Creation of Benchmark Data on JOYO and DCA Reactor Physics Experiments

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Abstract

Benchmark data have been created on reactor physics experiments performed in two reactors: the experimental fast reactor JOYO MK-I and Deuterium Critical Assembly (DCA). The data were prepared for the International Reactor Physics Experiment Evaluation Project (IRPhEP).

In JOYO data, five kinds of reactivity data were evaluated: (a) criticality, (b) control rod worth, (c) sodium void reactivity, (d) fuel replacement reactivity, and (e) isothermal temperature coefficient. In particular, the control rod worth, a key quantity in all the reactivity evaluations, was carefully evaluated, considering interaction effects.

In DCA data, three kinds of parameters measured with 1.2%UO_2 fuel were evaluated: (a) critical moderator level, (b) epithermal capture ratio of ^{238}U, (c) dysprosium thermal reaction rate distribution in a fuel assembly. Data are systematically arranged in eight kinds of core configurations, varying the assembly pitch and void fraction.

Each of evaluated data has a unique feature and will be useful to validate reactor physics calculation schemes.

KEYWORDS: IRPhE, JOYO MK-I, DCA, criticality, control rod worth, Fast reactor, Heavy water moderator, Thermal reactor

1. Introduction

Benchmark data have been created on reactor physics experiments performed in two reactors. One is on the experimental fast reactor "JOYO" and the other is on Deuterium Critical Assembly (DCA). Both reactors are located in Oarai Research and Development Center of Japan Atomic Energy Agency (JAEA). The data were prepared for the International Reactor Physics Experiment Evaluation Project (IRPhEP). [1-2]

"JOYO" is a sodium-cooled fast reactor with plutonium-uranium mixed oxide (MOX) fuel. The initial criticality was achieved in a breeder core of JOYO called "MK-I" in 1977. JOYO has experienced large-scale changes in its core configuration twice and the current core called "MK-III" is in operation mainly for irradiation purposes. The following parameters were evaluated for benchmark data: (a) criticality, (b) control rod worth, (c) sodium void reactivity, (d) fuel replacement reactivity, and (e) isothermal temperature coefficient, all of which were measured in a startup test.

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DCA is a tank type critical assembly with a maximum power of 1kWth. It was constructed in 1969 to investigate reactor physics of the advanced thermal reactor "FUGEN", a heavy-water-moderated, boiling-light-water cooled, pressure-tube-type power reactor developed in Japan. A variety of experiments has been carried out, using low-enriched uranium oxide or MOX fuel. The reactor stopped operation permanently in 2001 and is now under decommissioning. The nuclear parameters evaluated are: (a) critical moderator level, (b) epithermal capture ratio of ^{238}U (ρ^{28}), (c) dysprosium thermal reaction rate distribution in a fuel assembly; measured using 1.2wt% enriched uranium oxide fuel.

2. JOYO MK-I Data

2.1 Core Configuration

There are two experimental cores for the criticality evaluation in the JOYO MK-I startup test. The first core, having 64 core fuel subassemblies, is a clean core with all control rods fully withdrawn. The second core has 70 core fuel subassemblies with approximately a half insertion of two regulation rods into the core fuel region. Various measurements were performed in the cores or slightly rearranged cores.

A horizontal cross-sectional view of a core and general specifications are shown in Fig.1 and Table 1, respectively. Six control rods are installed in the third row (numbering from the core center). The control rods consist of two regulation rods (RR1 and RR2) and four safety rods (SRs).

The criticality was achieved by the control rod operation. Under a stable temperature condition and a constant sodium flow rate, the control rod positions were carefully adjusted to realize a stable condition of neutron flux monitors.

Figure 1: JOYO MK-I core map (70 fuel subassemblies)

