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Analyses of Criticality and Reactivity for TRACY Experiments Based on JENDL-3.3 Data Library

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The parameters on criticality and reactivity employed for computational simulations of the TRACY supercritical experiments were analyzed using a recently revised nuclear data library, JENDL-3.3. The parameters based on the JENDL-3.3 library were compared to those based on two former-used libraries, JENDL-3.2 and ENDF/B-VI. In the analyses computational codes, MVP, MCNP version 4C and TWOTRAN, were used. The following conclusions were obtained from the analyses: (1) The computational biases of the effective neutron multiplication factor attributable to the nuclear data libraries and to the computational codes depend not on the TRACY experimental conditions such as fuel conditions. (2) The fractional discrepancies in the kinetic parameter and coefficients of reactivity are within ~5 % between the three libraries. By comparison between calculations and measurements of the parameters, the JENDL-3.3 library is expected to give closer values to the measurements than the JENDL-3.2 and ENDF/B-VI libraries. (3) While the reactivity worth of transient rods expressed in the $ unit shows ~5 % discrepancy between the three libraries according to their respective β_{eff} values, there is little discrepancy in that expressed in the $\Delta k/k$ unit.

KEYWORDS: ***criticality, reactivity, supercritical experiment, TRACY, JENDL-3.3, JENDL-3.2, ENDF/B-VI, MVP, MCNP4C, TWOTRAN***

1. Introduction

In order to contribute to the safety assessment of nuclear fuel reprocessing plants, the studies on criticality accident phenomena of fissile solution have been conducted using the Transient Experiment Critical Facility[1] (TRACY) in the Japan Atomic Energy Research Institute. In the studies the accurate prediction of neutronic characteristics during power bursts is required because these characteristics control all transient behaviors during criticality accidents such as precipitous risings of radiation, temperature and pressure.

The following parameters are necessary to predict the neutronic characteristics of supercritical experiments of TRACY: criticality, excess reactivity, solution level coefficient of reactivity, temperature coefficient of reactivity, void coefficient of reactivity and kinetic parameter. These parameters depend not only on fuel conditions but also on nuclear data used in computational simulations. Such parameters were previously evaluated and employed for computational simulations of the TRACY experiments,[2,3] however, they were obtained under only a few cases of fuel conditions, and all the obtained parameters have not been in comparison among nuclear data libraries. It is thus encouraged to evaluate the dependencies of the parameters on the TRACY experimental conditions and on nuclear data.

In the present work, the above-mentioned parameters have been analyzed and re-evaluated using a recently revised nuclear data library, JENDL-3.3.[4] The comparisons of the parameters between the JENDL-3.3 library and two former-used libraries, JENDL-3.2[5] and ENDF/B-VI,[6] are also discussed.

2. Experiments

TRACY is a pulsed reactor using 10-wt%-enriched uranyl nitrate solution as fuel. The core tank of TRACY is annular shape and its dimensions are about 7.6-cm-inner-diameter (I.D.), 50-cm-outer-diameter (O.D.) and 2-m-height, respectively. TRACY has two exchangeable transient rods, 1.8-$- and 3-$-worth, which can insert the excess reactivity up to 3 $. The TRACY supercritical experiment is initiated by the withdrawal of one of the transient rods from the TRACY core or by continuous feed of the fuel solution beyond a critical solution level.

The experimental conditions and results[7-10] selected for the present analyses are listed in **Table 1.** The fuel conditions in this table cover those in almost all the former TRACY experiments.

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Table 1 Experimental conditions and results.

Run	Fuel conditions				Results		
	Uranium concentration b) (gU/l)	Free nitric acid molarity b) (mol/l)	Solution density b) (g/cm³)	Solution temperature c) (°C)	Critical solution level (cm)		Transient rod worth ($)
					Transient rod withdrawn	Transient rod inserted	
R010	433.4	0.85	1.60300	23.9	45.188	49.722	1.82
R025	426.1	0.85	1.59393	25.5	45.990	50.819	1.85
R035	405.5	0.78	1.56633	25.1	48.361	54.063	1.90
R036	406.9	0.77	1.56885	25.7	48.206	53.863	1.90
R062 a)	421.9	0.77	1.58946	25.7	46.073	54.063	2.82
R064 a)	430.0	0.76	1.59746	25.5	45.306	52.847	2.80
R066 a)	404.7	0.75	1.56493	25.6	48.491	57.963	2.88
R068	394.2	0.77	1.55118	25.8	50.120	60.725	2.92
R076 a)	396.2	0.74	1.55305	25.3	49.626	59.897	2.91
R090 a)	390.3	0.71	1.54512	25.7	50.553	61.521	2.93
R105 a)	399.0	0.71	1.55570	25.2	49.084	59.035	2.91
R107	388.8	0.70	1.54249	25.6	50.887	62.220	2.96
R116	392.9	0.70	1.54782	25.8	49.987	60.560	2.93
R119	373.7	0.66	1.52183	26.2	53.777	67.419	3.00
R130 a)	372.7	0.66	1.52075	26.1	53.813	67.460	3.00
R143	375.9	0.64	1.52371	24.7	52.760	65.504	2.98

