Recent Application of Nuclear Data to Fast Reactor Core Analysis and Design in Japan

Makoto Ishikawa

Japan Nuclear Cycle Development Institute (JNC), 4002, Narita-cho, O-arai-machi, Ibaraki, 311-1393, Japan
e-mail: ishikawa@oec.jnc.go.jp

Abstract. Some improvements of the nuclear data application to the fast reactor core field have been accomplished in Japan. The present paper introduces the recent status from three categories, group constant and analytical tools, database of fast reactor experiments, and application to FBR design study. A policy of JNC is to open the products generated from the application study to the world as much as possible to keep the traceability, reproducibility, and accountability.

INTRODUCTION

In Japan, the effort to develop the large fast breeder reactors is still continuing in the various technical fields in order to realize the safe, economic, uranium-resource saving, environmental burden-reducing, and non-proliferating FBR plant concepts, under the name of "FBR Cycle Feasibility Study" that is strongly related to the international cooperation of Generation-IV project. The improvement and application work of nuclear data are being performed in the context of the national strategy. The purpose of the present paper is to introduce the recent status of nuclear data application to fast reactor cores in Japan. The contents are classified into three categories; one is the development of group constants and analytical tools related to nuclear data, the next is the consistent evaluation of the Japanese Evaluated Nuclear Data Library (JENDL) by the fast reactor experimental database, and the last is the improvement of prediction accuracy and application to fast reactor design study.

GROUP CONSTANTS AND ANALYTICAL TOOLS RELATED TO NUCLEAR DATA

Generation of a New Group Constant Structure and its Effects

In the last three decades, the Japanese standard format of group constant sets for fast reactor analysis was based on the 70-group Russian ABBN-type [1] structure with self-shielding factor tables. While the format was quite accurate in treating the self-shielding effect and also very efficient from a computer-performance viewpoint then, it is now recognized that there is some room for improvement. One is the small number of energy groups, 70, applying the various types of fast reactors with different neutron spectra. Traditionally, the future commercial FBR core in Japan was considered only as the sodium-cooled MOX-fueled type; therefore, the weighting function to generate the 70-group constant from JENDL was always the neutron spectrum of one typical FBR core, actually the prototype reactor MONJU. However, Japan and the international society are now pursuing the possibility of various FBR types such as metal- or nitride-fueled, and Pb/Bi-, gas-, or even water-cooled FBR cores. In such a situation, the 70-group constant, which assumes only one type of neutron spectrum, is not suitable for the design study.

The other is the insufficiency of the resonance-peak interaction treatment among different nuclides in the cell calculation. When an ABBN-type group constant is generated, the background cross section from other co-existent isotopes is treated as flat within an energy group. To remedy this defect of the ABBN format, an extra parameter, the so-called R-parameter [2], was introduced with which the resonance interaction effect of only three important nuclides, U-235, U-238, and Pu-239, was roughly considered by the number density ratios of them in the cell. The recent investigation showed those shortcomings sometimes resulted in unacceptably large errors compared with the high requirement for modern nuclear analysis.

CP769, *International Conference on Nuclear Data for Science and Technology,*
edited by R. C. Haight, M. B. Chadwick, T. Kawano, and P. Talou

After some surveys of other group constant formats like the sub-group method [3], JNC concluded to use the combination of an ultra-fine energy group structure and a fine ABBN-format, which is called SLAROM-UF format [4] hereafter. Below 50 keV, where all resonance peaks in JENDL are covered except the U-238 unresolved resonance, approximately 100,000-group constant structure is applied. As depicted in Fig. 1, individual resonance peaks are expressed exactly as the original nuclear data library with these ultra-fine energy groups.