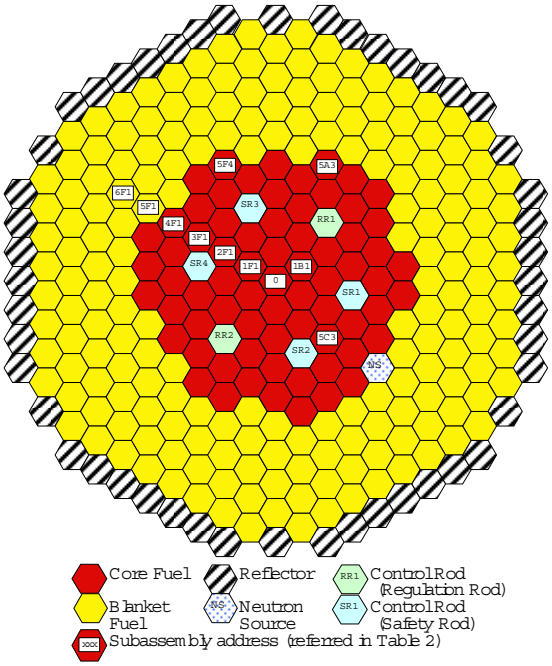


Table 1: General specifications of JOYO MK-I

Item	Specification
Fuel type	U-Pu Mixed Oxide
Coolant	Liquid Sodium
Reactor thermal power (MW)	50
Number of primary loop	2
Primary flow rate (t/hr)	Approx. 2200
Core inlet temperature (□)	250 (low power) 370 (high power)
Core fuel height (mm)	600±2.0
Axial blanket thickness (mm)	400±2.0(Upper, Lower)
Radial blanket thickness (mm)	Approx. 300
Equivalent core diameter (mm)	Approx. 720
Plutonium weight (kg)	Approx. 130
Uranium weight (kg of ^{235}U)	Approx. 140
Fissile Pu ratio	80.4±1.0
U enrichment (w/o)	23.0±0.3
^{10}B enrichment (w/o)	>91.0
Sub-assemblies pitch (mm)	81.5±0.2
Number of blanket subassembly	179
Uranium weight in blanket (t)	Approx. 7

2.2 Control Rod Worth Calibration

Results of the control rod worth evaluation are used in the other reactivity evaluations. Thus, the data were carefully evaluated considering the control rod interaction effect by 3-D transport calculation.

2.2.1 Measured Data

Data on all the six control rods were measured by either the period method or the replacement method. Firstly, the RR1 worth was measured by the period method over its full stroke (from 0mm to 700mm). In the very first step, a critical condition was achieved by holding RR1 at its lowest position (0mm), RR2 at 620.4mm and all four SRs fully at the upper position. Then, RR1 was withdrawn step by step, in which the period was measured and RR2 was inserted to recover the critical condition. The total number of measurement steps was 33.

Next, the RR2 worth was measured by the replacement method using the RR1 worth for the compensation. The replacement method was also applied in the calibration of all four safety rods. Compensation of the SR1 worth and the SR2 worth was made by mainly RR2 and RR1, respectively. Compensation of the SR3 worth and the SR4 worth was made by mainly RR2 and RR1, respectively.

2.2.2 Calibration Procedure

The interaction effect f_L between a target control rod X at a position x and other control rods at positions y_i $(i = 1,\ldots 5)$ is defined as

$$f_L(x, y_1, y_2, \cdots, y_5) \equiv \frac{\Delta\rho(x, y_1, y_2, \cdots, y_5)}{\Delta\rho(x, L, L, \cdots, L)}, \tag{1}$$

where, $\Delta\rho$ is differential control rod worth, and L is the upper end position (700mm for RRs, 900mm for SRs).

Using f_L, the non-interacted worth is obtained from the measured. The non-interacted worth of RR1 at the position $rr1$ was obtained from a measured value of differential control rod worth under a RR2 insertion at the position $rr2$, $\Delta\rho_{\exp}(rr1, rr2, L, L, L, L)$, as

$$\Delta\rho(rr1, L, L, L, L, L) = \frac{\Delta\rho_{\exp}(rr1, rr2, L, L, L, L)}{f_L(rr1, rr2, L, L, L, L)}, \tag{2}$$

The interaction effect was calculated in all the measurement steps.

The non-interacted worth of the other rods, measured by the replacement method, was evaluated from worth of the compensating rod and interaction effects. For the evaluation, the non-interacted worth of the compensating rod was fitted to the 6th order polynomials and the calculated interaction effects to the 4th order polynomials. Figs.2 and 3 show the non-interacted worth of RR1 and interaction effects from RR2 to RR1, respectively.

This procedure was used to evaluate excess reactivities in the other reactivity measurements.