a) The cases selected for the calculations of the kinetic parameter and coefficients of reactivity.
b) Measured at 25 °C.
c) Mean temperature measured at the criticality states with a transient rod withdrawn and inserted.

3. Calculations

The computational analyses were performed using a continuous energy Monte Carlo code, MVP,[11] and a two-dimensional S_N transport code, TWOTRAN, in the SRAC system.[12] The former code was used for the evaluation of criticality and reactivity worth of transient rods. The latter was used to obtain coefficients of reactivity and kinetic parameter, which were evaluated from very small difference of k_{eff}'s and a set of forward and adjoint fluxes, respectively. In addition to the codes, a worldwide used continuous energy Monte Carlo code, MCNP version 4C (MCNP 4C),[13] was also used for the comparison of criticality to the MVP code.

A calculation model of TRACY is illustrated in **Fig. 1**. For the Monte Carlo calculations all components inside the TRACY tank were modeled by the combination geometry as precisely as possible, while a simplified model (R-Z geometry) was adopted for the S_N transport calculations: all the inside components and the parts beyond the solution level were left out of consideration. Each Monte Carlo calculation employed 300 generations of 50,000 neutrons, and the first 100 generations were excluded from the statistical process of tallies. For the S_N transport calculations, macroscopic cross section data were collapsed into a 17-group energy structure under the scattering conditions P1 and S8 by using a one-dimensional S_N transport code, ANISN,[12] in the SRAC system. The atomic number densities of fuel solution were obtained through a density equation.[14]

4. Results and Discussions

4.1 Criticality

The effective neutron multiplication factor (k_{eff}) was calculated using the MVP code and then the computational bias of k_{eff} attributable to the nuclear data libraries was evaluated. The k_{eff} bias also attributable to computational codes was compared among the MVP, MCNP4C and TWOTRAN codes. The comparison of the k_{eff} bias among the JENDL-3.3, JENDL-3.2 and ENDF/B-VI libraries is shown in **Fig. 2** and the comparison among the three codes is shown in **Fig. 3**.

The evaluated k_{eff} bias attributable to the three libraries supported the analytical results of the TRACY experiment reported in the original paper regarding release of the JENDL-3.3 library.[4] The overestimation of k_{eff} found in the JENDL-3.2 library was reduced by the modification of the neutron absorption cross section of ^{14}N, the fission spectrum and thermal fission cross section of ^{235}U in the JENDL-3.3 library.[4]

In regard to the k_{eff} bias attributable to the computational codes, there was no difference in its results between the MVP and MCNP4C codes. Relatively large differences in the results between the TWOTRAN code and the two Monte Carlo codes were due to the difference in calculation methods rather than in geometry models.

An additional fact was obtained that the k_{eff} bias depended neither on the transient rod position nor on the fuel conditions in the TRACY experiments, as seen in **Figs. 2 and 3**.