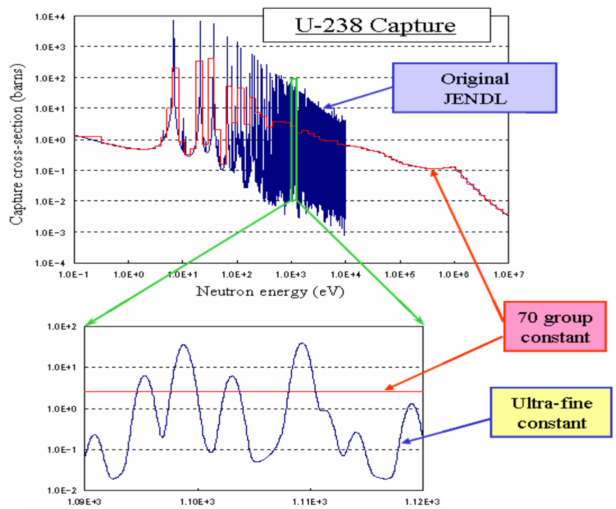

FIGURE 1. Ultra-fine group structure compared with 70-group constant (U-238 capture cross-section, JENDL-3.2)

Above 50 keV, where the cross-section energy dependence of structural materials such as iron, chromium, or nickel is very important in FBR cores, a 900-group ABBN-type structure is used.

The effect of the SLAROM-UF group constant is demonstrated in Table 1. It can be found that the improvement of prediction accuracy is significant for sodium-void reactivity and sample Doppler reactivity, which are sensitive to the evaluation of the resonance-shielding effect, though the magnitude also depends on the kind of experimental cores. The SLAROM-UF group constant based on JENDL-3.2 [5] and the related new cell code were opened to the public in April 2004.

Development of a Nuclear-Data Covariance Processing Code

Recently in Japan, it has been recognized that the covariance of evaluated nuclear data is indispensable to rationally evaluate the prediction accuracy of reactor core parameters with clear accountability. In the past, some utility codes were developed to process the covariance data in the ENDF format library such as the ERRORR module in NJOY system [6] or the PUFF code developed by ORNL, but their performances are not always satisfactory to treat the recent versions of JENDL. In order to utilize the covariance data [7] in JENDL, JNC has developed and released a covariance processing code ERRORJ [8] that is positioned as an extended version of the ERRORR module. The ERRORJ code possesses the new capability to process the covariance data of the Reich-Moore resolved-resonance parameters, unresolved-resonance parameters, P1 components of elastic scattering for scattering average cosine used in core calculation, and secondary neutron energy distributions of fission reaction. An example of covariance data from various libraries processed by ERRORJ is shown in Fig. 2. Curiously, the values of ENDF and JEF are identical in the whole energy range. The detail of ERRORJ will be reported in another presentation at this conference.

TABLE 1. Effect of the new group constant, SLAROM-UF format, compared with the ABBN 70-group constant.

Nuclear Parameters Measured	Experimental Cores	C/E (Calculation/Experiment) Values	
		Old (ABBN, 70-group)	**New (SLAROM-UF)**
Criticality	ZPPR-9	0.9934	0.9955
	BFS-62-3A	0.9968	0.9985
Sodium-Void Reactivity	ZPPR-9	1.03	0.98
	BFS-62-3A	0.71	0.97
Sample Doppler Reactivity	ZPPR-9	0.91	0.96
	BFS-62-4	0.86	0.95

Development of a Cross-Section Sensitivity Analysis System

The cross-section sensitivity coefficient for reactor core parameters is considered a very powerful tool to understand quantitatively the physical mechanism of nuclear data library changes tracing the components of nuclides, reactions, and energy ranges. Further, the sensitivity coefficient combined with cross-section covariance enables us to evaluate the prediction error of reactor core parameters, and to improve the design accuracy.

