MONJU Experimental Data Analysis and its Feasibility Evaluation to Build up the Standard Data Base for Large FBR Nuclear Core Design

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Abstract

MONJU experimental data analysis was performed by using the detailed calculation scheme for fast reactor cores developed in Japan. Subsequently, feasibility of the MONJU integral data was evaluated by the cross-section adjustment technique for the use of FBR nuclear core design.

It is concluded that the MONJU integral data is quite valuable for building up the standard data base for large FBR nuclear core design. In addition, it is found that the application of the updated data base has a possibility to considerably improve the prediction accuracy of neutronic parameters for MONJU.

KEYWORDS: MONJU, Proto-type FR, FBR, Data base, Nuclear core design, Cross-section adjustment, Power plant, Pin-type, High order plutonium

1. Introduction

MONJU is the proto-type fast reactor in Japan [1]. Initial criticality was achieved in April of 1994 and the first start-up core was assembled in May of 1994. A series of reactor physics parameters on the initial core was measured by November of 1994 as a part of the system start-up tests. In the tests, such neutronic parameters as criticality and control rod worth were measured [2]. These data must be considerably valuable for an improvement of accuracy and reliability for nuclear core design of future innovative reactor as the current data base is poor in number of integral data of nuclear power plants and similarity to such a reactor. The present paper describes the results of analyses on MONJU reactor physics experiments using the deterministic detailed calculation scheme for fast reactor cores developed in Japan. In addition, their consistency evaluation with formerly evaluated integral data, which are represented by JUPITER data, will be reported in terms of extension of the standard data base for large FBR nuclear core design.

2. Brief Description of MONJU

MONJU is a middle-scale sodium-cooled fast reactor, which generates the electric output of 280MWe. Table 1 presents the major specification of the MONJU reactor [1]. The reactor core

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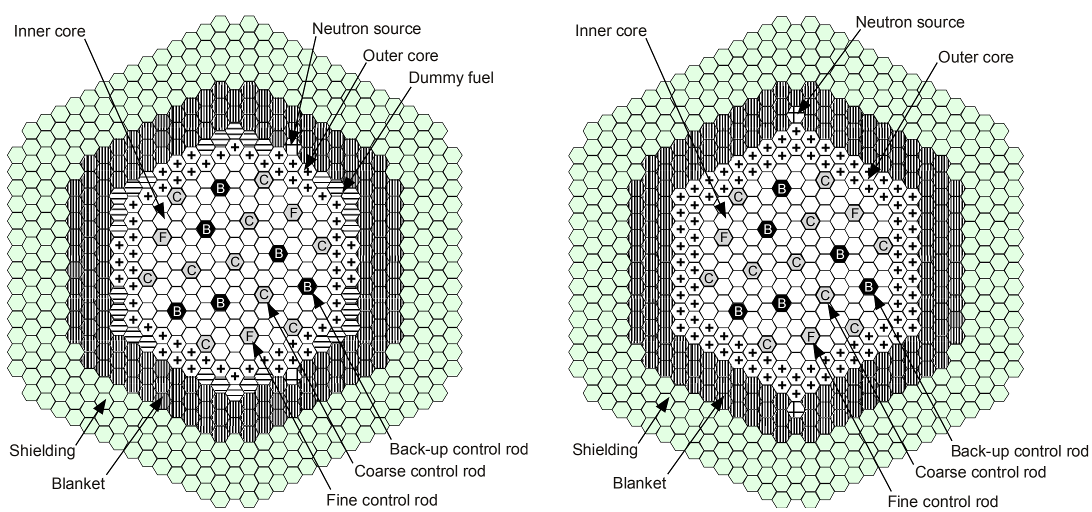
consists of driver fuel region, blanket fuel region and control rod positions or subassemblies. In the subassemblies of driver and blanket fuel regions, mixed oxide (PuO_2-UO_2) or uranium oxide (UO_2) fuel pellets are stacked in the fuel pins and 169 or 61 fuel pins are bundled inside the wrapper tube. The driver fuel region is zoned into inner and outer cores by differing the plutonium fraction to flatten the radial power distribution. As for control rod subassemblies, 3 types of control rods are applied and their major specification is shown in Table 2. Pellets of ^{10}B-enriched boron carbide are stacked in the absorber pins and 19 absorber pins are bundled inside the protection tube, which are moved in the guide tube.

