Experimental Analysis Results on BN-600 Mock-up Core Characteristics at the BFS-2 Critical Facility

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ABSTRACT

Experimental analysis was performed on BFS-2 critical experiment (BFS-62-1, 2, and 3A cores), by both JNC and IPPE analysis systems. The objective of the analysis is to examine the accuracy of analysis systems on important nuclear parameters for the design of BN-600 hybrid core, which is presently discussed for one of the feasible ways to dispose the Russian surplus weapons plutonium. The analyzed items include the criticality, the spectral indices, the Sodium Void Reactivity Effect (SVRE), the Control Rod Worth (CRW) and the fission reaction rate distribution. Analysis accuracy obtained by both JNC and IPPE systems are described and compared.

1. INTRODUCTION

In order to effectively dispose Russian Surplus Weapons Plutonium, utilization of BN-600 (Russian commercial fast reactor) has been discussed as one of the practical options. Since the current BN-600 core consists of uranium dioxide fuel, verification of nuclear characteristics analysis accuracy on mixed-dioxide (MOX) fueled core is required prior to the installation of the MOX fuel into the core. In 1999, according to the agreement between JNC and IPPE, a series of critical experiments (BFS-62 cores) was initiated using the BFS-2 critical facility of IPPE[1]. Measurements have been finished in the first five configurations by the end of 2001, and the experimental analysis is in progress by both JNC and IPPE. These results are expected to be useful not only for supporting the fast reactor option in Russian Surplus Weapons Plutonium disposition program, but also for enhancing reliability of JNC's nuclear characteristics analysis system for fast reactor cores.
In this paper, experimental analysis results are reported mainly on the mockup assembly of BN-600 hybrid core (BFS-62-3A), as well as the first two uranium dioxide fueled cores (BFS-62-1 & 2).

2. EXPERIMENT

2.1 FEATURES OF BN-600 HYBRID CORE

The key features of the BN-600 hybrid core[2] arrangement are as following:

- There are four sub-zones in the core (in contrast to three zones in the current UO_2 fuelled core) having different fuel types and enrichment values;
- Stainless steel reflector is installed instead of uranium radial blanket.

Changes in the core arrangement would result in the following consequences that can be estimated through critical experiment and its analysis:

- Neutron balance change, which is mainly caused by plutonium introduction (having nuclear characteristics different from those of uranium) into the core and by the increase of neutron leakage from the core owing to the use of stainless steel reflector, which is more transparent but less absorbing as compared to the uranium blanket;
- The plutonium introduction results in changing of the sodium void reactivity effect (SVRE), which is one of the key items in BN-600 hybrid core safety substantiation;
- The transfer to the hybrid core has to be achieved by the currently operating control rod system. Under possible changes in BN-600 neutron field, it is necessary to analyze the changes in control rod worth with high enough accuracy.

Other changes would occur in some reduction of β_{eff} value and insignificant change of Doppler temperature reactivity effect (DTRE).

2.2 EXPERIMENTAL CONFIGURATIONS AND THE ANALYZED PARAMETERS

General specification and the relationships of BFS-62 cores, which will be discussed in this paper, are summarized in Table I.

Table I General Specification of Analyzed Cores and the Relationship

Core	BFS-62-1	BFS-62-2	BFS-62-3A
LEZ (U) (15% enrichment)	511FR* +24SR** +24CR***	511FR +24SR +24CR	595FR +54RWF**** +24SR +72CR
Equivalent Radius(cm)	63	63	73
MEZ (U) (18% enrichment)	348 FR +48CR	348 FR +48CR	222FR
δ Radius(cm)	20	20	10
PEZ (U, Pu) (17% enrichment)	-	-	336FR
δ Radius(cm)	-	-	14
HEZ (U) (21% enrichment)	499FR	480FR	237FR
δ Radius(cm)	19	18	8
Equivalent Radius of the Fuel Region (cm)	102	101	105
Radial Blanket (UO_2 depleted)	1840	1840 +8 rows SS +2 rows B_4C in 120°degree	1840 +8 rows SS +2 rows B_4C in 120°degree
Objective	Mockup of the current BN-600 core	Replacement of UO_2 blanket with SS reflectors, based on BFS-62-1 core	Mockup of the BN-600 hybrid core

Note ; All absorber rods are withdrawn in each core.
* FR ; Fuel Rod
** SR ; Safety Rods Mockup
*** CR ; Compensator Rods Mockup
**** RWF ; Rods Without Fuel (fuel pellets replaced by SS and Al)

All the configurations of BFS-62 critical experiment have been planned in order to obtain measurement data that are important to evaluate possible changes in nuclear characteristics caused by the planned transition of the BN-600 core. Main measurement items were the criticality, the sodium void reactivity effect (SVRE), the control rod worth (CRW), the fission rate distribution and the reaction rate ratios (spectral indices).
Detailed description of each configuration (from BFS-62-1 to BFS-62-5) and all the measurement procedures are shown in another paper, "BN-600 Hybrid Core Mock-up at BFS-2 Critical Facility", which is also presented in the present conference.
Considering the features of BN-600 hybrid core which was described in the last section, the SVRE and the effect at stainless steel reflector are most interesting. The effect of stainless steel reflector that replaced UO_2 blanket in the peripheral on the fission reaction rate distribution results in increasing the power heating in the radial reflector. The SVRE effect would become more positive according to the installation of the MOX fuel. Moreover, as it is one of the most important issues for BN-600 designers to verify "non-positive" SVRE in a partially MOX-fueled BN-600 core, the analysis accuracy obtained for SVRE in BFS-62-3A should play an important role in the design assessment. In order to measure the reactivity worth with enough accuracy, sodium pellets in both fuel region and the upper axial blanket region were voided in approximately one-sixth of the whole core in both BFS-62-2 and BFS-62-3A configurations.