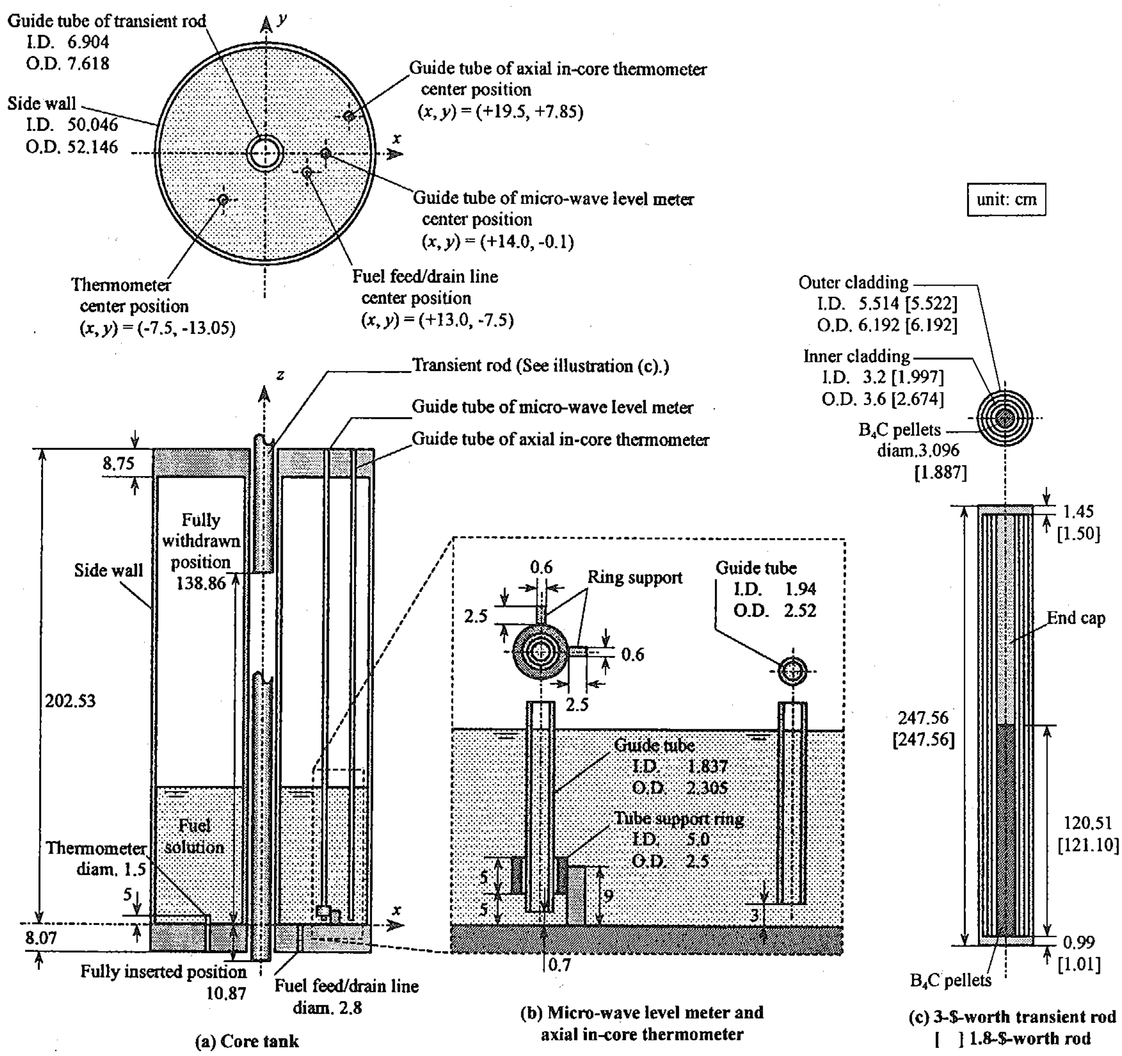

Fig. 1 Calculation model of TRACY.

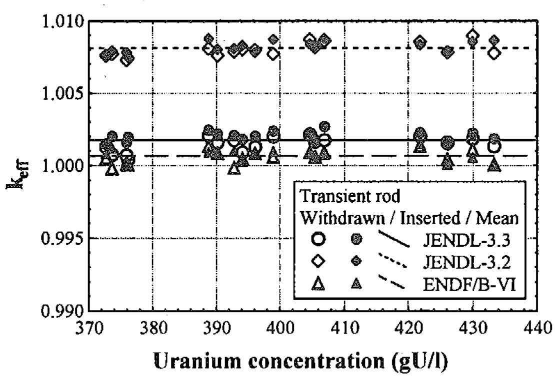

Fig. 2 Comparison of the computational bias of k_{eff} among nuclear data libraries. The k_{eff}'s were calculated using the MVP code. The fractional standard deviation (1σ) of each calculation was ± ~0.02 %. The k_{eff} biases attributable to the JENDL-3.3, JENDL-3.2 and ENDF/B-VI libraries were evaluated to be 1.0018±0.0005, 1.0081±0.0004 and 1.0007±0.0004, respectively.