During the system start-up tests, Initial Critical (IC-) core and First Start-up (FS-) core were constructed, which are illustrated in Fig. 1 [2]. IC-core consists of 168 driver fuel and 30 dummy fuel (made of stainless steel) subassemblies, and excess reactivity was compensated only by half-insertion of the central control rod. Therefore, IC-core can be recognized as control rod free. On the other hand, FS-core was constructed by replacing all dummy fuel subassemblies in IC-core with outer core fuel subassemblies, and one of critical states was settled by one-third-insertion of all coarse and fine control rods.

Table 1: Major specification of the MONJU reactor

Items	Normal operation state	During system start-up tests	
		IC-core	FS-core
Thermal output [MW]	714	0	0 - 321
Core height [m]	0.93	0.93	0.93
Core equivalent diameter [m]	1.79	1.66	1.79
Blanket thickness (Upper/Lower/Radial) [m]	0.30/0.35/0.30	0.30/0.35/0.30	0.30/0.35/0.30
No. of fuel subassemblies (Inner/Outer cores)	108 / 90	108 / 60	108 / 90
Subassembly pitch [mm]	115.6	115.6	115.6
Pu fissile fraction (Inner/Outer cores) [wt%]	16 / 21	15 / 20	15 / 20



Initial Critical Core (IC-core or ICC) First Start-up Core (FS-core or FSC)

Fig. 1: Core layouts of the MONJU initial critical and first start-up cores

Table 2: Major specification of the MONJU control rod subassemblies

Items	Coarse	Fine	Back-up
No. of subassemblies	10	3	6
^{10}B fraction [wt%]	39	39	90
Absorber stack length [m]	0.80	0.80	0.93
Absorber volume fraction [%]	19	19	26

In a series of the MONJU reactor physics tests, several neutronic parameters were measured. Major neutronic parameters are criticality, control rod worth, fixed absorber worth, fuel subassembly worth, reaction rate distribution, isothermal temperature coefficient, coolant reactivity and burn-up coefficient. In the present study, criticality and control rod worth were analyzed because experimental uncertainty of these parameters is favorable for consistency evaluation with formerly evaluated integral data. Regarding the criticality, as IC-core is control rod free, lower analytical modeling uncertainty can be promised in comparison with that of FS-core. On the other hand, analysis for FS-core, which is a core with control rods insertion, is suitable for evaluating the accuracy of core parameters in the normal operation state of power reactors. Concerning the control rod worth, central one is chosen because it was measured by the asymptotic period method and interaction by other control rod movement was almost negligible.

3. Experimental Data Analysis

3.1 Method of Analysis

MONJU experimental data analysis was performed by using the detailed deterministic calculation scheme for fast reactor cores developed in Japan [3]. In the cell calculation, JFS-3-J3.2R was used, which is the 70 neutron energy groups nuclear constant set based on the JENDL-3.2 [4], because it has been applied for the compilation of the current standard data base for large FBR nuclear core design developed in Japan. Cells of fuel region were heterogeneously constructed in the 1-dimensional (1D) multi-ring model as shown in the left side of Fig. 2, whose applicability has been validated by using the 2D exact cell model with deterministic and Monte Carlo calculations [5, 6]. In addition, cells of control rod region were also constructed with the 1D multi-ring model as illustrated in the right side of Fig. 2, which was verified by Monte Carlo calculations for whole core geometry [6]. Self-shielding effect was treated by the table look-up method with back-ground cross-sections obtained by Tone’s formula [7]. Cross-sections were homogenized by typical flux-weighted method for cells of fuel region and by the reaction rate ratio preservation method [8] for cells of control rod region.