3. ANALYSIS

3.1 THE GOAL OF ANALYSIS

The goal of analysis on critical experiments performed in BFS-62 cores is to determine the analysis accuracy of the neutronics parameters that are closely relating to the characteristics of BN-600 hybrid core, and to verify used analysis codes and nuclear data. If the Russian analysis systems that have supported stable operation of BN-600 for long years are verified for the accuracy, the transfer of BN-600 reactor to the hybrid core can be realized in conditions of minimal project resources. That determines the advanced requirements to the calculation accuracy. IPPE is responsible for the quality assurance of Russian analysis systems.
JNC has also developed an independent analysis system (nuclear data library, cell calculation code, core calculation code, and so on) for fast core analysis. If reasonably good accuracy is confirmed by both the systems, reliability of experimental analysis results on BFS-62 cores should be enhanced for future discussion on analysis uncertainty in BN-600 hybrid core.

3.2 THE JNC ANALYTICAL SCHEME

In the JNC's analysis[3], 70-energy-grouped nuclear constant set based on JENDL-3.2 (JFS-3-J3.2R)[4] was basically used for the standard scheme. The self-shielding effect was considered by the factor table interpolation method. One-dimensional (1-D) heterogeneous cell calculations were performed by adopting Tone method[5], in which the background cross section is evaluated by collision probability density calculation. Whole core calculation was made by the CITATION-FBR code[6] (based on diffusion theory) with 3-D Hex-Z model, adopting the anisotropic diffusion coefficient. Transport and mesh-size correction were made by using the MINIHEX[7] code (3-D, Hex-Z geometry), or the TWOTRAN-II code[8](2-D R-Z geometry).
In order to obtain more exact evaluation, additional analyses were made by adopting a newly developed nuclear constant set[9]. It aims at more accurate treatment of the resonance interaction, and consists of the basic and the fine parts. The basic part was produced by the NJOY code system[10], having infinite dilution cross sections and self-shielding factors. The VITAMIN-J 175-group

structure[11] was temporarily used in the present analysis. The fine grouped part was prepared by the TIMS-1 code system[12], and cross sections were produced for the energy range below 40.8keV with 9,150 energy group structure by a constant lethargy. This new constant set is called UF (Ultra fine)-175 in this paper.

3.3 THE IPPE ANALYTICAL SCHEME

In Russia, the nuclear calculations of BN-600 are based on three-dimensional (Hex-Z geometry) diffusion codes, TRIGEX[13] and JAR[14] using ABBN-93[15] nuclear data. The code TRIGEX in complex with a cell code FFCP[16] allowing to consider heterogeneous structure of BFS assemblies was used in analysis of experiments. The FFCP code is based on the first flight collision probability method taking into account the resonance effects based on the sub-group approach. Besides the heterogeneity correction, other corrections were evaluated in TRIGEX analysis as following. The mesh and the group corrections were estimated by a comparison with calculations with a smaller spatial and energy mesh. The transport correction was estimated using the TWODANT code[17] in R-Z geometry. Comparison of calculated and experimental data in these conditions allows to evaluate the calculation prediction accuracy of the simulated parameters including the precision of the establishment of corrections.
To evaluate the constant's uncertainty and also the precision of the establishment of corrections, the Monte-Carlo method was applied by using the MMK-KENO code[18] with an exact 3-D reactor model ("as-built model").
All calculations were made by using 299-group constant system ABBN-93 with CONSYST code for preparing nuclear constants. Precise calculations with the MMK-KENO code as well as the calculations with the TWODANT code were performed directly in 299-group approximation. For calculations by TRIGEX and FFCP codes, 299-group constants were reduced to 26 or 18 groups.

3.4 COMPARISON OF THE ANALYTICAL SCHEMES

Comparison of both institutes' analytical schemes is summarized in Table II.

Table II Comparison of Analytical Schemes between IPPE and JNC

Items	JNC	IPPE
Nuclear data library	JENDL-3.2	FOND-2.2[19]
Nuclear constant set	JFS-3-J3.2R	ABBN-93
Number of energy groups	70	299, then collapsed into 26
Heterogeneity & resonance treatment in cell calculation	Tone method f-table interpolation method	Subgroup method
Diffusion calculation (Base result)	Finite difference method in 3D-Hex-Z modeling with anisotropic D	Corrected course mesh method in 3D-Hex-Z modeling with anisotropic D
Used code for transport correction	MINIHEX(3D-Hex-Z model) TWOTRAN-II(2D-RZ model)	TWODANT (2D-RZ model)
Analysis method adopted for further detailed investigation	Newly developed nuclear constant with ultra fine structure (UF-175)	Multi-grouped Monte Calro (MMK-KENO code)

It is evident that both the schemes have similar functions, however, they are completely independent each other from the viewpoints of both nuclear data and analysis method. Accordingly, reliability of analysis results on BFS-62 cores can be enhanced when both the analysis schemes show reasonable agreement with the measurement values.

