Analysis of RMWR (Reduced-Moderation Water Reactor) Mockup Experiments in FCA using JENDL-3.2 and JENDL-3.3

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Abstracts

The critical experiments have been carried out at the FCA to estimate the accuracy of prediction of the core characteristics in the design study of RMWR. A part of the experiments was analyzed by two different conventional deterministic methods: the SRAC code system and a standard calculation code system for a fast reactor, and a probabilistic method, MVP. The C/E values were compared between the JENDL-3.2 and JENDL-3.3 libraries. The calculation overestimates the criticality. The C/E values of the central fission rate ratios become larger in the order, F49/F25, F37/F25, F28/F25. For C28/F25 the calculation agrees with the measured one within twice of the experimental error. For the moderator void effect the calculation underestimates the measurement. When the C/E values are compared between JENDL-3.2 and JENDL-3.3, there is no large C/E discrepancy between JENDL-3.2 and JENDL-3.3 except for the criticality. For the criticality JENDL-3.3 gives larger C/E value than JENDL-3.2.

1. Introduction

To estimate the accuracy of prediction of core characteristics in the design study of RMWR[1), a program of critical experiments was planned at the fast critical facility, FCA. This program consisted of two phases; the first phase critical experiments in the mock-up core composed of uranium fuel plates and the second ones in the mock-up core composed of a combination of uranium and plutonium fuel plates to simulate MOX fuel. The principal aim of the first phase experiments is to study the basic characteristics of the RMWR core and that of the second phase is focused on the nuclear characteristics of the MOX fueled core. The first phase experiments were carried out between 2001 and 2002. The second phase ones are under way

These experiments have been analyzed by using JENDL-3.2[2)] and JENDL-3.3[3)] libraries and their results have been compared.

2. Brief Description of Mockup cores

The FCA-RMWR mockup core is a coupled system of a central test zone and a driver zone. The test zone is represented by a close to rectangular prism with about 38 cm in square base and 61 cm in height, as shown in Fig. 1. It is surrounded by the enriched U driver zone and two blanket zones; an inner blanket zone of 30cm in thickness containing a significant amount of depleted uranium oxide and sodium, and an outer blanket zone of 15cm in thickness containing only depleted uranium metal. The test zone is composed of a combination of uranium/plutonium fuel plates and moderator material (foamy polystyrene) plates to simulate the neutron energy spectrum of RMWR (Fig. 2). The principal cell averaged parameters of the test zone are shown in Table 1. The cell averaged fissile enrichment of the test zone is 15 atom % of ^{235}U and the hydrogen to nuclear fuel atomic number ratio (H/Fuel) is systematically changed from 0.1 to 0.8.

The measurements were made for criticality (keff value), central fission rate ratios, moderator void reactivity worth, central sample reactivity worth and ^{238}U Doppler effect.

3. Calculation method

The nuclear reactions in the RMWR core are dominantly occurred in the intermediate neutron energy range. Two different conventional deterministic methods, therefore, were used to analyze the experiments; the SRAC code system and a standard calculation code system for a fast reactor (FR code system). The cell averaged effective cross sections for each cell were obtained by the collision probability calculation with a one-dimensional infinite slab model and the group constant set generated from the JENDL-3.2 or JENDL-3.3 libraries. The effective cross sections in the resonance energy range were calculated by the table-look-up method of resonance shielding factors based on the narrow resonance approximation. The criticality and the forward and adjoint fluxes were calculated by the three-dimensional transport calculation code THREEDANT with P0-S8 approximation. The infinite dilution cross section of fission reaction for the detector was used in the central fission rate calculation while the cell averaged effective capture cross section was used in the central capture rate calculation,

In order to minimize the uncertainties of core geometrical modeling and data processing for the multi group cross sections generation, a probabilistic calculation system, a continuous-energy Monte Carlo code MVP, was also used to analyze the criticality (keff value) and the central fission rate ratios. In this analysis, the numbers of neutron histories were 2 millions and 5 millions for the criticality and the central reaction rate ratios with considering the geometrical model of the detector, respectively.

4. Comparison between Calculation and Experiment

The comparison between calculated (C) and experimental (E) results, C/E value, is shown in

Table 2 to 4.

Criticality

The calculation overestimates the criticality. In the deterministic methods, the anisotropic effect of neutron leakage, caused by the plate-type fuels and materials, cannot be considered. If the effect is considered, the calculated value will become about 1~2% smaller. When the results are compared between JENDL-3.2 and JENDL-3.3, JENDL-3.3 gives larger C/E values than JENDL-3.2.