To calculate the cross-section sensitivity coefficient in Japan, the SAGEP code based on the generalized perturbation theory [9] had been developed by Osaka University. JNC introduced SAGEP in early 90s and implemented the capability to calculate the sensitivity of scattering average cosine, fission spectrum, and excited level-wise inelastic

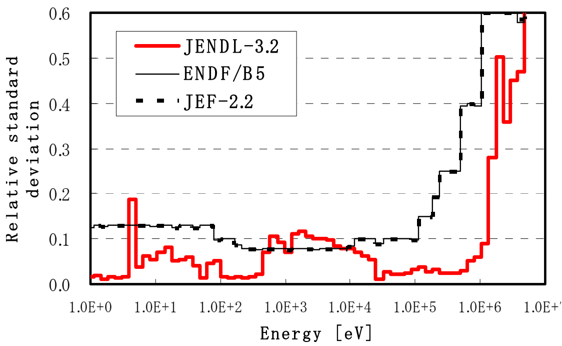

FIGURE 2. Various libraries' standard deviations of the U-235 capture cross section processed by the ERRORJ code.

scattering matrix for the fast reactor core analysis application. Further, in cooperation with Osaka University and Hitachi Ltd., JNC has completed: (1) the sensitivity calculation system for burnup-related core parameters such as fuel component changes by power operation or burnup reactivity loss, and (2) the sensitivity calculation system for Doppler reactivity, which is needed to introduce the self-shielding factors in the generalized perturbation theory. Figure 3 shows the sensitivity coefficients for the reactivity of a sample Doppler experiment in ZPPR. The sensitivity analysis system [10] developed by JNC is opened to the public.

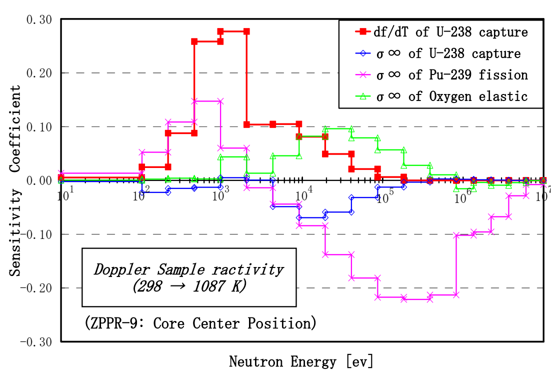

FIGURE 3. Sensitivity coefficient for a sample Doppler reactivity of ZPPR experiment (sample material: natural uranium dioxide).

CONSISTENT EVALUATION OF NUCLEAR DATA BY THE DATABASE OF FAST REACTOR EXPERIMENTS

Since the accuracy and performance of the recent evaluated nuclear data library, that is, JENDL in Japan, has been improved significantly by the continuous effort of nuclear-data researchers, the importance of the integral experimental data is also increasing to verify the quality of a library. In JNC, a standard database of fast reactor core experiments has been compiled both for the verification of nuclear data in a consistent manner and for the study of reactor physics analytical methods. Further, a hot topic of nuclear data in reactor application is related to the isotopic composition changes of fuel and the accumulation and/or transmutation of minor actinides (MAs) by core burnup from the viewpoint of the environmental burden-reduction and the development of innovative fuel-cycle schemes.

A Standard Database of Fast Reactor Experiments in Japan

For the last ten years, JNC has made an extraordinary effort to collect the fast reactor experimental data from their original documents, and has continued to evaluate those data by applying JENDL and the most detailed analytical methods. The work to compile the fast reactor database has almost been completed and the contents are summarized in Table 2. Some of these data such as ZPPR and JOYO, which can be released by JNC, are being prepared to provide the archiving activity of OECD/NEA, "International Reactor Physics Benchmark Experiments Project (IRPhE)" to leave these precious data as a fortune of the next generation.

JNC has evaluated these experimental data consistently by JENDL-3.2 and the most detailed analytical method of JNC. The main features of analysis are: 1) a 70-group ABBN-type constant set as the basic nuclear data, 2) cell heterogeneity treatment by Tone's method based on the collision probability, 3) cell streaming modeling by the Benoist's anisotropic diffusion coefficient, 4) three-dimensional whole core diffusion calculation, 5) correction by neutron transport theory and mesh-size effect, 6) correction by fine group calculation with the SLAROM-UF format. The performance of the JENDL-3.2 library and the JNC analytical method is summarized in Table 3 with the form of rough C/E values for various core parameters and facilities. The criticality of large MOX cores is a little underestimated by 0.5-0.7%. Reaction rate distribution is satisfactory, while the U-238 capture/Pu-239 fission ratio(an index of breeding ratio) may have room for improvement. The major reactivity parameters are predicted well enough to make a rational fast reactor core design.