Base core calculation was made for the 3D Triangle-Z (Tri-Z) geometry with the 70-group structure by using the diffusion theory. For best estimation, spatial mesh-size, transport theory and ultra-fine group corrections were taken into account. Spatial mesh-size correction was estimated by comparing the base mesh-size result with finer or coarser mesh-size results. Transport theory correction was evaluated by using the 3D Cartesian (XYZ) geometry model with the finite difference method [9], which was compared with those calculated for the 3D Hexagonal-Z (Hex-Z) geometry by the nodal transport method [10-12] to ensure the reliability. Further ultra-fine group correction, which is to reduce the truncation error caused by application of the

70-group constant set, was calculated by the cell calculation with about 100,000-group structure (ultra-fined below 50keV) and the subsequent core calculation with the VITAMIN-J type 175-group structure [13, 14]. For the calculation of effective delayed neutron fraction, yield data evaluated by R.J.Tuttle [15] was adopted.

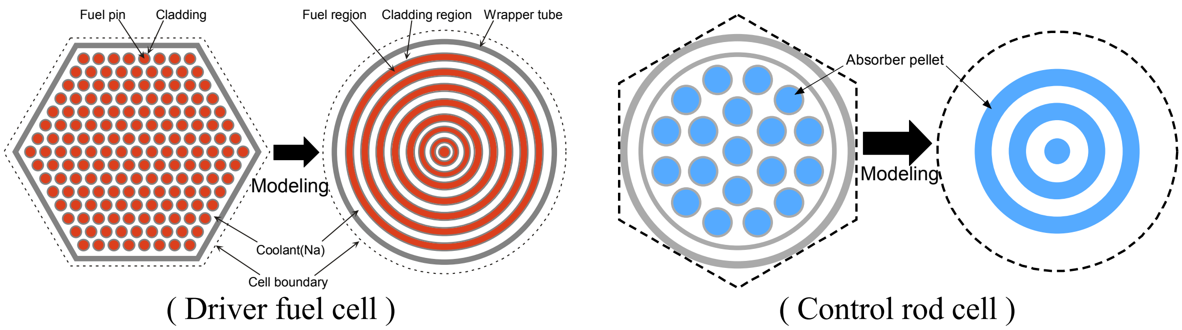


Fig. 2: The cell modeling to treat the fuel and control rod heterogeneity

3.2 Result of Analysis

Table 3 presents the analytical result on the criticality for both IC-core and FS-core. A slight underestimation was observed compared with experimental value as well as most of criticality in the data base. Larger C/E value by 0.001 in FS-core would be caused by the underestimation of reactivity by one-third-inserted control rods as shown in the result of control rod worth analysis.

Table 3: Result of the criticality for MONJU cores

Items	IC-core	FS-core
C/E value (Before correction)	0.9892	0.9914
Ultra-fine group correction	+0.0002	+0.0001
Transport correction	+0.0072	+0.0070
Spatial mesh-size correction	-0.0014	-0.0021
C/E value (After correction)	0.9952	0.9964

Table 4 describes the analytical result on the central control rod worth for FS-core. C/E value after all corrections was a few percent lower than unity, however, it agreed well with that obtained by other calculation scheme based on the same cross-section library JENDL-3.2 [16], therefore, the present result would be quite reliable.

Table 4: Result of the central control rod worth for MONJU FS-core

Items	Central
C/E value (Before correction)	1.007
Ultra-fine group correction	0.999
Transport and spatial mesh-size correction	0.969
Correction for control rod interaction effect	0.990
C/E value (After correction)	0.965

4. Feasibility Evaluation for the Nuclear Core Design

In the present chapter, the feasibility of MONJU integral data will be discussed for the FBR nuclear core design. That is, the analyzed data on MONJU reactor physics experiments are evaluated in terms of their utilization for an improvement of the prediction accuracy of neutronic parameters for future innovative fast reactor cores. Detailed discussion is described in the following sections.