Central reaction rate ratio

The C/E values of the central fission rate ratios become larger in the order, $^{239}Pu/^{235}U$ (F49/F25), $^{237}Np/^{235}U$ (F37/F25), $^{238}U/^{235}U$ (F28/F25). From these results, the calculation code systems have a tendency to give a harder neutron spectrum. There is no significant difference in the C/E values between the SRAC and FR systems except for the F49/F25 in the XXI-1D core. The MVP calculation with JENDL-3.3 gives large underestimation for the F49/F25 in the XXII-1 (65V) core. We need further investigation in the analysis of it. For the ratio of ^{238}U capture to ^{235}U fission rates (C28/F25), the MVP calculation agrees with the measured data within twice of the experimental error. When the C/E values of the C23/F25 are compared between JENDL-3.2 and JENDL-3.3, there is no large difference between them.

Reactivity Worth Measurement

As the moderator void effect, the reactivity worth caused by the void fraction change of polystyrene plates from 65% to 95% in the central cell of the test zone were measured. The calculation underestimates the measured value by 10% and more. The C/E values obtained by JENDL-3.2 agree with those by JENDL-3.3.

As the ^{238}U Doppler effect, the Doppler reactivity worth of the UO_2 sample due to sample temperature change from room temperature to 800°C was measured. The calculation with using JENDL-3.2 and JENDL-3.3 agrees with the experiment within twice of measurement error.

In the analysis of Pu sample reactivity worth, the SRAC system shows the C/E dependency on Pu contents. When the C/E values are compared between JENDL-3.2 and JENDL-3.3, there is no large discrepancy.

5. Summary

The critical experiments at the FCA have been carried out to estimate the accuracy of prediction of core characteristics in the design study of RMWR. The first phase experiments, of which the purpose is to study the basic characteristics of the RMWR core, were finished. The second phase experiments, which are focused on the nuclear characteristics of the MOX fueled core, are under way. The analyses were carried out by two different conventional deterministic methods, the SRAC code system and a standard calculation code system for a fast reactor (FR code system), and a probabilistic method, MVP. The C/E values were compared between the JENDL-3.2 and JENDL-3.3 libraries.

In the analysis of the criticality, the calculation overestimated the measurement and JENDL-3.3 gives larger C/E values than JENDL-3.2. For the central reaction rate ratio, the C/E values become larger in the order, F49/F25, F37/F25, F28/F25. On the other hand, the calculation agreed with the measured C28/F25 within twice of the experimental error. When the C/E values were compared between JENDL-3.2 and JENDL-3.3, there was no large difference between them except for the MVP calculation for the F49/F25 in the XXII-1 (65V) core. In the analysis of the reactivity worth measurement, the calculation showed the underestimation in the moderator void effect and the C/E dependency in the Pu sample worth. However, there was no large C/E discrepancy between JENDL-3.2 and JENDL-3.3.

The further detail analyses will be carried out to solve the problems, such as the underestimation in the moderator void effect and the C/E dependency in the Pu sample reactivity worth after completing the experiments in the other cores with different moderator voidage fraction.

References

1) S. Uchikawa : “Status and Future Program of Research and Development on Reduced-moderation Water Reactors”, in Summary of the 6th Workshop on the Reduced-moderation Water Reactor, JAERI-Conf 2003-020 (2003).
2) T. Nakagawa, *et al.* : *J. Nucl. Sci. and Technol.*, **32**, 1259 (1995).
3) S. Shibata, *et al.*: *ibid.*, **39**, 1125 (2002).

Table 1 Cell averaged parameters of the test zone in FCA RMWR mockup cores

Core name	1st phase	2nd phase			RMWR
	XXI-1 D2	XXII-1(45V)	XXII-1(65V)	XXII-1(95V)	
Enrichment (%)	15.2	15.8	15.8	15.8	10
Vm/Vf *	1.7	0.6	0.6	0.6	0.18
Void fraction (%)	80	45	65	95	68
H/Fuel **	0.50	0.81	0.52	0.091	

* Volume fraction of moderator to fuel plates in a cell

** Atomic number ratio between Hydrogen and Fuel materials in a cell

Table 2 Ratio of calculated to measured criticality (keff value)

Core name	SRAC		FR		MVP	
	J-3.2	J-3.3	J-3.2	J-3.3	J-3.2	J-3.3
XXI-1 D2	1. 0196 ±0.02%	-	1. 0155 ±0.02%	-	1. 0036 ±0.04%	-
XXII-1 (65V)	1. 0057 ±0.04%	1. 0071 ±0.04%	1. 0003 ±0.04%	1. 0013 ±0.04%	1. 0099 ±0.05%	1. 0107 ±0.04%

Table 3   Comparison of central reaction rate ratios between calculation and experiment

Reaction	Core name	SRAC		FR		MVP	
		J-3.2	J-3.3	J-3.2	J-3.3	J-3.2	J-3.3
F28/F25	XXI-1 D2	1.10	-	1.10	-	1.00±6%	-
	XXII-1 (65V)	1.18	1.17	1.18	1.17	1.20±8%	1.15±10%
F37/F25	XXI-1 D2	1.06	-	1.03	-	0.92±5%	-
	XXII-1 (65V)	1.06	1.07	1.07	1.07	1.08±7%	1.02±8%
F49/F25	XXI-1 D2	0.84	-	0.99	-	0.89±12%	-
	XXII-1 (65V)	0.99	0.99	1.00	1.00	0.92±11%	0.83±11%
C28/F25	XXII-1 (65V)	1.03	1.04	1.13	1.13	1.01±2%	1.05±2%