4. ANALYSIS RESULTS BY BOTH STANDARD METHODS

4.1 CRITICALITY

Criticality estimation results are summarized in Table III. JNC's C/E (Calculation to Experiment) values show underestimation from 0.3 to 0.5%, and the deviation among the three cores is around 0.1%. The observed underestimation is relatively smaller than those obtained in fully Pu-U-fuelled critical assemblies (0.6 to 0.8% for ZPPR-9, 13A and 17A)[20]. IPPE's C/E values are higher than JNC's ones and show better agreement with the measurement. The difference of C/E values among the three cores is around 0.1%, only. The obtained results show that the applied method and calculation codes provide quite good agreement of k-eff calculation accuracy. In the IPPE's scheme, the agreement of calculation and experiment is observed within 0.3%Δk/k.
More detailed investigation devoted for checking the consistency of those C/Es on BFS-62 cores is described in the next chapter (**5.1**).

Table III Summary of Experimental Analysis Results on the Criticality (C/E Values)

[Diffusion calculation k_{eff} results corrected by transport and mesh effects]

Analysis Method	BFS-62-1	BFS-62-2	BFS-62-3A
Deterministic Method of JNC*1	0.9953	0.9965	0.9953
Deterministic Method of IPPE*2	0.9978	0.9986	0.9977
Monte-Carlo Method of IPPE*3	0.9967*4	0.9985*4	1.0011*4

*1 Based on the CITATION-FBR code; 70 energy groups.
*2 Based on the TRIGEX code; 26 energy groups.
*3 Based on MMK-KENO code with "as-built" modeling; 299 energy groups.
*4 Statistic error is 0.0004.

4.2 SPECTRAL INDICES

Two types of spectral indices have been measured by processing fission reaction rates that were measured at the center of each core. The defined indices are the fission rate ratio between U238 and U235 (F8/F5) and another one between Pu239 and U235 (F9/F5).
Base calculation was performed by whole core diffusion code by both JNC and IPPE standard schemes. Correction factor was applied in order to take into account the reaction rates' distribution in the center cell and the actual detector position. The transport effect on indices was found to be negligible.
For F9/F5 index, the influence of heterogeneous structure of BFS assemblies is not much (about 0.4%). Heterogeneity influence on F8/F5 index is more essential (see Fig. 1), and the heterogeneity correction is about 4%.
The analysis results are shown in Table IV. All the results show reasonable agreement with the measurement values.

Table IV Experimental Analysis Results on the Spectral Indices (C/E Values)

[diffusion calculation results corrected on the cell effect]

	F8/F5			F9/F5		
	JNC	IPPE	Uncertainty* (%)	JNC	IPPE	Uncertainty* (%)
BFS-62-1	0.996	1.018	±1.8	0.992	1.011	±2.1
BFS-62-2	0.970	0.991	±1.8	0.995	1.013	±2.1
BFS-62-3	1.019	1.025	±2.0	0.992	0.997	±1.6

* Uncertainty ; measurement error (1 sigma value)

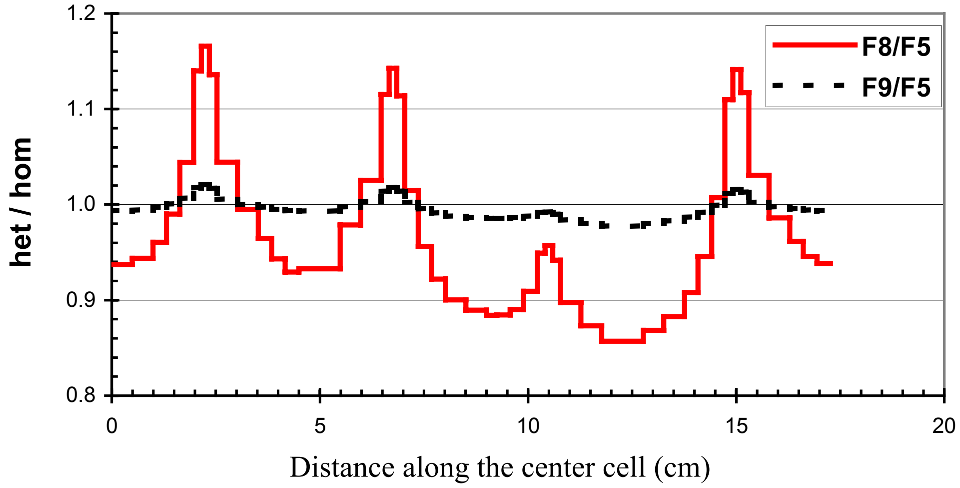

Fig. 1 Distribution of Fission Cross-Sections Ratios Along Inner Core Cell

4.3 SODIUM VOID REACTIVITY EFFECT (SVRE)

Base calculations for BFS-62-2 and BFS-62-3A were performed in both institutes by using each own diffusion code with 3-D Hex-Z model. Correction for the transport and mesh effect was evaluated with RZ model by 2-D TWOTRAN-II code in JNC's scheme, and by 2-D TWODANT code in IPPE's scheme.