4.1 Method of Feasibility Evaluation

For addition of new integral data to nuclear data base, their verification or consistency evaluation with formerly evaluated integral data is essential. Consistency can be evaluated in terms of experimental, analytical modeling and cross-section induced uncertainties. The cross-section adjustment technique [17] is one of the most effective methods to evaluate the consistency among integral data. In the cross-section adjustment, cross-sections and their covariance would be adjusted so as to maximize the probability that the cross-sections could give more reliable calculation (C) values comparing with experimental (E) values based on the Bayesian theory.

In the present study, experimental and analytical modeling uncertainties were tentatively estimated according to the JAEA's accumulated experience in cross-section adjustment study. Cross-section induced uncertainty was obtained by using the covariance data evaluated for JENDL-3.2 [18], which were processed into group-wise data by a covariance processing code ERRORJ [19]. Table 5 shows the summary of MONJU parameters applied to the cross-section adjustment. In the present paper, all uncertainties are specified by 1 standard deviation.

Table 5: Summary of MONJU parameters applied to the cross-section adjustment

Items	Criticality		Central control rod worth (FS-core)
	(IC-core)	(FS-core)	
C/E value (Before adjustment)	0.9952	0.9964	0.965
Experimental uncertainty	±0.0004	±0.0004	±0.012
Analytical modeling uncertainty	±0.0023	±0.0023	±0.010
Cross-section induced uncertainty	±0.0087	±0.0084	±0.025

For uncertainty of delayed neutron yield, Tuttles's evaluation [15] was applied as well as nominal data. Consistency can be evaluated from the adjusted C/E values, that is, consistency can be judged from the alterations of C/E values to near unity by the adjustment. Formerly evaluated integral data for cross-section adjustment were selected according to the experience in the preparation of the adjusted nuclear constant set ADJ2000 [17]. Table 6 summarizes the integral data used for the present cross-section adjustment. Most of integral data were obtained by JUPITER [20-22], which is the joint program of critical experiments promoted by US-DOE and PNC (the former of a part of JAEA) using the ZPPR facility and various types of middle- and large-size core were constructed in the program.

Table 6: List of the integral data used for the present cross-section adjustment

Cores	Core parameters			Major core features	
	Criticality	Control rod worth	Others	Scale (Size)	Fuel*1
ZPPR (9 - 17)	15	18	131	Middle	Pu
ZPPR (18, 19)	4	17	14	Large	Pu & EU
FCA (X-I, XVII-I)	2	-	7	Small	Pu
JOYO (MK-I)	1	-	6	Small	Pu & EU
MASURCA (CIRANO)	-	-	2	Middle	Pu
BFS-2 (BFS-58-i-I1)	1	-	7	Middle	Pu & EU
BFS-2 (BFS-62-1, 2)	2	1	-	Middle	EU
Los Alamos	5	-	-	Very small	EU, Pu
MONJU	2	1	-	Middle	Pu

*1: EU means Enriched Uranium. Most of cores includes Depleted Uranium (DU).

4.2 Result of Feasibility Evaluation

Major results of the cross-section adjustment were shown in Fig.4 for criticality and Fig.5 for control rod worth, respectively. No obvious difference was observed between adjustments with and without the MONJU integral data. These results indicate the consistency between MONJU and other fast reactor cores in integral data because C/E values of MONJU can be reasonably adjusted without any unreasonable degradation of C/E values on other fast reactor cores. Further, the adjustment reduced cross-section induced uncertainty of MONJU core parameters to one eighth for criticality and to one third for control rod worth in comparison with that before cross-section adjustment.