Further, the Japanese Nuclear Data Committee (JNDC) released the latest JENDL, version 3.3 [16], in May 2002. The effects of nuclear data library changes from JENDL-3.2 to 3.3 are shown in Table 4 for some typical core parameters. Generally the impact of the JENDL revision is not significant for fast reactor cores,

TABLE 2. A standard database of fast reactor experiments in Japan (compiled by JNC up to 2004).

Facility (Institute, Country)	Name of Experimental Core (Total Number)	Core Features	Core parameters*1) collected in database	Open to Public
ZPPR <JUPITER Program> (ANL-W, USA)	**ZPPR-9, 10A -10D/2** (7)	600-800 MWe-class, two-region homogeneous MOX cores.	keff, RR, CRW, SVR, SSW, and DR(sample).	**Yes.**[11]
	ZPPR-13A, 13B/1-13B/4, 13C (6)	650 MWe-class, Radially-heterogeneous MOX cores.	keff, RR, CRW, SVR, SSW, DR(sample), and ZMRR.	
	ZPPR-17A, 17B, 17C (3)	650 MWe-class, Axially-heterogeneous MOX cores.	keff, RR, CRW, SVR, and SSW.	
	ZPPR-18A, 18B, 18C, 19A, 19B (5)	1,000 MWe-class, two-region homogeneous MOX cores with enriched-uranium regions.	keff, RR, CRW, and SVR.	
FCA (JAERI, Japan)	**FCA-XIIV-1** (1)	a 650 liter-sized core with MOX fuel in the core center region and enriched-uranium fuel in the periphery.	keff, RR, SVR, and DR(sample).	No.
	FCA-X-1 (1)	a 130 liter-sized one-region homogeneous core to simulate the JOYO Mk-II core with blanket.	keff.	
ZEBRA <MOZART Program> (Winfrith, UK)	**MZA** (1)	a 550 liter-sized one-region MOX core as a clean benchmark.	keff, RR, and SVR.	**Yes.**[12]
	MZB, MZC (2)	2,300 liter-sized two-region homogeneous MOX cores to simulate the prototype fast reactor MONJU.	keff, RR, CRW, and SVR.	
BFS (IPPE, Russia)	**BFS-58-I1** (1)	a 2,200 liter-sized uranium core with non-uranium Pu fueled region in the core center part.	keff, RR, and SVR.	No.
	BFS-62-1 -62-5, 66-1 (6)	3,400 liter-sized three or four-region enriched-uranium and/or MOX fuel cores with or without radial blankets.	keff, RR, CRW, SVR, SSW, and DR(sample).	
	BFS-67, 69, 66 (3)	10 kg of NpO_2 loading cores in central MOX region with weapon-grade Pu, high enriched Pu, and degraded Pu.	keff, RR, CRW, and SVR.	
MASURCA (CEA, France)	**ZONA-2B** (1)	a 380 liter-sized core in the CIRANO experiment series, which aimed at the study of plutonium burner cores	keff and ZMRR.	No.
SEFOR (General Electric, USA)	**SEFOR CORE-I, II** (2)	a 20MWt fast power reactor core fueled with mixed PuO2-UO2 and cooled with sodium.	DR(whole core).	**Yes.**[13]
Los Alamos (LANL, USA)	**FLATTOP-Pu, FLATTOP-25, JEZEBEL, JEZEBEL-Pu, GODIVA** (5)	sphere-shaped cores of approx. ten centimeter in diameter with metallic fuel consisted of Pu-239, or degraded Pu, or U-235.	keff.	**Yes.**[14]
JOYO (JNC, Japan)	**JOYO Mk-I** (1)	a 240 liter-sized fast power reactor core with mixed Pu and enriched-uranium fuel.	keff, CRW, SVR, Isothermal temperature reactivity, ZMRR, and Burnup reactivity.	**Yes.**[15]

*1) keff: Criticality, RR: Reaction rate, CRW: Control rod worth, SVR: Sodium void reactivity, SSW: Small sample worth, DR: Doppler reactivity, ZMRR: Zone material replacement reactivity

except for the sodium-void reactivity of uranium-fueled cores. From the sensitivity study, the extreme changes of C/Es for sodium void reactivity in BFS were found resulting from the considerable change in the resolved resonance parameters of U-235 from JENDL-3.2 to 3.3. We are now consulting with JNDC personnel about this problem.