Fig. 2 Non-interacted cumulative rod worth (RR1)

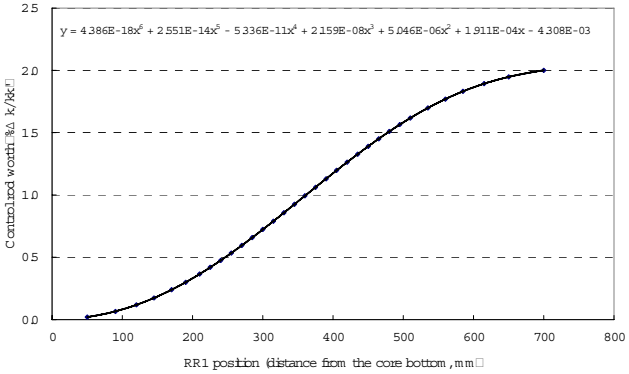
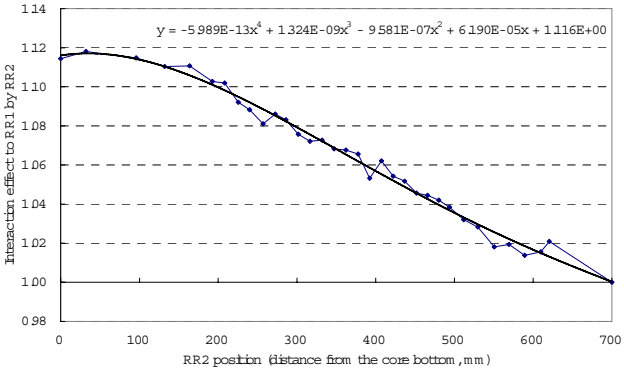


Fig. 3 Interaction effect (to RR1 by RR2)



2.3 Description of measured parameters

Measurement and evaluation procedures on the other parameters are described briefly in this section. Temperature was all adjusted to a standard value of 250□ based on the evaluated value of the isothermal temperature coefficient. Evaluated results are summarized in Table 2, together with uncertainty and calculation results.

(1) Criticality

The criticality with 64 core fuel subassemblies was achieved at a temperature of 204.7□. All the four SRs and RR1 were fully withdrawn; hence, the excess reactivity was evaluated from the RR2 position by using its rod worth curve.

Another criticality measurement was performed with 70 core fuel subassemblies at a temperature of 200.0□. In this case, both RR1 and RR2 were inserted in the core (close to the mid-plane of the core fuel region). The benchmark model treats the insertions as they are.

(2) Sodium void reactivity

A special fuel subassembly was prepared for the measurement. The subassembly has a valve mechanism that can control sodium flowing into the wrapper tube. A standard fuel assembly was replaced by the special subassembly and the sodium void reactivity was evaluated from a reactivity change by the voiding. The reactivity was measured at 8 addresses, of which 6 data were judged acceptable as benchmark data. The influence of the fuel composition difference between the special subassembly and the averaged value was corrected in the benchmark data.

(3) Fuel replacement reactivity

The fuel replacement reactivity was measured at 6 addresses in total. The core fuel subassembly was replaced by the radial blanket fuel subassembly at four addresses around the 5th row. At the center of the core and in the radial blanket region, the subassembly was just removed, that is, replaced to sodium. The influence of the fuel composition difference between the replaced subassembly and the averaged value was corrected in the benchmark data.

(4) Isothermal temperature coefficient

The isothermal temperature coefficient was measured several times at low power conditions by

changing the core temperature ranging from 170 to 250□. During the measurements, the primary coolant flow rate was maintained at approximately 20% of the rated value. Good linearity between the excess reactivity and the coolant temperature was observed. The gradient of the approximate line corresponds to the measured isothermal temperature coefficient.

2.4 Uncertainty Evaluation

Uncertainties associated with measurement technique, geometry, and material compositions were evaluated in all the parameters. Extensive efforts were devoted to collect evidence data for all the possible uncertainty sources from operation records, measured data, design specifications, etc. The uncertainties were classified into systematic and random components.

Major sources in the uncertainty evaluation are explained below:

(1) Criticality: The isotopic compositions of ^{235}U and ^{239}Pu are the dominant sources.

In the random uncertainty associated with the compositions, the dominant source is the metal existence ratio in the powder (uncertainties in the ratio of 0.05% for UO_2 and 0.09% for PuO_2, based on measured data). The systematic uncertainty is also dominantly determined by the ratio, where a target uncertainty (0.1%) recommended by IAEA [3] was employed.

The uncertainty due to measurement technique is dominant on the following reactivities.

(2) Control rod worth: In the random component (0.5~0.6% or 2¢), a fitting error in the worth curve is the largest source with an uncertainty of 2¢. It was estimated by averaging absolute differences in a rod total worth between those measured (after interaction correction) and those obtained from the fitted worth curve.