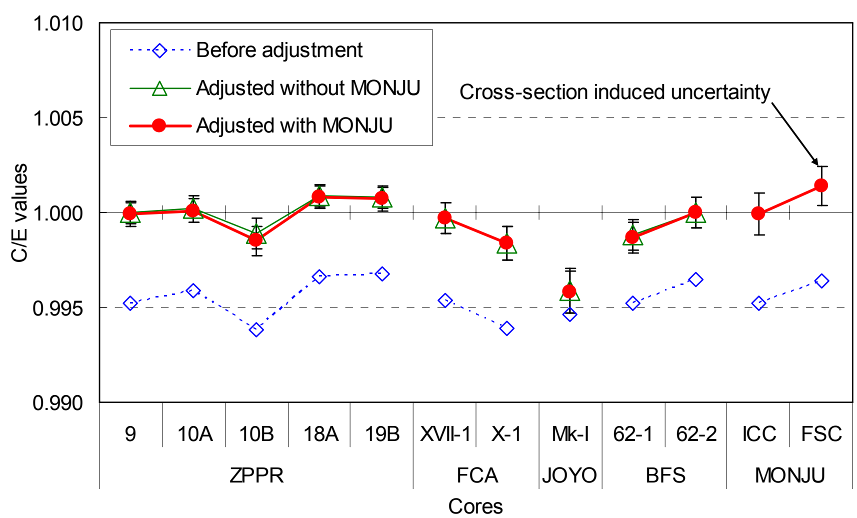


Fig. 4: Results of the cross-section adjustment for criticality

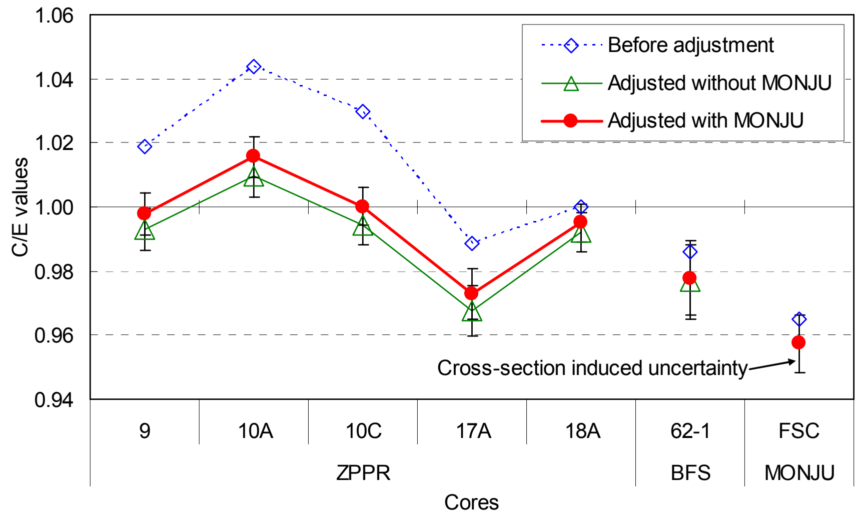


Fig. 5: Results of the cross-section adjustment for central control rod worth

Major influences in the cross-section adjustment due to the addition of the MONJU data are observed in (n,α) reaction cross-section of ^{10}B and delayed neutron yield (Symbolized by β in figures) of high order plutonium. As for former influence, the adjustment of ^{10}B (n,α) reaction cross-section was increased twice as much as that without MONJU data as shown in Fig. 6, which would be caused by newly adding the worth data of control rod with ^{10}B-enriched boron because such an integral data has not been included in the previous adjustment. Concerning the latter one, the adjustment result of ^{241}Pu delayed neutron yield is shown in Fig. 7 as a representative result. The adjustment of ^{241}Pu delayed neutron yield was comparatively large, which was supporting the newer value [23].

