
Core Specification	UO2 -Homo.	MOX -Homo.	MOX -Homo.	PWR Mockup	BWR Mockup	BWR Mockup
Water to Heavy Metal(H/HM)	5.1	5.2	6.2	5.8	5.0	9.0
Water to Fuel Vol.frac.	1.8	1.8	2.1	2.0	1.7	3.1
Fuel pitch(cm)	1.32	1.32	1.39	1.32	1.13	1.35
Enrichment (to Total Pu)	UO2 3.7%	MOX 7.0%, etc.	MOX 7.0%	MOX 7.0%	MOX 7%, etc.	MOX 7%, etc.
(1): Critical Mass	○	○	○	○	○	○
(2): Boron Concentration	○	○	○	○	—	—
(3): Buckling	○	○	○	—	—	—
(4): Spectrum Index	○	○	○	—	—	—
Power Distribution						
(5): Radial	○	○	○	○	○	○
(6): Axial	○	○	○	○	○	○
(7): Isothermal Temp. Coeff.	○	○	○	—	—	○
Reactivity Worth						
(8): Boron	○	○	○	○	—	○
(9): Absorvers	○ (4 Types)	○ (4 Types)	○ (2 Types)	—	—	—
(10): Burnable Poison	—	—	—	—	○	○
(11): Cluster Control Rod	—	—	—	○	—	—
(12): Criciform Control Blade	—	—	—	—	—	○
(13): Two Dimensional Void	—	—	○	—	○	—
(14): Water Hole	○	○	—	—	—	—
(15): Water Rod	—	—	—	—	○	—
(16): β_{eff}	○	○	—	—	—	—

○ : Measurement was performed.

(2) Analyses of MISTRAL and BASALA Experiments

Nuclear data used for analyses are mainly JENDL-3.2 and -3.3, and partially ENDF/B-VI and JEF-2.2. The application of JENDL-3.3 is also limited to k_{eff} and spectrum index since the release of the JENDL-3.3 cross section libraries was on the way of experiment analyses.

Core and/or cell calculation codes are SRAC diffusion (CITATION) and transport (TWOTRAN) routines, and continuous energy Monte Carlo code MVP was also used as shown in Table 3. Thermal cut-off energy of 1.82 eV is optimized so as to take into account ^{240}Pu resonance and effective energy range of up-scattering.

Geometrical calculation models for MVP were as built as experiment. For analysis by SRAC, two dimensional XY geometry for a quarter of core is basically used when the 1/4 symmetry exits, while the two dimensional XY geometry for full core was employed for non-1/4 symmetry core. Then, experimental axial buckling was applied to Z-axes, and sixteen group cross section set was prepared by group collapsing from PEACO hyperfine group cross sections. For integral boron worth, three dimensional XYZ geometery for a quater of core was adopted.

Table 3 Nuclear Data and Reactor Constant

a)	Nuclear Data: JENDL-3.2 and Partially JENDL-3.3
b)	Group Constant SRAC 107 group set $\Longleftarrow$ Spectrum Index(5X5Rods) 16 group set condensed from PEACO hyperfine group cross sections and neutron spectra $\Longleftarrow$ Single Ros Cell: Fuel+Clad+Overclad+Moderator Thermal cut-off=1.82 eV
c)	Cell Calculation SRAC 107 group set Pij-method
d)	Core Calculation SRAC-CITATION -TWOTRAN MVP: History: $10000 \times 1000 = 10^7$, Criticality: $\frac{\Delta k}{k} = 0.02\%$, Power Distribution: $\frac{\Delta P}{P} = 1\%$

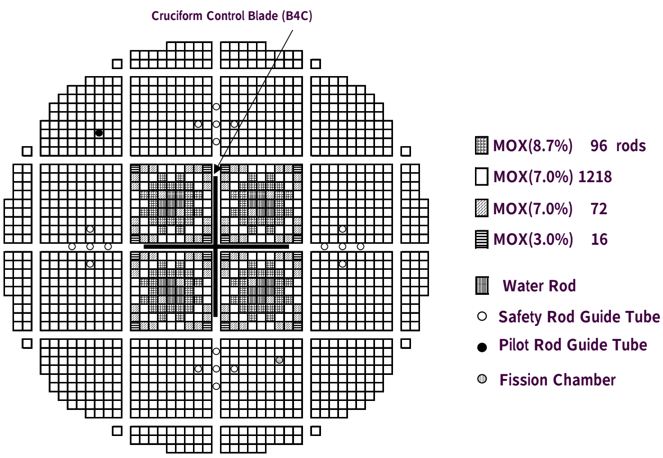

Fig. 2 BASALA Core 2 with Cruciform Control Blade for BWR

(3) Results of MISTRAL Experiment, BASALA Experiment, Criticality Safety and PIE Data Analyses