Evaluation of Composition Changes of Fuel and MA by Reactor Core Burnup

To verify the burnup chain and related nuclear data, especially capture cross section of major actinides and MAs, JNC has launched the analysis of the post-irradiation data of the experimental fast reactor JOYO. One of the targets is the composition changes of core fuels. So far, all 70 specimens of the Mk-I core, where the burnup of fuel was up to five atomic %, were preliminarily evaluated by JENDL and three-dimensional diffusion calculation combined with a point burnup code ORIGEN2. The C/E values of burnup composition changes for nuclides such as U-235, U-236, U-238, Pu-239, and Pu-241 showed good consistency with JENDL in general [17].

The other target is the nuclear data of MAs. Two special assemblies loaded with MA isotopes such as Np-237, Am-241, Am-243, and Cm-244 were irradiated in the JOYO Mk-II core, the equivalent full power days of which was 251-276 total. A post-irradiation test of the samples has been conducted since 2003. Some results of preliminary analysis are shown in Table 5. It is suggested that the value of one important parameter for MA composition changes, the Am-241 isometric ratio, would be around 0.85. The detail of the JOYO MA irradiation analysis will be reported by another presentation [18] at this conference.

TABLE 3. Summary of C/E values for Fast Reactor Core Parameters by JENDL-3.2 and the Most Detailed Analytical Method of JNC

Core Parameter and Facility	C/E Value Range
Criticality	
: ZPPR, ZEBRA	**0.993-0.995**
: FCA, JOYO	0.994-0.996
: BFS	0.997-0.999
: Los Alamos	0.992-1.003
Reaction Rate Distribution	
: ZPPR(Pu-239 Fission, Core)	**0.99-1.01**
U-238 Capture/ Pu-239 Fission Ratio	
: ZPPR	**1.02-1.04**
: BFS	0.99-1.03
Control Rod Worth	
: ZPPR, ZEBRA, BFS	**0.95-1.05**
Sodium Void Worth	
: ZPPR, FCA, BFS	**0.90-1.05**
Doppler Reactivity	
: ZPPR, FCA, BFS, SEFOR	**0.95-1.05**
Burnup Reactivity	
: JOYO	**1.05**

TABLE 4. C/E Changes by nuclear data from JENDL-3.2 to JENDL-3.3.

Core Parameter and Experimental Core	C/E Change
Criticality	
: ZPPR-9	+0.1 %
: BFS-62-3A	-0.4%
Sodium Void Worth	
: ZPPR-9 (Step 3)	-2%
: BFS-62-3A (LEZ region)	**-27 %**

TABLE 5. C/E values of MA sample irradiation data in JOYO with the parameter of Am-241 isomeric ratio.

Nuclear Data Library	JENDL-3.2			JENDL-3.3
Am-241 Isomeric Ratio (Parameter)	0.80	**0.85**	**0.88**	**0.85**
Am-242m/Am-241				
(Irradiation position)				
- Above-core	1.67	1.28	**1.04**	1.30
- Core midplane	1.30	**0.99**	0.81	**1.07**

IMPROVEMENT OF PREDICTION ACCURACY AND APPLICATION TO FAST REACTOR DESIGN STUDY

One of the main objectives in studying nuclear data in a fission energy field would be to improve the design accuracy of power reactor cores for assuring safety, reliability, and economy. To develop future fast reactor concepts, the results of the nuclear data study are efficiently applied in Japan.