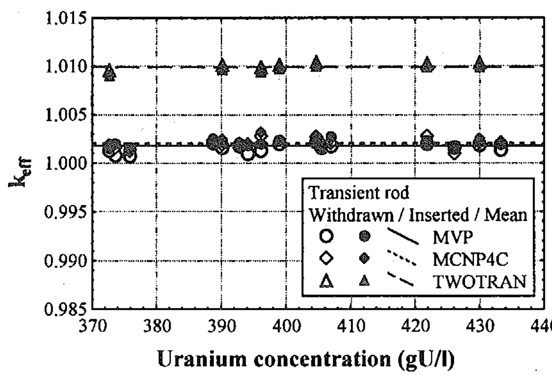

Fig. 3 Comparison of the computational bias of k_{eff} among computational codes. The k_{eff}'s were based on the JENDL-3.3 library. The fractional standard deviation (1σ) of each Monte Carlo calculation was ± ~0.02 %. The k_{eff} biases attributable to the MVP, MCNP4C and TWOTRAN codes were evaluated to be 1.0018±0.0005, 1.0020±0.0005 and 1.0099±0.0004, respectively.

4.2 Kinetic Parameter

The effective delayed neutron fraction (β_{eff}), the prompt neutron generation time (Λ) and the kinetic parameter (β_{eff}/Λ) were calculated using the TWOTRAN code. **Figures 4, 5 and 6** show the comparison of these three parameters, respectively, between the JENDL-3.3 library and the two former-used libraries.

Systematic discrepancy in β_{eff} between the three libraries was observed in **Fig. 4**. The β_{eff} based on the JENDL-3.3 library was ~1 % and ~5 % below that based on the JENDL-3.2 and ENDF/B-VI libraries, respectively. This discrepancy was attributed to the difference in the macroscopic neutron production cross section multiplied by β_{eff} in the thermal neutron range between the libraries rather than delayed neutron constants and fission spectra which also depend on the nuclear data libraries.

In **Fig. 5**, little discrepancy in Λ was seen between the three libraries. This agreement resulted from the fact that there were negligibly small differences in the macroscopic neutron production cross section, the forward and adjoint fluxes calculated using the three libraries.

The β_{eff}/Λ showed systematic discrepancy in its results between the three libraries according to their respective β_{eff} values. The JENDL-3.3 library was found to give the β_{eff}/Λ values closer to the measurement than the JENDL-3.2 and ENDF/B-VI libraries, as seen in **Fig. 6**.

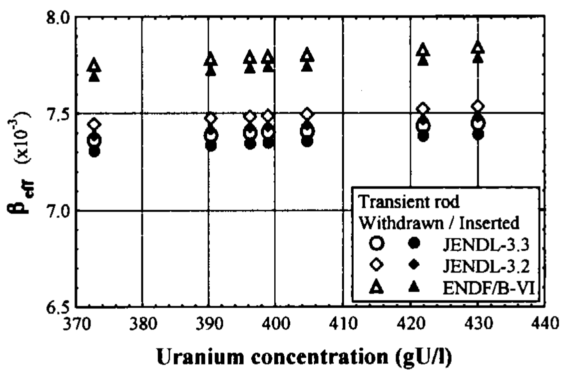

Fig. 4 Effective delayed neutron fraction.

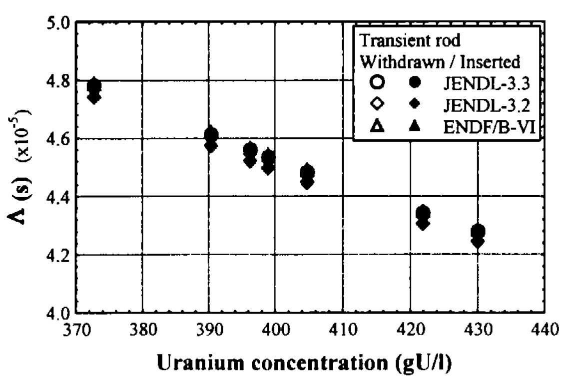

Fig. 5 Prompt neutron generation time.