The results of experiment analyses are shown in references 4) and 5) in detail, and a short summary is shown in Table 4 where discrepancies in unit of experimental error are shown. Only k$_{eff}$ trends are shown in Fig. 3 for individual critical core and five nuclear data files.

As shown in Fig. 3, the k$_{eff}$-values for MOX cores are increasing with core from about 1.0032 of MISTRAL Core 4 (UO2-REF) to 1.0074 of BASALA Core 1, although k$_{eff}$ for UO_2 core is around unity. The increment of k$_{eff}$ is expected that the transmutation effect from ^{241}Pu with half-life $= 14.290$ y to ^{241}Am. This effect has been already corrected for Fig. 3. The effect, however, could not flatten the k$_{eff}$ trend. Further study has been continued and thus the reason of increment will be made soon clear.

In order to interpret above increasing trend, the transmutation effect to the neutron balance is intuitively formulated and the following expression can be obtained

$$\frac{\delta keff}{keff} = \left\{ \frac{\nu\sigma_f^{Am241} - \nu\sigma_f^{Pu241}}{\nu\sigma_f^{Pu241}} \cdot \frac{<\nu\Sigma_f\phi>^{Pu241}}{<\nu\Sigma_f\phi>} - \frac{\sigma_a^{Am241} - \sigma_a^{Pu241}}{\sigma_a^{Pu241}} \cdot \frac{<\Sigma_a\phi>^{Pu241}}{<\Sigma_a\phi>} \right\}$$
$$\times \left\{1 - exp(-\lambda^{Pu241} t)\right\}; \qquad (1)$$

where λ^{Pu241} means ^{241}Pu decay constant for half-live=14.290 y, $<\nu\Sigma_f\phi>=$ Total neutron production reaction rate and $<\nu\Sigma_f\phi>^{Pu241}=$ ^{241}Pu neutron production reaction rate. Eq.(1) means that the changes of effective neutron production and absorption cross sections take place between ^{241}Am and ^{241}Pu, and their effect to k$_{eff}$ is given by the multiplications of fractional isotopic reaction rates, and the second term with the decay constant of ^{241}Pu is increasing function, similar trend to Fig. 3, with respect to time t in the range from 0(EPICURE MH1.2) to about 7(BASALA Core1) years. In the MISTRAL and BASALA experimental analyses, direct k-difference method was used instead of this simple expression.

The k$_{eff}$ for criticality safety is shown in Table 4, k$_{eff}$ $= 1.0007 \pm 0.0011$ for 39 LEU-SOL-THERM Benchmark Configuration and , k$_{eff}$ $= 0.9980 \pm 0.0019$ for 24 LEU-COMP-THERM Benchmark Configuration, respectively, i.e. the criticality is fairly well predicted within $\pm 0.2\%$.

As mentioned previously, JENDL3.3 had been used only for the k$_{eff}$'s for all cores and the spectrum indexes of the MISTRAL cores because of the release time of SRAC and MVP libraries. For the spectrum index, the fission reaction rate ratio of U-238 to U235, denoted by F28/F25, is about 28% underestimation ($= 0.72 \pm 1.8\%$) and similarly for those of Pu240 to Pu239 (F40/F49) is about 11% underestimated ($= 0.89 \pm 2.8\%$) beyond the experimental errors, In general such a threshold reaction is generally difficult to enter into an agreement with the experimental data because it is significantly affected by the higher energy flux, while the other non-threshold reaction are predicted within experimental errors.

As summarized in Table 4, the other nuclear performance parameters are well predicted by JENDL-3.3 or -3.2 nuclear data. Typical example is the reactivity worth of cruciform B_4C control blade for BWR whose (C/E-1)-value is within experimental error (6%), and also the effective delayed neutron fraction is fairly well predicted.