The comparison of calculated and experimental results as for cumulative SVRE for all the fuel regions is given in Table V, with a unit of cent. For each critical condition, a specific calculation value of β_{eff} was used: for BFS-62-2 β_{eff}=0.00716(JNC), 0.00726(IPPE); for BFS-62-3A β_{eff}=0.00619(JNC), 0.00617(IPPE).

Considerable underestimation for the cumulative SVRE (in absolute value) was confirmed in both cores, by both the standard schemes. The inconsistency lies in a conservative side, however, the deviation reached several times as large as the measurement error (one sigma). Considering the total weight of the voided sodium (164kg in BFS-62-2, and 179kg in BFS-62-3A) and the β_{eff} values, it is estimated that the accuracy of SVRE analysis is within 0.2pcm/kg in traditional configuration of BN-600 by IPPE's scheme.

A comparison of region-wise SVRE analysis results on BFS-62-3A is illustrated in Fig. 2. As for JNC's results, absolute values of SVRE were smaller than the experimental values in most of the fuel regions. On the other hand, an increase of the difference between the calculated and experimental values was observed from the core center to the outward in IPPE's results, which is, however, rather small. Effects of nuclear constants on the SVRE analysis in JNC's scheme are described in the next chapter (**5.2**).

Table V Summary of Experimental Analysis Results on the Cumulative SVRE
[Diffusion calculation results corrected by transport and mesh effects]

Assembly	Amount of Voided Tubes	JNC		IPPE		Uncertainty***
		C*	C-E**	C	C-E	
BFS-62-2	238 (in a sector)	-24.7	+4.6	-25.2	+4.1	±0.7
BFS-62-3A	259 (in a sector)	-22.9	+5.5	-25.8	+2.6	±0.8

*C ; Calculated value, **E ; Experimental value, *** Measurement error (1 sigma value) [Unit in cent]

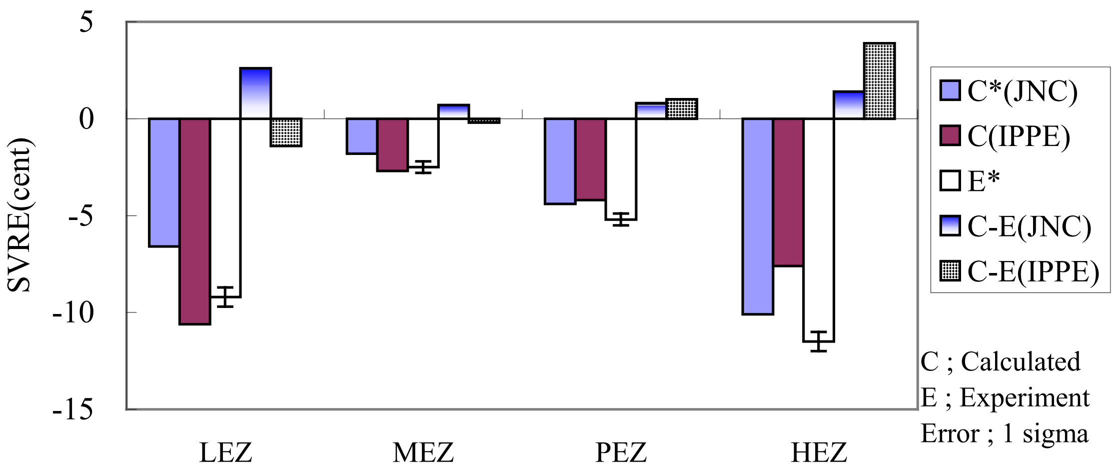

Fig. 2 Region-wise Comparison of SVRE analysis results on BFS-62-3A

4.4 CONTROL ROD WORTH (CRW)

All the measurement results in 3 assemblies are plotted in Fig. 3. Effect of Pu fuel installation is found in increase of the 3rd ring CRW and decrease of all the other (located in inner regions) CRWs, from the comparison between BFS-62-3A and BFS-62-2. The effect on CRW ranges from -15 to +10 cents. Effect of the replacement of UO_2 blanket by SS reflector was found much smaller (within 2 cents), from the comparison between BFS-62-2 and BFS-62-1.

The comparison results of calculation and experimental data for the CRW are given in Fig. 4. The CRWs were obtained by direct calculation, and transport and mesh-size correction were done in both the schemes. Transport correction factor ranged from 0.95 to 0.97. Calculation value of β_{eff} for BFS-62-1 is equal to the BFS-62-2 one: 0.00716(JNC), and 0.00726(IPPE).

In the whole, the discrepancies between calculation and experiment did not exceed 6%, which can be assumed as a reasonable evaluation of the calculation accuracy. A compensation between the transport correction and the mesh correction was observed in both the schemes and the combined correction factor lies only in a few %.

Effects of more detailed nuclear constants (UF-175) on JNC's results were found mainly for inner regions of the core, but less than 2 %.