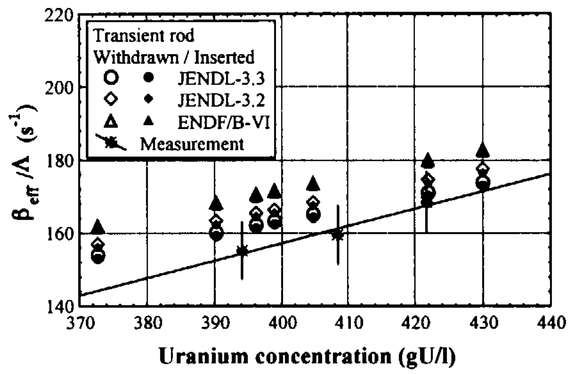

Fig. 6 Kinetic parameter, the ratio of effective delayed neutron fraction to prompt neutron generation time. The measurements were obtained by the pulsed neutron method.[10)]

4.3 Coefficients of Reactivity

All the following coefficients of reactivity were calculated using the TWOTRAN code. The solution level coefficient of reactivity ($d\rho/dH$) was calculated with the level difference of 1 cm beyond each critical solution level. The temperature coefficient of reactivity (α_T) was calculated by uniformly varying the solution temperature from 25 to 40 °C. The void coefficient of reactivity (α_V) was also calculated by uniformly varying the void fraction from 0 to 5 vol%. **Figure 7** shows the solution level coefficient of reactivity expressed as a function of the critical solution level. The comparison of temperature and void coefficients of reactivity are also shown in **Figs. 8 and 9**, respectively.

It was found from **Figs. 7, 8 and 9** that there were systematic discrepancies in all the coefficients between the three libraries. These discrepancies are summarized in **Table 2**. By comparison between the JENDL-3.3 and JENDL-3.2 libraries, the fractional discrepancy in the temperature coefficient was ~2 % above that of the solution level and void coefficients which had no connection with temperature variation. This difference was attributable to the modification of the resonance parameters of ^{235}U in the JENDL-3.3 library.[4)] By comparison between the JENDL-3.3 and ENDF/B-VI libraries, while there were ~5 % discrepancies in the three coefficients expressed in the \$ unit according to the β_{eff} values, no discrepancies in those expressed in the $\Delta k/k$ unit were observed. By comparison to the measurements shown in **Fig. 7**, the solution level coefficient of reactivity based on the JENDL-3.3 library was found to be in more satisfactory agreement with the measurements than those based on the JENDL-3.2 and ENDF/B-VI libraries.

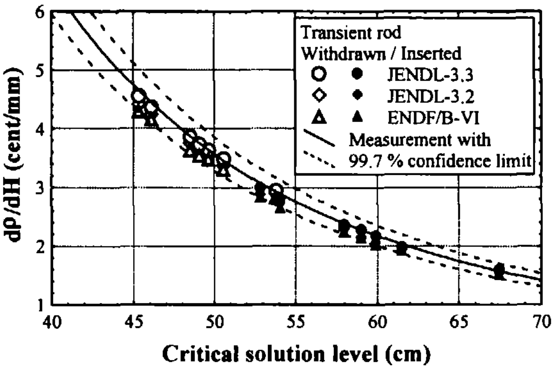

Fig. 7 Solution level coefficient of reactivity. The measurement indicates the curve fitted to the dρ/dH's measured in the cases shown in Table 1.

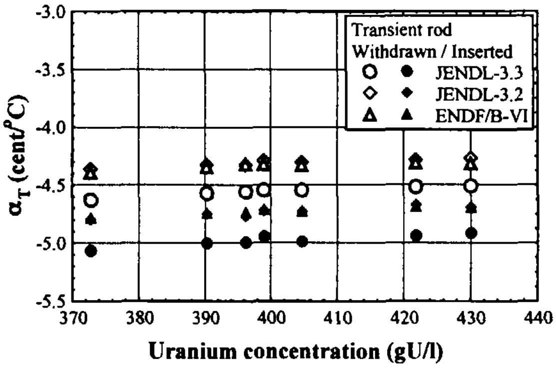

Fig. 8 Temperature coefficient of reactivity. This coefficient was evaluated assuming uniform variation of solution temperature.

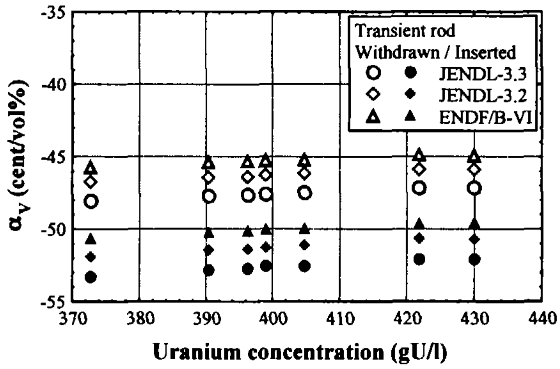

Fig. 9 Void coefficient of reactivity. This coefficient was evaluated assuming uniform variation of void fraction.