The analysis of PIE data for a BWR core in FUKUSHIMA site gives well agreement of atomic

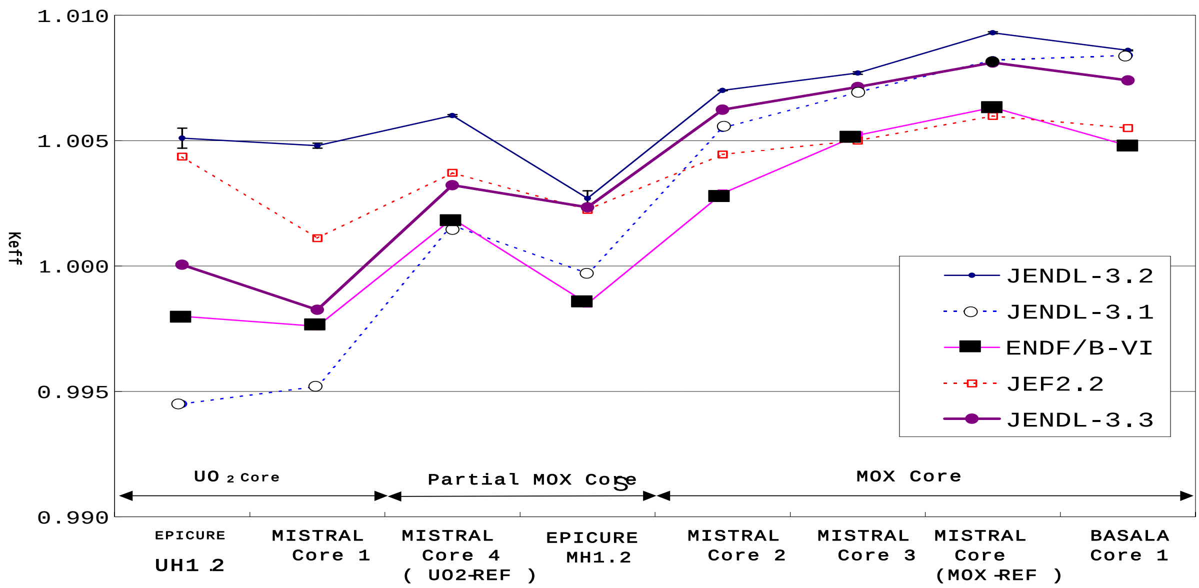

Fig. 3 Criticalities of EPICURE, MISTRAL and BASALA Cores

number densities with the experimental data except ^{241}Am.

Table 4 Summary of Benchmark Test, MISTRAL, BASALA, Criticality Safety and IPE Data Analysis