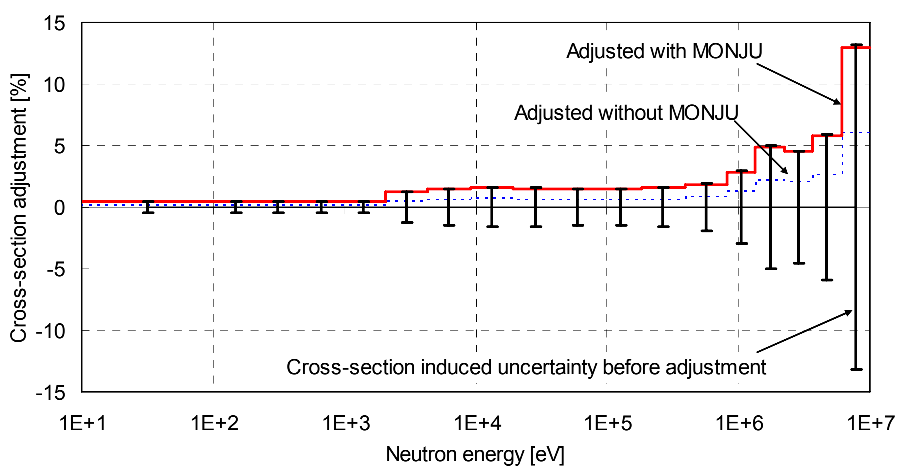


Fig. 6: Result of the adjustment for ^{10}B (n,α) reaction cross-section

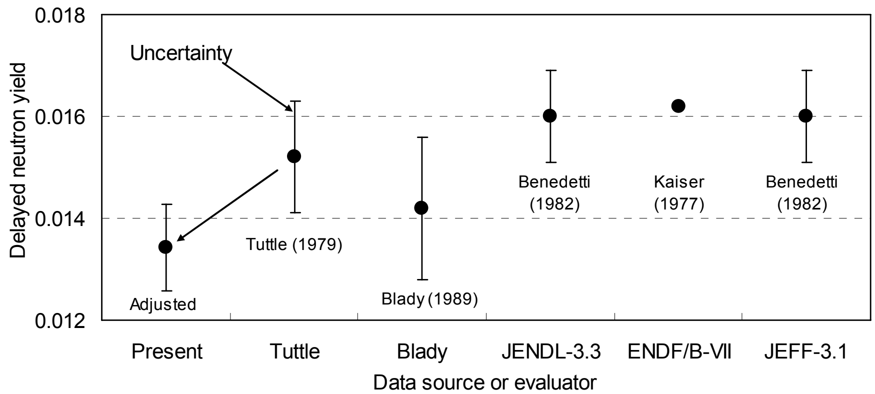
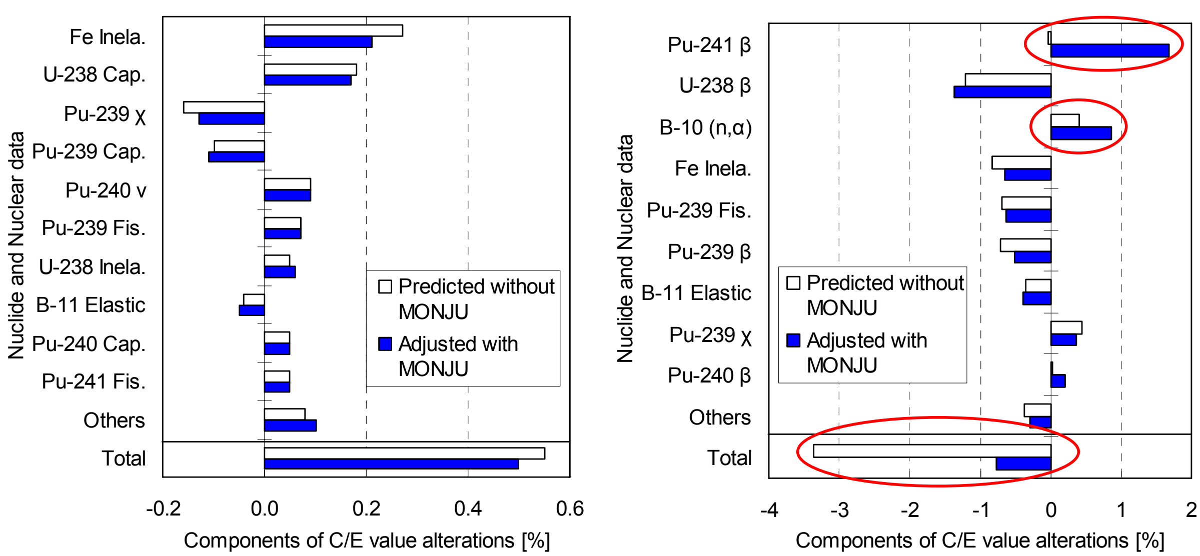