Table 2 Summary of comparison of coefficients of reactivity between nuclear data libraries.

Library	JENDL-3.2 / JENDL-3.3	ENDF/B-VI / JENDL-3.3
dρ/dH	0.97 (0.98)[a]	0.95 (1.00)
α_T	0.95 (0.96)	0.95 (1.00)
α_V	0.97 (0.98)	0.95 (1.00)

a) Comparison in the \$ unit, ( ) comparison in the Δk/k unit.

4.4 Reactivity Worth of Transient Rod

The reactivity worth of each transient rod was calculated using the MVP code. **Figure 10** shows the full-worth of the rod obtained as the excess reactivity by the full-stroke withdrawal of the rod. **Figure 11** shows an S-shaped curve of rod worth obtained as the subcriticality by the partial rod insertion.

While the reactivity worth expressed in the \$ unit showed ~5 % discrepancy at maximum between the three libraries according to their respective β_{eff} values, there was little discrepancy in that expressed in the Δk/k unit. It was found from **Figs. 10** and **11** that every calculation based on the three libraries reproduced the measurements of the full-worth and the S-shape curve of rod worth within 99.7 % confidence limits.

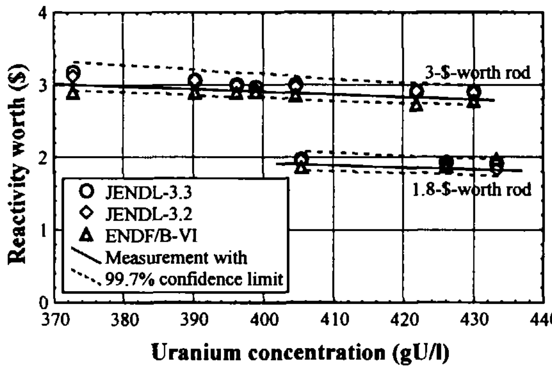

Fig. 10 Full-worth of 1.8-\$- and 3-\$-worth transient rods. The fractional standard deviation (1σ) of each calculation was ± ~2 %.

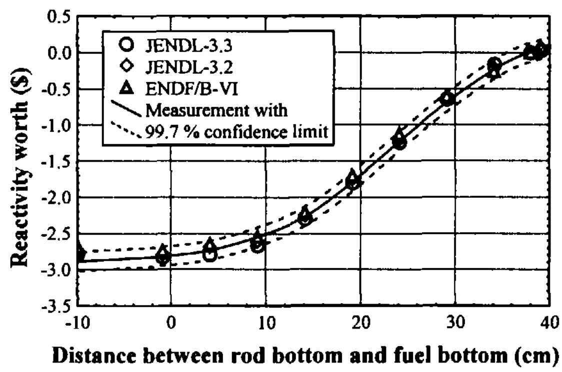

Fig. 11 S-shaped curve of reactivity worth of 3-\$-worth transient rod. The fractional standard deviation (1σ) of each calculation was ± ~2 %. The measurements were obtained by the neutron source multiplication method and the pulsed neutron method.[10)]

5. Conclusions

The parameters on criticality and reactivity employed for computational simulations of the TRACY supercritical experiments were analyzed using a recently revised nuclear data library, JENDL-3.3. The parameters based on the JENDL-3.3 library were compared to those based on two former-used libraries, JENDL-3.2 and ENDF/B-VI. The following conclusions were obtained from the analyses:

(1) Criticality: The computational biases of k_{eff} attributable to the nuclear data libraries and to the computational codes were evaluated, and it was found that the k_{eff} bias depended neither on the transient rod position nor the fuel conditions in the TRACY experiments.

(2) Kinetic parameter and coefficients of reactivity: The fractional discrepancies in the parameter and coefficients were within ~5 % between the three libraries. By comparison between calculations and measurements, the JENDL-3.3 library is expected to give closer values to the measurements than the JENDL-3.2 and ENDF/B-VI libraries.

(3) Reactivity worth of transient rods: While the reactivity worth expressed in the $ unit showed ~5 % discrepancy at maximum between the three libraries according to their respective β_{eff} values, there was little discrepancy in that expressed in the $\Delta k/k$ unit. Every calculation based on the three libraries satisfactorily reproduced the measurements of the full-worth and the S-shape curve of rod worth.

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