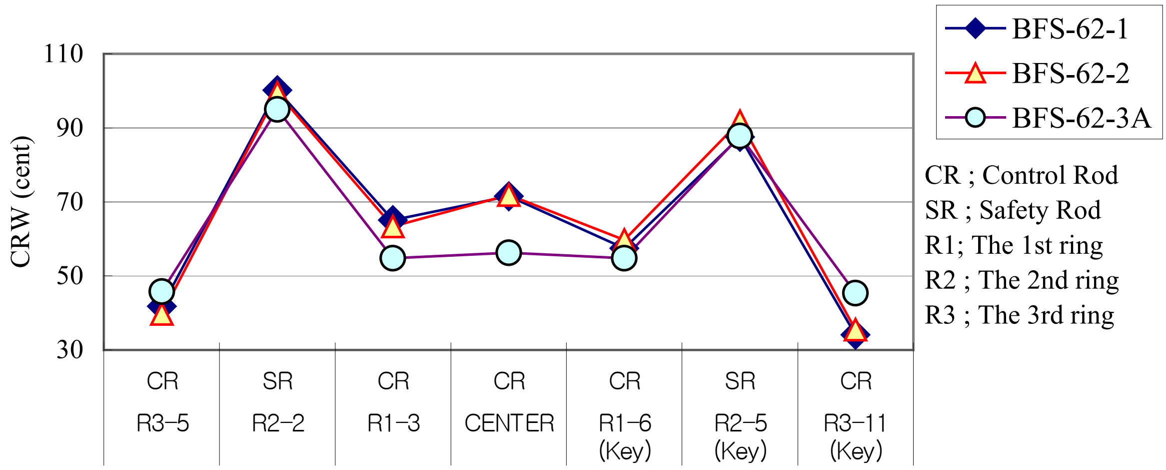

Fig. 3 Experimental values of CRW in BFS-62 cores

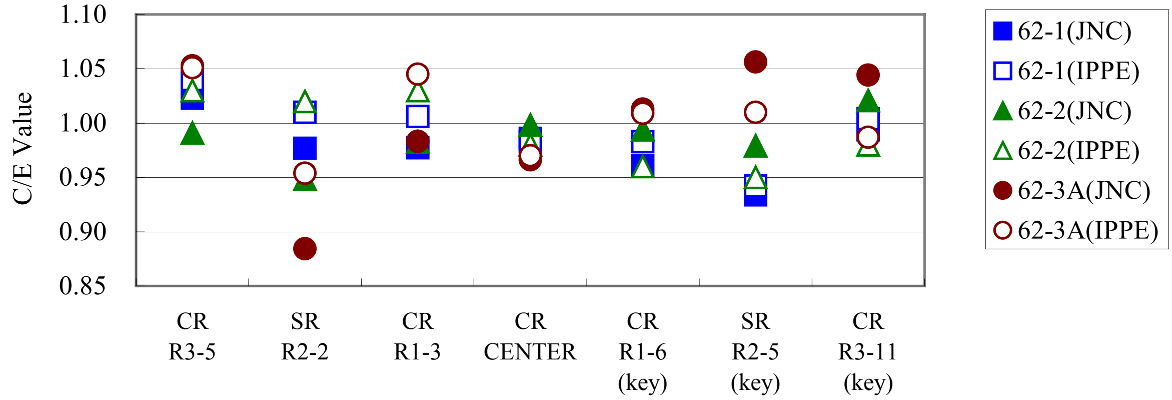

Fig. 4 Comparison of C/E Values of CRW in BFS-62 cores

The effect of heterogeneous structure on CRW was not mocked up in the BFS critical assemblies. IPPE's evaluation of heterogeneity correction for BN-600 control rod worth gives the value from 5-6% (for compensator rods) to 15% (for safety rods). The error of its determination is estimated as ±2-3%, which can not increase much overall discrepancy of control rod worth calculation.

4.5 FISSION REACTION RATE DISTRIBUTION

Three types of radial distributions of fission reaction rates (Pu-239, U-235, U-238) were analyzed and compared with the experimental values. C/E values for U-235 fission rate distribution obtained by IPPE in BFS-62-1, BFS-62-2 and BFS-62-3A are compared in Fig. 5. Maximal disagreements between the calculation and the experiment for U-235 fission rate in the fuel region were about 4%, in the uranium blanket can reach 8-12%. In the steel reflector, the difference reached 20-30%. C/E values for Pu-239 fission rate distribution obtained in the three assemblies showed the same pattern. On the other hand, as for U-238 fission rate distribution, the scale of maximum disagreements was about 10% in the core, and about 15% in the peripheral region, as shown in Fig. 6. JNC's results showed the similar tendency for both the items, as well.