Fig. 7: Result of the adjustment for ^{241}Pu delayed neutron yield

Figure 8 shows the comparison in the C/E value alterations and their components for MONJU FS-core parameters between predicted one without MONJU and adjusted one with MONJU. As for the criticality, there is little difference between 2 adjustments, which means that the current data base can well predict the MONJU criticality without MONJU integral data. On the contrary, for the control rod worth the difference of 3% in C/E value alteration is observed and it is dominated by the adjustment of ^{241}Pu delayed neutron yield. This result tells that ^{241}Pu delayed neutron yield has not been effectively adjusted before addition of the MONJU control rod worth data, therefore, it is found that the MONJU control rod worth data is superior in evaluation of the ^{241}Pu delayed neutron yield because the MONJU control rod worth has sufficient sensitivity to ^{241}Pu fission reaction and its experimental and analytical uncertainties are adequately small. Thus, it is clarified that the MONJU integral data are quite valuable to reinforce the reliability of ^{10}B and high order plutonium nuclear data.



(Criticality) (Central control rod worth)

Fig. 8: Comparison in the C/E value alterations and their contributions for MONJU FS-core parameters by cross-section adjustments between without and with MONJU

4. Conclusion

The present study clarified that the addition of the MONJU integral data improves the accuracy of the nuclear data, which is represented by ^{10}B and high order plutonium. Therefore, it is concluded that the MONJU integral data is quite valuable to build up the standard data base for large FBR nuclear core design. In addition, the adjustment reduced cross-section-induced uncertainty of MONJU core parameters to one eighth for criticality and to one third for control rod worth in comparison with that before cross-section adjustment. Thus, it is found that the application of the updated data base has a possibility to considerably improve the prediction accuracy of neutronic parameters for MONJU.

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References

1) F.Nakashima, Y.Kaise, et al.: “Core Performance and Characteristics of the Prototype Fast Breeder Reactor MONJU”, Proc. Inter. Conf. on Fast Reactors and Related Fuel Cycles (FR’91), Vol.**1**, pp.8.2-1 - 9, Kyoto, Japan (1991).
2) S.Sawada, T.Suzuki, et al.: “Reactivity Characteristics Evaluations of the Initial Core of MONJU”, Proc. Inter. Conf. on the Physics of Reactors (PHYSOR 96), Vol.**2**, ppE-76 - 85, Mito, Japan (1996).
3) M.Ishikawa: “Consistency Evaluation of JUPITER Experiment and Analysis for Large FBR Cores”, Proc. Inter. Conf. on the Physics of Reactors (PHYSOR 96), Vol.**2**, ppE-36 - 45, Mito, Japan (1996).
4) T.Nakagawa, K.Shibata, et al.: “Japanese Evaluated Nuclear Data Library Version 3 Revision-2: JENDL-3.2”, J. Nucl. Sci. Technol., **32** [12], pp.1259-1271 (1995).
5) K.Yokoyama, J.Tommasi, et al.: “Analyses of Experiments in the JOYO Fast Reactor using the ERANOS and JNC Code Systems”, Proc. Inter. Conf. on the New Frontiers of Nuclear Technology: Reactor Physics, Safety and High-Performance Computing (PHYSOR 2002), Seoul, Korea (2002).
6) Y.Nagaya, K.Okumura, et al.: “MVP/GMVP II: General Purpose Monte Calro Codes for Neutron and Photon Transport Calculations based on Continuous Energy and Multigroup Methods”, JAERI 1348 (2005).
7) T.Tone: “A Numerical Study of Heterogeneity Effects in Fast Reactor Critical Assemblies”, J. Nucl. Sci. and Technol., **12** (8), P.467 (1975).
8) T.Kitada, S.Kosaka, et al.: “New Control Rod Homogenization Method for Fast Reactors”, J. Nucl. Sci. Technol., **31** [7], pp.647-653 (1994).
9) T.Yamamoto and T.Takeda: “Present Status of “TRITAC” Code Development and its Application to Fast Reactor Calculations”, Proc. Inter. Conf. on Mathematics and

Computations, Reactor Physics, and Environmental Analyses, Vol.**1**, pp.470-479, Portland, Oregon (1995).