In their systematic component (1~3%), an uncertainty of 2.1% in a base calibration data of RR1 worth, is a major source. In RR1 worth, a systematic uncertainty of 1.5% is the major component, where a portion (20%) of the interaction effect was assumed an uncertainty.

(3) Sodium void reactivity: In both components, uncertainty in the core temperature 1.3□ is a major source, with resulting uncertainties in reactivity of 0.5 and 1.0¢, for random and systematic components, respectively. Another major source is repeatability of critical measurement including thermal expansion effect of the control rod drive mechanism, causing uncertainties of 0.3-0.8¢. Although the values are small, they are significant compared to the measured reactivities.

(4) Fuel replacement reactivity: In the random component (1-8%), an uncertainty in core temperature is a major source, with an uncertainty in reactivity of 1¢. Another major source is repeatability of critical measurement (0.9¢). In the systematic component (2-9%), uncertainties in the control rod worth calibration of 1-5¢ are dominant.

(5) Isothermal temperature coefficient: The random component (1.6%) comes from error-weighted least square fitting, where uncertainties in temperature (0.6%) and in reactivity data (0.6%) are considered. The systematic component (1.2%) comes from a variation of measurement results performed under a similar core configuration.

2.5 Sample Calculation

Table 2 includes sample calculation results based on two different nuclear data, JENDL-3.2 and JENDL-3.3. In the calculation, firstly, diffusion calculation in a benchmark Tri-Z model was conducted based on the effective cross section set prepared by heterogeneous cell calculation, and then corrections were applied on various effects including the transport, the mesh, and a detailed energy structure.

A simplified benchmark model is also available in the benchmark data, which uses homogeneous cell calculation and Hex-Z diffusion core model.

Table 2: Evaluation results on JOYO MK-I experiments

Parameter	Sub parameter	Benchmark exp. data	Exp. uncertainty (%) Random	Exp. uncertainty (%) Systematic	(C-E)/E (%) JENDL-3.2	(C-E)/E (%) JENDL-3.3
Criticality	64 S/A	1.0011	0.08	0.10	-0.55	-0.57
	70 S/A	0.9981	0.08	0.10	-0.59	-0.60
Control rod worth	RR1	387.5 (¢)	0.6	1.5	-1.1	-1.7
	RR2	385.8	0.6	2.5	-2.1	-2.7
	SR1	407.0	0.6	3.4	-3.3	-3.9
	SR2	410.7	0.6	2.4	-2.5	-3.1
	SR3	396.4	0.5	3.4	-2.5	-3.1
	SR4	400.3	0.6	2.5	-2.0	-2.7
Sodium void reactivity	0[(a)]	-6.32 (¢)	18.5	9.9	17.3	12.7
	1F1	-7.06	15.5	9.1	7.4	3.2
	2F1	-7.41	16.5	9.2	7.3	3.9
	3F1	-5.93	18.6	11.3	18.8	16.2
	4F1	-5.53	21.7	11.8	25.5	24.1
	6F1	-0.58	159.4	80.4	-221	-219
Fuel replacement reactivity[(b)]	(F->B) 4F1	-92.85 (¢)	1.6	3.0	-6.4	-6.7
	5A3	-64.76	2.1	3.2	-1.5	-1.9
	5C3	-83.14	1.7	3.0	-6.1	-6.4
	5F4	-76.88	1.9	3.2	-6.0	-6.4
	(F->Na) 0	-262.37	0.8	1.8	2.3	2.3
	(B->Na) 5F1	-16.46	8.4	9.2	-15.6	-16.3
Isothermal temperature coefficient	65 S/A	0.769 (¢/°C)	1.6	1.2	2.0	1.0

(a) Subassembly address (see Fig.1)

(b) F: fuel, B: radial blanket, Na: Na channel

3. DCA Data

3.1 Core Configuration

A horizontal cross-sectional view of a core is shown in Fig.4 and general specifications are summarized in Table 3. Absorber rods were equipped but not shown in the figure and the table. They were not inserted in the moderator region in the evaluated experiments. One of prepared benchmark models describes the presence of absorber rods above the moderator.

A specific feature of DCA core is use of cluster shape fuel assembly placed in a double-walled aluminum tubes. The unit cell was arranged in a square lattice pitch. The criticality was achieved by adjusting heavy water moderator level in the core tank. The inner aluminum tube is filled with either light water or air to simulate a coolant void fraction. Intermediate void fractions were simulated by using a mixture of D_2O, H_2O, and H_3BO_3.