Performance of a Unified Cross-Section Set based on Bayesian Adjustment Technique

To utilize the past critical experimental data and power reactor operational experience to the reactor core design work, it is widely considered in the fast reactor field of Japan that the most powerful method is to adjust the cross sections based on the Bayesian theory with a least-square technique, where all related information including C/E values, experimental and analytical errors, sensitivity coefficients of various experimental cores and parameters, and cross-section covariance, is synthesized with physical consistency. In Japan, the adjusted cross-section set is called a "unified cross-section" set, which means a physical combination of the integral experimental information with differential nuclear data.

The performance of the cross-section adjustment is demonstrated in Fig. 4. It is indicated that the latest unified cross-section set based on JENDL-3.2 can predict well the criticality of various cores such as small or large, homogeneous or heterogeneous, Pu or uranium-fueled, critical experiment or power reactor, within 0.2%dk except several small cores.

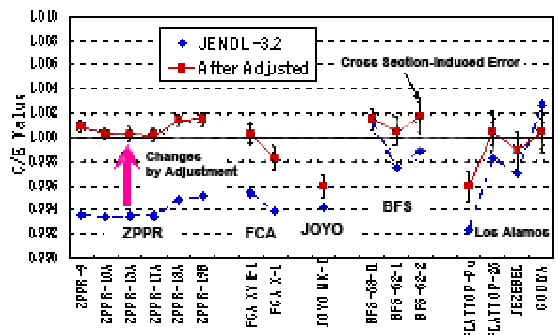

FIGURE 4. Adjusted results of JENDL-3.2 – criticality.

Evaluation and Improvement of Design Accuracy of FBR Cores in Japan

The evaluation method of design accuracy is based on the Takeda et al.'s comprehensive work [19]. By their derivation, the design error of a core parameter consists of three parts: (1) the error induced by nuclear data error, which can be calculated by multiplying the sensitivity coefficients and covariance of related isotopes in the matrix manner; (2) the error caused by the inevitable approximation of the exact transport equation such as cell heterogeneity modeling, neutron streaming treatment and discretization of energy, space and/or angle, etc.; and (3) the error from integral experiments when some mock-up experimental data are used as bias correction factors. From past experience, it is judged in Japan that the dominant component of the total design error tends to be the first one, the cross-section-induced error. The use of a unified cross-section set generated by the cross-section adjustment technique is considered most efficient to improve the design accuracy.

Table 6 is an example of the cross-section-induced error evaluation for a 1,500 MWe-class sodium-cooled MOX fast reactor core recently designed in the FBR feasibility study work of Japan. As found, the total errors of core parameters are markedly reduced after cross-section adjustment compared with original JENDL, and we can also clarify what isotopes contribute to the total error, which depends on the physical characteristics of core parameters.

TABLE 6. Accuracy of a Large Sodium-cooled MOX Core Design (Contribution from Cross-section error Before(B) and After(A) Adjustment, unit: % (1 sigma))

Error and Components	Criticality		Burnup Reactivity	
	B	A	B	A
Total Error	**0.82**	**0.26**	**5.2**	**3.1**
Pu-239 fission spectrum	**0.37**	0.01	0.1	0.1
Fe Inelastic	**0.38**	**0.11**	0.2	0.1
U-238 Capture	0.02	0.02	**3.1**	**1.4**
U-238 Inelastic	**0.29**	0.05	0.3	0.1
Pu-239 Capture	**0.27**	**0.18**	**1.6**	**1.5**
Pu-239 Fission	**0.36**	0.07	**2.0**	0.6
FPs of Pu-239 Capture	0.13	**0.13**	**1.9**	**1.9**
Beta of U-238	-	-	**1.9**	0.4
Beta of Pu-239	-	-	**1.4**	0.6
Others	0.30	0.12	1.4	1.3

CONCLUDING REMARKS

On the whole, it can be concluded that the performance of the current nuclear data and reactor analysis method has already reached a milestone in realizing the conventional sodium-cooled MOX-fueled FBR core with sufficient reliability. One request from users to nuclear data researchers would be to make the covariance data of important nuclides more consistent and accountable among major libraries in the world.