* Measurement error : F28/F25 ±6%, F37/F25 ±5%, F49/F25 ±4%, C28/F25 ±2%

Table 4   Comparison of reactivity worth between calculation and experiment

Item		Core name	SRAC		FR	
			J-3.2	J-3.3	J-3.2	J-3.3
Moderator void effect *		XXII-1 (65V)	0.84	0.85	0.87	0.87
U-238 Doppler effect ** (Sample: UO_2)		XXI-1 D2	(1.07) †	-	1.09	-
		XXII-1 (65V)	(0.83) †	(0.84) †	1.00	1.01
Pu sample ***	Pu (92)	XXII-1 (65V)	0.92	0.92	0.95	0.95
	Pu (81)	XXII-1 (65V)	1.02	0.96	1.10	1.05
	Pu (75)	XXII-1 (65V)	1.17	1.15	0.92	0.90

* Reactivity worth due to moderator void fraction change from 65% to 95% in the central cell of the test region.

** Doppler sample reactivity worth due to sample temperature change from room temperature to 800 °C.

*** Pu (92) : 239/240/241/242=91.7/8.0/0.2/0.1, Pu (81) : 239/240/241/242=80.3/18.6/0.7/0.4, Pu (75) : 239/240/241/242=73.0/23.1/1.7/2.2.

† Calculation was carried out by the diffusion code (CITATION) with 2-dimensional model.

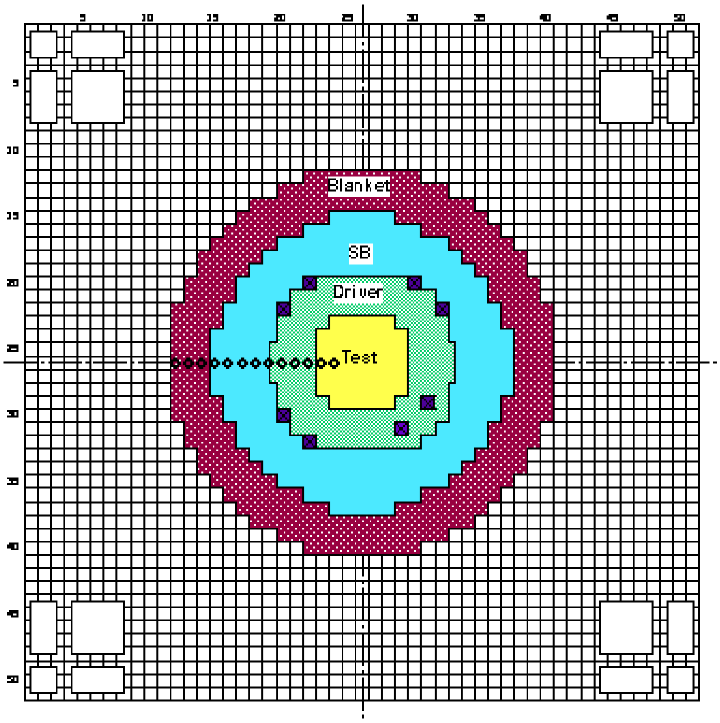

Fig. 1 Cross-section view of the first phase FCA-RMWR core

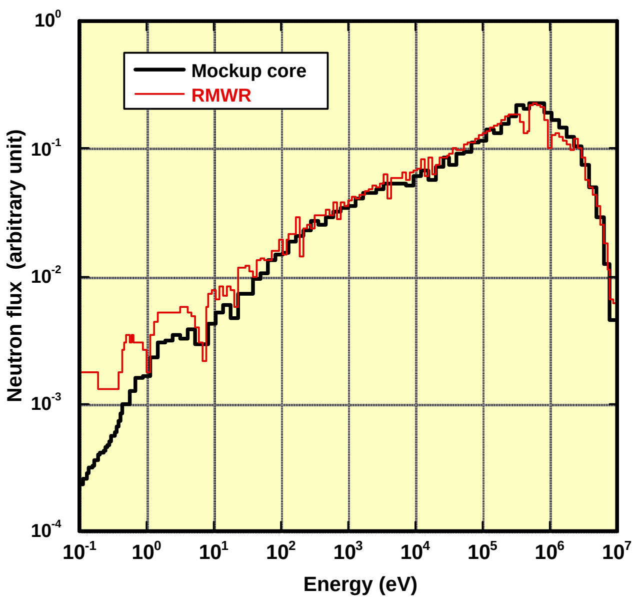

Fig. 2 Calculated neutron energy spectra