Figure 4: DCA core arrangement (22.5cm lattice pitch)

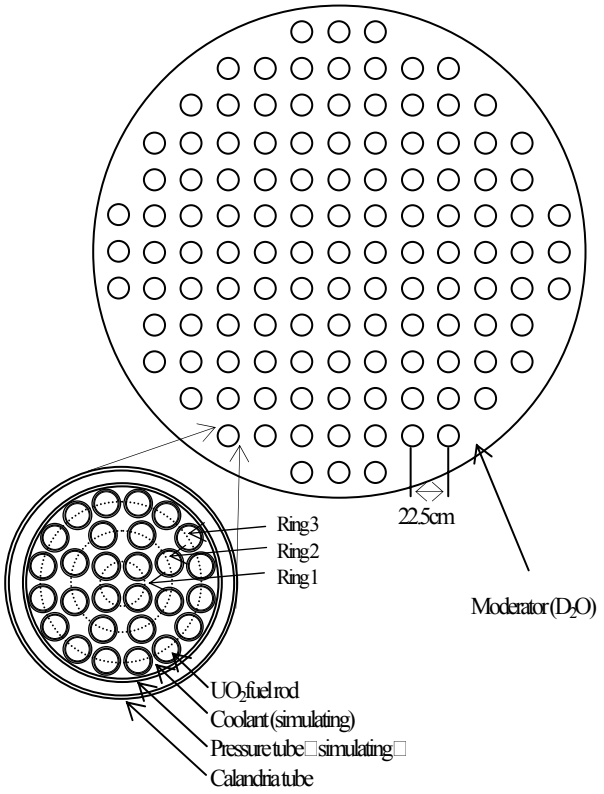


Table 3: General specifications of DCA

Item	Specification
Core type	D_2O-moderated tank type critical assembly
Power	1kwth (max)
Core tank (Al-alloy)	
Inner diameter	(300.5 ± 0.1) cm
Height	(310.00 ± 0.03) cm
Lattice (subassembly) pitch	(22.50 ± 0.03) cm or (25.00 ± 0.03) cm
Fuel cluster	
Enrichment, wt.%	1.203 ± 0.002
U element content, wt.%	87.85 ± 0.02
Density, g/cm^3	10.36 ± 0.05
Pellet diameter,	(1.480 ± 0.003) cm
Fuel rod stack length	(199.8 ± 0.1) cm
Cladding (Al-alloy) O.D	(1.672 ± 0.002) cm
Pressure tube (Al-alloy) O.D.	(12.10 ± 0.01) cm
Calandria tube (Al-alloy) O.D.	(13.65 ± 0.01) cm
Moderator(D_2O)	
Purity, mol%	99.47 ± 0.07
Density, g/cm^3	1.1044 ± 0.0008
Coolant (simulating) (H_2O)	Intermediate voiding conditions were simulated by a mixture of D_2O, H_2O, and H_3BO_3

3.2 Description of measured parameters

Measurement procedures on the three kinds of nuclear parameters are described briefly in this section. The parameters were systematically measured in the eight core configurations with lattice pitches of 22.5 and 25.0 cm, varying simulating coolant void fraction among 0, 30, 70, and 100 %. Evaluated results are summarized in Tables 4-6, together with uncertainty and calculation results.

(1) Criticality

The critical moderator level was measured at a power of 1W. To avoid an influence of (γ,n) reaction in deuterium, the critical level was measured before power ascension.

When the simulating coolant was supplied in the pressure tube, the coolant level was adjusted to the critical D_2O level as closely as possible by repeating experiments.

(2) ρ^{28}: The epithermal capture ratio of ^{238}U

In the measurement, ^{239}Np γ-ray activities of irradiated depleted uranium foils, with and without cadmium (Cd) cover, R_{DU}^{Bare} and R_{DU}^{Cd}, were measured and ρ^{28} was determined by

$$\rho^{28} = \frac{1}{R_{DU}^{Bare} / R_{DU}^{Cd} - 1}. \quad (3)$$

Irradiations of the foils with and without Cd cover were carried out one at a time. The foils were inserted between fuel pellets in a rod on each ring shown in Fig.4.

(3) Dysprosium (Dy) thermal-capture reaction-rate distribution in a fuel assembly

The reaction-rate distribution was measured by counting ^{165}Dy β-rays activities of irradiated Dy foils with or without Cd cover. The Dy foils were inserted between fuel pellets in a rod on each ring. The reaction-rate was normalized at the 1st ring.

The Dy foils with and without Cd cover were irradiated independently. The accumulated power in the irradiation was monitored with bare gold foils attached outside the core tank.