To present an overall view of analysis accuracy, statistic amounts of discrepancies on radial fission rate distribution are summarized in Table VI. As for the fuel region, mean values and the standard deviation values for each data show that the analysis accuracy was almost same level as the

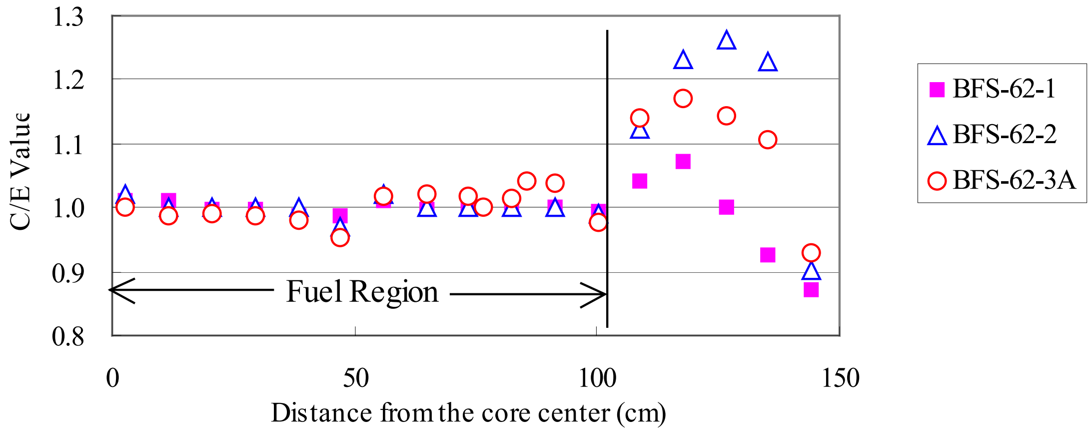

Fig. 5 C/E values for U-235 Fission Rate distribution

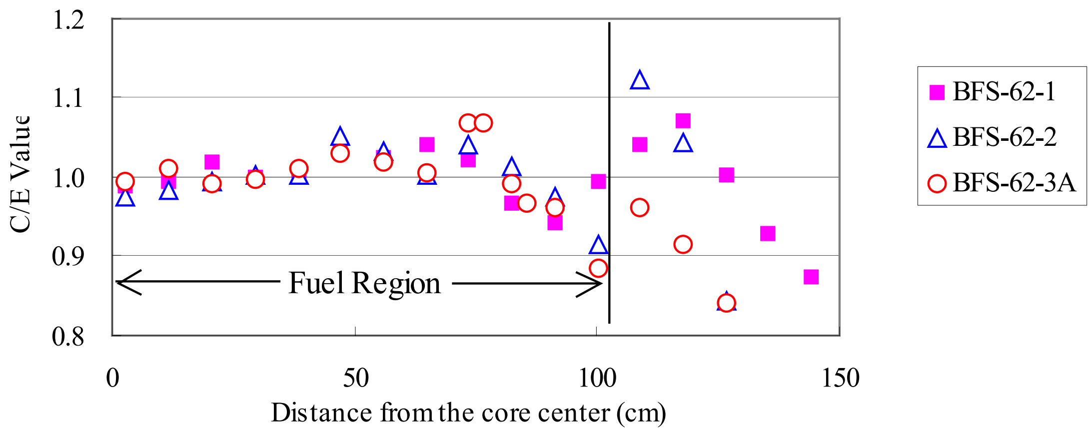

Fig. 6 C/E values for U-238 Fission Rate distribution (Evaluated by IPPE's sheme)

measurement uncertainty. Especially, it should be noted that IPPE's scheme showed excellent agreement with the measurement. On the other hand, as for the peripheral region, significant overestimation was observed for Pu-239 and U-235 fission rate distribution in both BFS-62-2 and BFS-62-3A assembly. It suggests that both institutes' analysis schemes tend to overestimate the fission rate distribution of Pu-239 and U-235 in the SS reflector region, and the given problem requires further investigations.

Clarification of the problem is quite urgent for IPPE from the viewpoint of analysis accuracy estimation of the neutron irradiation of irremovable reactor constructions, neutron fluxes on control devices, etc. Neither the adoption of UF-175, nor continuous-energy Monte Carlo method with JENDL-3.2 could solve the problem in JNC's scheme, so further examination using newly released JENDL-3.3[21], is intended in the near future.

At the same time, from the viewpoint of calculation of the power distribution, the noted discrepancies are not so essential because the heat power distribution in the steel reflector is determined mainly by the gamma rays. Neither TRIGEX, nor JAR codes are intended for performing such calculations and will not be verified for this purpose.

Table VI Mean Value and Root Mean Square of Discrepancies for Radial Fission Rate Distribution in BFS-62 cores

Assembly	Fission Nuclide	**Fuel Region**(11 or 13 data) JNC Mean* ± S.D.**	**Fuel Region** IPPE Mean ± S.D.	**Peripheral Region**(4 data) JNC Mean ± S.D.	**Peripheral Region** IPPE Mean ± S.D.
BFS-62-1 [U Fuel] [UO_2 Blanket]	Pu-239	0.8 ± 1.6	0.2 ± 1.4	6.4 ± 8.2	3.3 ± 5.9
	U-235	-1.4 ± 1.1	-0.1 ± 0.8	5.5 ± 9.0	1.1 ± 6.2
	U-238	4.2 ± 5.3	-0.2 ± 3.8	4.3 ± 5.1	-0.7 ± 7.2
BFS-62-2 [U Fuel] [SS Reflector]	Pu-239	2.5 ± 3.3	-0.1 ± 1.5	34.1 ± 7.7	27.9 ± 8.1
	U-235	0.5 ± 2.2	-0.2 ± 1.2	26.1 ± 5.7	21.2 ± 6.2
	U-238	7.4 ± 6.0	0.1 ± 3.7	-5.6 ± 12.9	-8.9 ± 21.9
BFS-62-3A [Hybrid core] [SS Reflector]	Pu-239	2.2 ± 3.6	0.3 ± 2.8	32.0 ± 11.0	17.2 ± 3.5
	U-235	2.3 ± 3.9	0.2 ± 2.6	25.7 ± 8.2	13.9 ± 2.6
	U-238	1.9 ± 5.4	0.1 ± 4.7	-7.6 ± 14.5	-13.5 ± 9.4