The next needs for the nuclear data study from fast reactor cores would be related to the development of innovative core concepts and burnup-related characteristics. First, some nuclides such as N-15, Pb/Bi, Si, or Ti, which are possible for use as alternative fuel and coolant, must be studied in the future. Second, the heavy actinides such as higher-mass Pu, Np, Am, and Cm need to be improved with their covariance data. Finally, the nuclear data of important fission products also have needs from various viewpoints: Nd for burnup measurement, Xe and Kr as the tag gas for failed fuel detection system, Ru, Rh, Pd, Nd, Sm, and Eu in low-decontamination fuel, and I-129 and Te-99 as long-lived fission products.

REFERENCES

1. Bondarenko, I. I., et al., “Group Constants for Nuclear Reactor Calculations,” Consultants Bureau (1964).
2. Takano, H., et al., “TIMS-1: A Processing Code for Production of Group Constants of Heavy Resonant Nuclei,” JAERI 1267 (1980).
3. Ronen, Y., *Handbook of Nuclear Reactors Calculations*, Vol. I (1986), pp. 102-128.
4. Hazama, T., Chiba, G., et al., “SLAROM-UF: Ultra Fine Group Cell Calculation Code for Fast Reactor,” JNC TN9520 2004-001 (2004), [in Japanese].
5. Nakagawa, T., et al., *J. Nucl. Sci. Technol.* **32**, 1259-1271 (1995).
6. MacFarlane, R.E. and Muir, D.W., The NJOY Nuclear Data Processing System Version 91, LA-12740-M (1994).
7. Shibata, K., et al., “JENDL-3.2 Covariance File,” *Proc. Int. Conf. on Nuclear Data for Science and Technology (ND2001)*, Vol. 1 (2002), pp. 40-43.
8. Chiba, G. et al., “Revision and Application of Covariance Data Processing Code, ERRORJ,” in *International Conference on Nuclear Data for Science and Technology 2004*, AIP Conference Proceedings (Melville, New York, 2005).
9. Dragt, J.B., et al., *Nucl. Sci. Eng.* **62**, 117-129 (1977).
10. Ishikawa, M., et al., “Development of a Unified Cross-section Set ADJ2000 based on Adjustment Technique for Fast Reactor Analysis,” *Proc. Int. Conf. on Nuclear Data for Science and Technology (ND2001)*, Vol. 2, (2002), pp. 1073-1076.
11. Ishikawa, M., “Consistency Evaluation of JUPITER Experiment and Analysis for Large FBR Cores,” *Proc. Int. Conf. on The Physics of Fuel Cycles and Advanced Nuclear Systems (PHYSOR2004)*, Session 5C (2004).
12. Kaise, Y., et al., “Investigation of MOZART Experimental Data and Analysis of MOZART Experiment using JFS-3-J3.2R Group Constant,” JNC TJ9400 2003-009 (2003), [in Japanese].
13. Hazama, T., “Re-Evaluation of SEFOR Doppler Experiments and Analyses with JNC and ERANOS Systems,” *Proc. Int. Conf. on Physics of Reactors (PHYSOR96)*, Vol.2, pp.E-36-45 (1996).
14. Collins, P.J., “An Adjusted Cross Section Library for DFBR,” PNC TN9410 97-034 (1997).
15. Yokoyama K., et al., “Analysis of the Experimental Fast Reactor JOYO Mk-I Start-up Test and Operation Data,” PNC TN9400 2000-036 (2000), [in Japanese].
16. Shibata, K., et al., *J. Nucl. Sci. Technol.,* **39**, 1125-1136 (1995).
17. Yokoyama, K., et al., “Analysis of Post-Irradiation Experiments on Core Fuel Composition of the Fast Experimental Reactor ‘JOYO’ MK-I,” JNC TN9400 2004-009 (2004), [in Japanese].
18. Ohki, S., “Validation of MA nuclear Data by Sample Irradiation Experiments with the Fast Reactor JOYO,” ND2004, *op. cit.*
19. Takeda, T., et al., *Nucl. Sci. and Eng.* **103**, 157-165 (1989).