3.3 Uncertainty Evaluation

Uncertainties associated with measurement technique were considered in all the parameters. Those associated with geometry and material compositions were considered only in the criticality.

The evaluation was carried out on all the possible uncertainty sources and in all the core configurations.

Major sources in the uncertainty evaluation are explained below:

(1) Criticality: The systematic uncertainty due to geometry and material compositions is dominant.

Various sources induce the uncertainty in a similar magnitude of ~0.04%. Examples of sources are the density of aluminum-alloy (used in most of structural components), fuel density, fuel enrichment, and diameter of calandria tube. D_2O moderator density causes a largest uncertainty of 0.08% in the core with 22.5cm lattice pitch and 0% void fraction.

(2) ρ^{28}: The systematic uncertainty is the largest.

In the systematic uncertainty, the dominant source is on the foil setting with Cd cover. UO_2 pellets were placed in the Cd cover to prevent epithermal neutron streaming into the foil center. An experimental correction factor was applied to compensate the possible flux distortion by the insertion. However, one value was applied commonly to all the measured data, independent of foil locations and core configurations. Thus, 100% uncertainty was assumed on the correction.

(3) Dy reaction-rate distribution in a fuel assembly: The random uncertainty is the largest, which was based on reproducibility of the measurements.

3.4 Sample Calculation

Tables 4-6 also include sample calculation results on two different nuclear data, JENDL-3.2 and JEND-3.3 (criticality only). The calculations were carried out using a continuous energy Monte Carlo code.

Table 4: Evaluation results on DCA – criticality -

Pitch (cm)	Void fraction (%)	Benchmark data on critical level(cm)	Exp. Uncertainty[(a)] in criticality (%)	(C-E)/E (%)[(b)] JENDL-3.2	JENDL-3.3
22.5	0	97.00	0.12	0.02	0.17
	30	97.54	0.12	0.04	0.18
	70	103.01	0.12	0.31	0.54
	100	112.58	0.11	0.26	0.47
25.0	0	105.88	0.15	-0.02	0.18
	30	103.28	0.14	0.02	0.15
	70	103.45	0.12	0.05	0.23
	100	105.18	0.11	0.21	0.36

(a) Systematic component due to dimension and composition is dominant
(b) Statistical uncertainty 0.02%.

Table 5: Evaluation results on DCA - ρ^{28} -

Pitch (cm)	Void fraction (%)	Benchmark experimental value 1st ring	2nd ring	3rd ring	Exp. uncertainty	(C-E)/E (%)[(a)] 1st ring	2nd ring	3rd ring
22.5	0	1.080	0.928	0.777	Systematic	4.7	2.3	-5.2
	30	1.244	1.044	0.855	3.6%	3.7	3.2	-3.6
	70	1.561	1.316	1.077	Random	-5.7	-5.8	-7.7
	100	1.276	1.129	---[(b)]	3.0%	-6.9	-0.2	---
25.0	0	1.064	0.883	0.653	Systematic	-0.6	-1.1	-2.6
	30	1.252	1.042	0.766	3.6%	-5.1	-7.5	-8.2
	70	1.220	1.043	0.874	Random	3.8	-1.0	-6.7
	100	0.948	0.865	0.861	1.5%	-0.5	2.6	-0.6

(a) Relative statistical uncertainties of calculations are 0.5%, 0.3%, and 0.2%, for each ring.
(b) Data are rejected in the evaluation.

Table 6: Evaluation results on DCA -Dy reaction distribution -

Pitch (cm)	Void fraction (%)	Benchmark experimental value 2nd ring	3rd ring	Exp. uncertainty	(C-E)/E (%)[(a)] 2nd ring	3rd ring
22.5	0	1.228	1.685		1.5	2.9
	30	1.219	1.701		2.0	2.7
	70	1.207	1.683	Systematic	1.2	1.6
	100	1.102	1.377	1.2%	-0.2	2.8
25.0	0	1.254	1.810	Random	2.2	2.8
	30	1.248	1.795	1.5%	2.2	3.7
	70	1.208	1.718		2.2	2.9
	100	1.112	1.444		-0.4	-0.7

(a) Relative statistical uncertainties of calculations are less than 0.1%.

4. Conclusion

JOYO and DCA data were evaluated and complied into benchmark problems for the International Reactor Physics Experiment Evaluation Project (IRPhEP).

Each of evaluated data has a unique feature and will be useful to validate reactor physics calculation schemes.

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