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技術報告

Transport Criticality Analysis for FBR MONJU Initial Critical Core in Whole Core Simulation by NSHEX and GMVP

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技術報告

「もんじゅ」初臨界炉心臨界性の輸送計算解析
－３次元ノード法Sn輸送計算コードNSHEX及びモンテカルロ輸送計算コードGMVPによる－

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FBR MONJU Initial Critical Core (ICC) criticality problem has been solved by deterministic and Monte Carlo transport methods by the codes NSHEX and GMVP. The analysis has been carried out in different energy-groups approximations. As a result the effect of cross-section (XS) condensation from 70 into few energy-group structures by different collapsing methods has been evaluated. The 3D discrete-ordinate code NSHEX has been applied for wide range of core simulations-from whole core, considering the fissile, fertile and shielding regions to simplified models that simulate an increased neutron leakage. It has been found that there is room for improvement in the assessment of the neutron leakage in the few energy-group approximations. The good agreement between NSHEX and GMVP results, especially without XS collapsing, is pointed out as a conformation for the applicability of the code NSHEX in FBR 3D whole core calculations. Some practical conclusions have been extracted that are important for the implementation of the code NSHEX in the standard criticality analysis.

「もんじゅ」初臨界炉心の臨界性を，拡散近似でなく本来の輸送計算手法で解析した。決定論的手法として３次元ノード法Sn輸送計算コードNSHEX，また確率論的手法としてモンテカルロ輸送計算コードGMVPを用いた。中性子エネルギー離散化近似としては，70群を基本に18群，７群の少数群近似についてもサーベイした。その結果，通常の拡散計算とは異なり，輸送計算では無視し得ないエネルギー群依存性のあることを明らかにした。特にNSHEXコードによる少数群の解析では，群定数縮約方法に検討の余地のあることを明確化した。また，このエネルギー群依存性は，炉心モデルを簡素化した中性子漏えいの大きなケースで増加傾向となることも明らかにした。したがって，少数群による解析では，この点に十分な配慮が必要である。ただし，群縮約近似をしない70群の計算では，NSHEXの結果はGMVPと良く一致し，同コードの「もんじゅ」炉心への適用性を確認した。

キーワード

FBR, Transport Calculations, Transport Effect, NSHEX, GMVP, Energy-group Collapsing, MONJU

FBR，輸送計算，輸送効果補正，NSHEX，GMVP，エネルギー群縮約，「もんじゅ」

1. Introduction

Japanese FBR MONJU initial criticality was achieved in April 1994. The major reactor core characteristics were determined and analyzed by criticality experiments and physics analysis [1, 2] by JUPITER analysis system [3]. Recently the advanced core analysis system MEISTER has been developed [4] and utilized for FBR MONJU core

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physics test analysis. But the MEISTER system has been established based on the conventional diffusion approximation calculation. The transport corrections for core characteristics are required to be determined by the application of exact three-dimensional (3D) hexagonal transport code for accurate physics analysis. The transport code NSHEX[5] has been developed especially for FBR core physics calculations. NSHEX is a hexagonal geometry 3D transport code that solves the neutron transport problem by a discrete-ordinate nodal method. The anisotropic scattering effect is simulated by the extended P1 approximation. The neutron transport equation is solved using a nodal scheme with one mesh cell (node) per hexagonal assembly in plane. The node-internal spatial neutron flux distribution and transverse leakage distribution are simulated by second order polynomial-series approximation. Advanced methods for accurate description of radial and axial neutron leakage are incorporated into the code[6,7]. These methods have been verified by NEA/CRP 3D Neutron Transport Benchmark Model and by the large assembly-size KNK-II Model[8].

It is well known that the transport methods in core-physics analysis demand huge computational time and computer processing memory. The application of these methods is therefore limited and they are applied mainly to the evaluation of the transport effect corrections for the diffusion results. Such kind of analysis is sometimes performed for problems of smaller size, assuming inter-independency of the transport effect correction from the energy-group approximation or from the core modeling. This independency assumption however, should be precisely verified by transport analysis. As the progress in the computer technology reduces some of the limits in the application of the transport analysis, wider-ranging core problems became possible to be solved in reasonable time and memory size limit. As the size of the problem depends on the core model, energy-group approximation and the method-associated approximation, the 3D transport code NSHEX has been utilized mainly by low-order discrete-ordinate approximation (S_4) and up to 18 energy-group approximation in partial-core models for MONJU criticality analysis. This work presents the results from discrete-ordinate analysis by the code NSHEX for wider range of 3D core models-from very simplified models to the whole core models, that allows evaluation of the transport effect, associated with the core modeling. The extension of the energy-group approximations from 7- and 18- to 70-group allows the estimation of the energy-group approximation effect from different core models. As a result, the influence of the cross-section (XS) collapsing method has been found to be significantly larger than that based on the diffusion theory. For example, the effect of energy-group collapsing from 70 to 18 groups had been evaluated by the diffusion code system MEISTER to be at the order of 0.01%[9]. The transport results however show the effect to be more than several times larger, hence one of the major issue discussed in the work is the influence of the different XS preparation methods on the energy-group approximation effect in the transport analysis.