No.	Quantity	Reliability
(1)	Criticality	—(J3.3) Takano Benchmark Test: K$_{eff}$ C/E$\simeq 1.0000 \pm 0.005^{*)}$ —(J3.3) MISTRAL and BASALA: K$_{eff}$ C/E$\simeq 1.0000^{+0.0081^{*)}}_{-0.0017}$,where the lowest and highest dicrepancies are shown as errors. —(J3.3) Criticality Safety: K$_{eff}$ $= 1.0007 \pm 0.0011$ for 39 LEU-SOL-THERM, K$_{eff}$ $= 0.9980 \pm 0.0019$ for 24 LEU-COMP-THERM
(2)	Boron Concentration	Integral Worth $\lvert C/E-1\rvert \leq$Exp. error$\simeq 5.3\%$ except Core 4 with 10% overestimation
(3)	Spectrum Index	—(J3.3) $\lvert C/E-1\rvert \leq$Exp. error ($2.4 \sim 7.6$%), except F28/F25= $0.72 \pm 1.8^{\dagger)}$% (Exp.Error=10%) and F40/F49= 0.89 ± 2.8% (Exp.Error=5.9%), $^{\dagger)}$:MVP statistical error
(4)	Conversion Factor	—(J3.2) $\lvert C/E-1\rvert \leq$Exp. error ($1.4 \sim 3.0$%)
(5)	==== Power Distribution ====	
(5.1)	Radial	—(J3.2) $\lvert C/E-1\rvert \leq$Exp. error $\simeq 1.5$%
(5.2)	Axial	—(J3.2) $\lvert C/E-1\rvert \leq$Exp. error $\simeq 1.5$%
(6)	Isothermal Temperature Coefficient	—(J3.2) $\lvert C/E-1\rvert \leq 2\times$ Exp. error (gaussed from figure)
(7)	==== Worth ===========	
(7.1)	Absorvers	—(J3.2) $\lvert C/E-1\rvert \leq$Exp. error $\simeq 6.8 \sim 13$%, except UO_2-Gd_2O_3 C/E= 1.13 ± 8.2% in MISTRAL Core 3
(7.2)	PWR Cluster Control Rod	—(J3.2) $\lvert C/E-1\rvert \leq$Exp. error $\simeq 6$%
(7.3)	BWR Cruciform Control Blade	—(J3.2) $\lvert C/E-1\rvert \leq$Exp. error $\simeq 6$%
(7.4)	Void	—(J3.2) $\lvert C/E-1\rvert \leq 2\times$ Exp. error $\simeq 12$%
(7.5)	Water Hole	—(J3.2) $\lvert C/E-1\rvert \leq 2\times$ Exp. error $\simeq 14$%
(8)	β_{eff}	—(J3.2) $\lvert C/E-1\rvert \leq 2\times$ Exp. error $\simeq 3.2$%
(9)	Analyses of PIE data$^{\sharp)}$	Isotopic Weight $(C/E) \pm 1\sigma(standard\,deviation)$ ^{234}U=1.05 ± 0.04, ^{235}U=1.04 ± 0.03, ^{236}U=0.94 ± 0.01, ^{238}U=1.00 ± 0.00, ^{238}Pu=0.94 ± 0.09, ^{239}Pu=0.99 ± 0.04, ^{240}Pu=0.99 ± 0.02, ^{241}Pu=0.96 ± 0.04, ^{242}Pu=0.92 ± 0.03, ^{241}Am=1.04 ± 0.24, well agreement, but larger calculational error for ^{241}Am because of missing power history effect to burn up process.

$^{\sharp)}$:Presentations by M. Sasagawa, T. Yamamoto, M. Sugawara and M. Yamamoto as D41 of the 2003 Fall Meeting of the Atomic Energy Society of Japan.

Remarks

[1]: Further analysis by using JENDL-3.3 library: JENDL-3.3 reactor constants not yet fully used.

[2]: Except k_{eff} and spectrum index, JENDL-3.2 and/or -3.3 fairy well predict reactor parameters, as shown in the summary Table 4. The overestimation of K_{eff} with the time as shown in Fig. 3 should be further studied. The k_{eff} deviation from JENDL-3.2 to -3.3 mainly comes from the ^{235}U's less neutron production rate by amount of -0.5%, ^{238}U more by +0.25, and ^{239}Pu more +0.12%, while ^{240}Pu less absorption rate by -0.25% and ^{241}Am more by +0.25% which are in cancellation each other.

[3]: The k_{eff}-value is jumping up and down depending on the revision of JENDL and its monotonous improvement cannot be expected. That may imply a limitation of k_{eff} prediction accuracy of about 0.5%. The k_{eff} is most important and basic quantity, and it is a result of the best balance of reaction rates among overall contributions from many constituent isotopes. Consequently, it is very sensitive to small perturbation destroying the balance. Therefore, in order to keep the best balance, equally and highly graded nuclear data of all isotopes are unavoidable. In the same sense, resonance parameters of fuel isotopes are key quantities to be continuously and carefully evaluated. Therefore, sensitivity approach should be adopted even for nuclear data evaluation as well as the cross section adjustment. If current C/E-value is a limitation of approaching from nuclear data side, the cross section adjustment method will be effective in order to guarantee practical accuracy.

[4]: Cell and/or core calculation code: As shown in Section II.1, general purpose neutronics calculational code SRAC and Monte Carlo code MVP are widely used by many users like standard codes. No problem has been found so far. Therefore, these codes and their cross section libraries can be nominated as the standard cross section libraries in Japan, since a study of standardization has been made by the Standard Cross Section Working Group of JNDC.

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