* Mean ; average value of discrepancy (C/E value - 1.0) on all the analyzed points.
** S. D. ; standard deviation value of discrepancy (= C/E value - 1.0) on all the analyzed points.
Note ; Measurement uncertainties for U-235 and Pu-239 fission rate distributions are about 1.5-2 % in the fuel region and they increase up to 3-4 % in the peripheral region. For U-238, the uncertainties are about 2-3 % in the fuel region and 5-7 % in the peripheral region.

5. DISCUSSION OF THE RESULTS

5.1 CRITICALITY ANALYSIS ON BFS-62-3A CORE

Breakdown of criticality analysis on BFS-62-3A core is summarized in Table VII.

Table VII Comparison of k-effective Values on BFS-62-3A Core by Deterministic Method

Items	JNC's standard scheme	IPPE's standard scheme	Difference (%dk/kk') (IPPE-JNC)
Diffusion (Homo. model)	0.9838	0.9832	-0.06
Diffusion (Hetero. model)	0.9924	0.9915	-0.09
Transport correction	+0.0035	+0.0043	+0.03
Mesh-size correction		-0.0005	
Boundary effect	-	+0.0026	+0.26
After all corrections	0.9959	0.9977	+0.18

It is clear that both diffusion results with homogeneous model and heterogeneous model show similar values, and also the transport and the mesh size correction values are almost same in both institutes' results. The value of heterogeneity corrections for BFS assemblies comes to +0.6 - 0.8%Δk/k, the transport correction comes to +0.3 - 0.4%Δk/k, the rest corrections are less essentially. Propriety of those correction values were confirmed by precise calculations by Monte-Carlo method using MMK-KENO code in IPPE scheme.

On the other hand, Nuclear Data Library (NDL) effect between JFS-3-J3.2 and ABBN-93 was investigated[22] by using the JNC's standard scheme. The NDL effect on criticality of BFS-62-3A core was calculated as +0.31% ; (ABBN-93)-(JFS-3-J3.2). Influence of nuclear constant change from JFS-3-J3.2 to JFS-3-J3.2R, in which the weighting neutron flux spectrum had been corrected, was negligible on the criticality evaluation. In both BFS-62-1 and BFS-62-2 cores, the NDL effects were estimated as approximately +0.1%, which can explain about half of the difference between JNC and IPPE criticality C/E values (see Table 3). Furthermore, the difference between deterministic method results of IPPE and the Monte-Carlo calculation results showed increase only in BFS-62-3A estimation. Considering this information, it is possible that IPPE's criticality result on BFS-62-3A may be increased by about +0.3%.

Another note should be made concerning the precise taking into account of the resonance cross section self-shielding in the steel reflector region. In IPPE's scheme, the total cross section self-shielding in the reflector region was calculated using neutron flux density as a weight function instead of the current as it assumed in usual scheme of nuclear constants preparation. The substantiation of such approach was given by M.N. Nikolaev, A.M. Tsiboulia et al.[23] In particular, for BN-600 hybrid core the usage of resonance self-shielding factors, calculated with a current weight, leads to k-eff value reduction about 1%Δk/k. Such so-called "boundary effect" was not regularly considered in JNC's scheme, however, in the present study, the effect was confirmed approximately +0.2%, consistent with IPPE's evaluation.

In addition, JNC's more detailed analysis using UF-175 showed approximately +0.2% increase of k-effective and the final C/E value became 0.998 (without applying the boundary effect).

5.2 SODIUM VOID REACTIVITY EFFECT (SVRE) IN BFS-62-3A

Effects of nuclear constants on SVRE were investigated by using UF-175 in the JNC's scheme, and the results are shown in Fig. 7. The underestimated absolute values of SVRE were increased mainly in the inner regions of the core by using UF-175. Consequently, the C/E value on the total SVRE for all the voided fuel regions was increased from 0.81 to 0.93.