10) H.Ikeda and T.Takeda: “A New Nodal S_N Transport Method for Three-Dimensional Hexagonal Geometry”, J. Nucl. Sci. Technol., Vol.**31**, No.6, 497 (1994).

11) K.Sugino and T.Takeda: “An Improvement of the Transverse Leakage Treatment for Nodal S_N Transport Calculation Method in Hexagonal-Z Geometry”, J. Nucl. Sci. Technol., Vol.**33**, No.8, 620 (1996).

12) K.Sugino and T.Takeda: “A New Method of Treating Radial Leakage in Nodal S_N Transport Calculation for Hexagonal Geometry”, Proc. Inter. Conf. on the Physics of Reactors (PHYSOR 96), Vol.**1**, ppE-153 - 162, Mito, Japan (1996).

13) K.Sugino: “JUPITER Experimental Analyses Using a New Constant Set Based on JENDL-3.2”, J. Nucl. Sci. Technol., Supplement **2**, pp.1002-1005 (2002).

14) T.Hazama, G.Chiba, et al.: “Development of SLAROM-UF; a Cell Calculation Code with Fine and Ultra-fine group calculation”, J. Nucl. Sci. Technol. *(to be published)*.

15) R.J.Tuttle: “Delayed Neutron Yield in Nuclear Fission, Consultants”, Mtg. on Delayed Neutron Properties, IAEA, Vienna, INDC(NDS)-107/G+Special, P.29 (1979).

16) T.Kageyama, J.Tommasi et al.: “A Comparative Study of the MONJU Fast Reactor Physics Tests with the ERANOS and JAEA Code Systems”, Proc. Inter. Conf. on the Advances in Nuclear Analysis and Simulation (PHYSOR 2006), Vancouver, Canada (2006).

17) M.Ishikawa, K.Sugino, et al.: “Development of a Unified Cross-section Set ADJ2000 based on Adjustment Technique for Fast Reactor Analysis”, J. Nucl. Sci. Technol., Supplement **2**, pp.1073-1076 (2002).

18) K.Shibata, A.Hasegawa et al.: “JENDL-3.2 Covariance File”, J. Nucl. Sci. Technol., Supplement **2**, pp.40-43 (2002).

19) G.Chiba: “ERRORJ - Covariance Processing Code Version 2.2-“, JNC TN9520 2004-003 (2004).

20) M. J. Lineberry, S. G. Carpenter et al.: “Experimental Studies of Large Conventional LMFBR Cores at ZPPR”, Proc. of Fast Reactor Physics 1979, Aix-en-Provence, France, IAEA-SM-244/86, Vol.**1**, P.187 (1979).

21) H. F. McFarlane, S. G. Carpenter et al.: “Experimental Studies of Radially Heterogeneous Liquid-Metal Fast Breeder Reactor Critical Assemblies at the Zero-Power Plutonium Reactor”, Nucl. Sci. Eng., **87**, P.204 (1984).

22) P. J. Collins and S. B. Brumbach: “Experiments and Analysis for an Axially Heterogeneous LMR Assembly at ZPPR”, Proc. of the 1988 Int. Reactor Physics Conf., Jackson Hole, Wyoming, **2**, P.309 (1988).

23) M.C.Brady and T.R.England: “Delayed Neutron Data and Group Parameters for 43 Fissioning Systems”, Nucl. Sci. Eng. Vol. **103**, pp129-149 (1989).