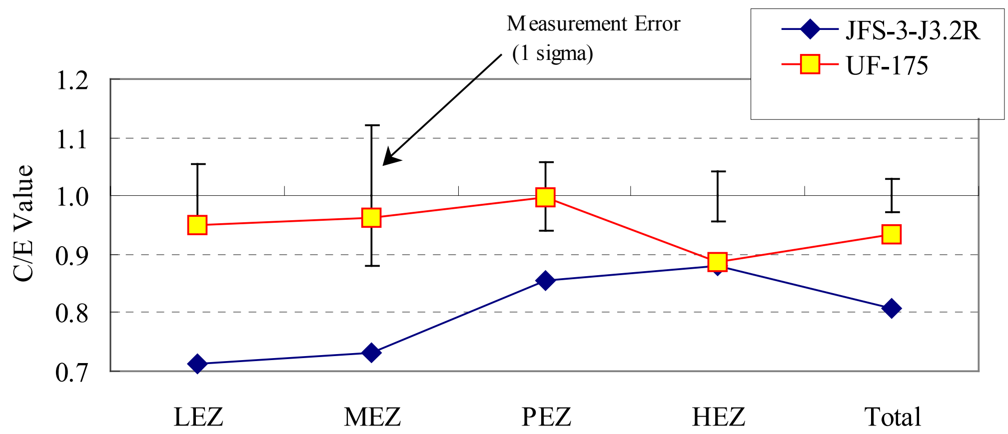

Fig. 7 Effects of Nuclear Constants on SVRE on BFS-62-3A

Breakdown of the calculated SVRE into the non-leakage term and the leakage term, which were obtained by exact perturbation theory based on the transport calculation results, is shown in Fig. 8. Effects of the UF-175 adoption appeared clearly as the decrease of non-leakage term values in all the fuel regions, while the leakage terms are hardly changed. Since the contribution of the non-leakage term to the total SVRE is larger in the inner core region, the effect of the UF-175 adoption appeared clearly in the inner regions of the core.

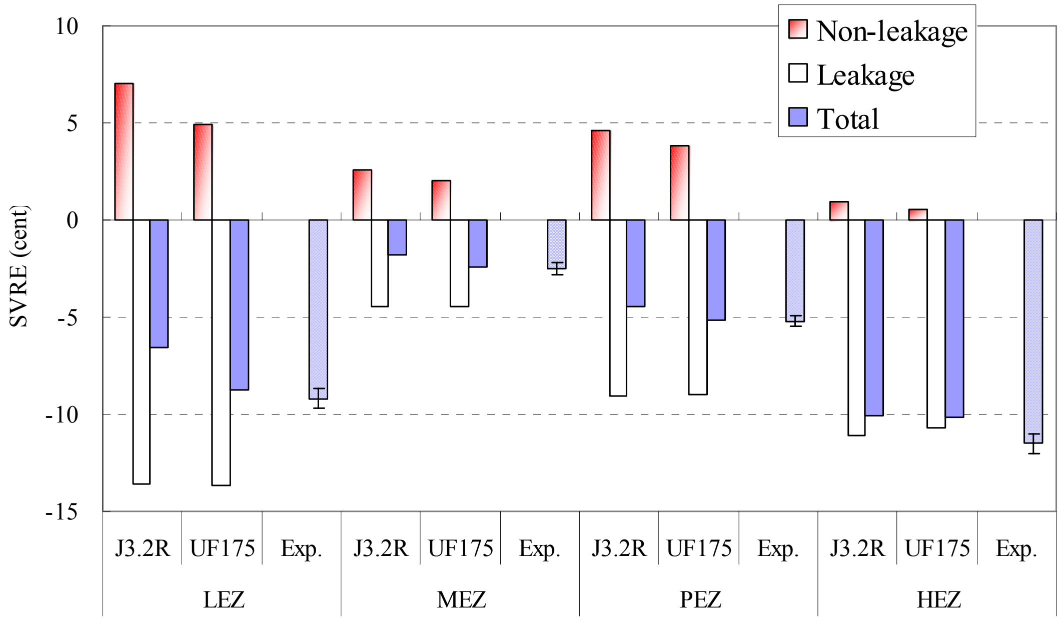

Fig. 8 Effects of Nuclear Constants on Region-wise SVRE Analysis on BFS-62-3A (JNC's scheme)

Though the SVRE measured in every fuel region is negative on BFS-62-3A, it is noted that the SVRE is the balance between the positive non-leakage term and the negative leakage term. Since the absolute value of the SVRE is quite small, analysis accuracy on both the non-leakage term and the leakage term should be carefully evaluated in the design of the BN-600 hybrid core. There exist slight differences between BFS-62-3A core and the designed BN-600 hybrid core. For example, higher density of the radial reflector in the BN-600 hybrid core comparing with that in BFS-62-3A tends to decrease the negative leakage term. Existence of plutonium, which should be increased by fuel burn-up in all the UO_2 fuel regions of the BN-600 hybrid core, tends to increase the non-leakage term. Those factors tend to increase the total SVRE of the BN-600 hybrid core. Accordingly, analysis accuracy obtained in the present study for the SVRE in BFS-62 cores can play an important role in the future discussion on analysis uncertainty of SVRE in the BN-600 hybrid core.

CONCLUSIONS

Through experimental analyses on BFS-62 cores, which properly simulated the current and the planned BN-600 cores, analysis accuracy for important nuclear parameters were examined by both JNC and IPPE schemes. Good agreement between the calculated and measurement values was confirmed on most parameters, including criticality, spectral indices, CRW and fission rate distribution in the fuel regions.

It is noted that both JNC and IPPE schemes showed certain underestimation of SVRE (positive side) in both BFS-62-2 and 62-3A cores. The inconsistency was successfully improved in JNC’s scheme by adopting a newly developed nuclear constant set with a fine group structure.

Furthermore, it should be also noted that considerable inconsistency between the calculated and measurement values was found for the fission rate distribution in the stainless steel reflector region, outside the fuel region.

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