The transport deterministic analysis has been carried out by the code NSHEX and has been verified by the Monte Carlo probabilistic analysis code GMVP[10]. The good agreement between NSHEX and GMVP results has been pointed out as a confirmation for the applicability of the code NSHEX to the MONJU whole core analysis, despite of some difficulties, mainly associated with the convergency of the neutron flux in the outer iterations.

2. Methods

2.1 MONJU Initial Critical Core (ICC) numerical models

MONJU ICC (Fig. 1A and B) consists of two plutonium enrichment zones (inner and outer core) of different Pu/(Pu+U) weight ratio: about 20% in the inner core and 28 % in the outer core. The core is surrounded by radial and axial (upper and lower) blankets, which contain depleted UO_2 of low enrichment. The fissile and fertile fuel zones

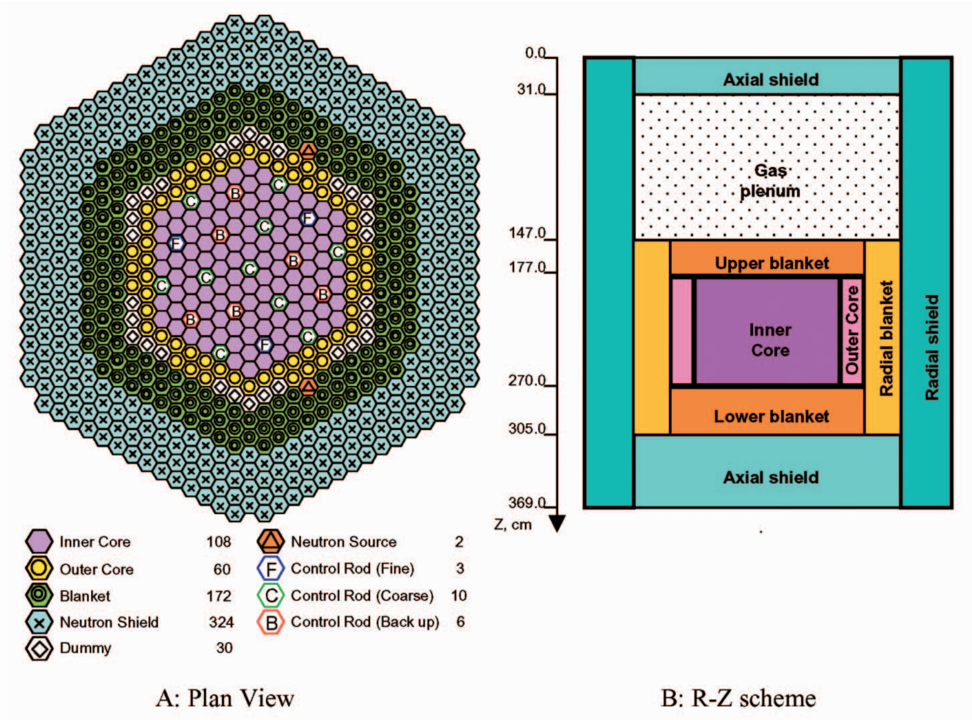

Fig. 1 Layout of FBR“ MONJU ”Initial Critical Core

are surrounded by axial and radial shields for reduction of the neutron fluence on the reactor vessel and the core internal structures. The major feature of MONJU ICC is that 30 sub-assemblies at the edge of the outer core are replaced by dummy-fuels. The core consists of 168 fuel sub-assemblies (108 in the inner core, 60 in the outer core), 30 dummy sub-assemblies in the outer core, 172 radial blanket sub-assemblies, 19 control rod sub-assemblies, 2 neutron sources and 324 shielding sub-assemblies. The hexagonal sub-assemblies are arranged in a grid of 11.56 cm inter-assembly pitch.

MONJU ICC criticality problem has been solved assuming control rods completely withdrawn and 200 °C isothermal conditions. The whole core numerical models (considering the core, blankets and shields) are based on 18 material compositions.

(1) The 3D ICC whole core model ICCB (Basic) is shown in Fig. 1A, B. It consists of 19 basic axial regions, subdivided into 56 axial layers (40,040 nodes). The model has been used for the GMVP calculations as well as for evaluation of the repreventativity of the other core models that have been simulated by the code NSHEX.

(2) The 3D ICC whole core model ICCR (Reduced), is based on the model ICCB, where the gas plenum region between the upper axial blanket and the upper shield has been removed (Fig. 2B). Additionally the radial shield is simplified by removing three of the four radial shield layers (Fig. 2A). Despite the reduction of 17 axial layers and 34.4% of the sub-assemblies, the model ICCR provides accurate 3D description of MONJU ICC. The effect of the reduction of the gas plenum on k_{eff} has been evaluated by 7 and 18 energy-group analysis by up to S_6 order approximation and it has been estimated to be less than 0.014% Δ *k*. The influence of the partial reduction of the radial shield has been evaluated by 7 energy-group calculations by up to S10 approximation and it has been found out to be so small that it was negligible-less then 0.004% Δ *k*. The total number of the nodes in the model ICCR is 54% less than in the model ICCB, which allows calculations to be performed in practically acceptable time-approximately 20

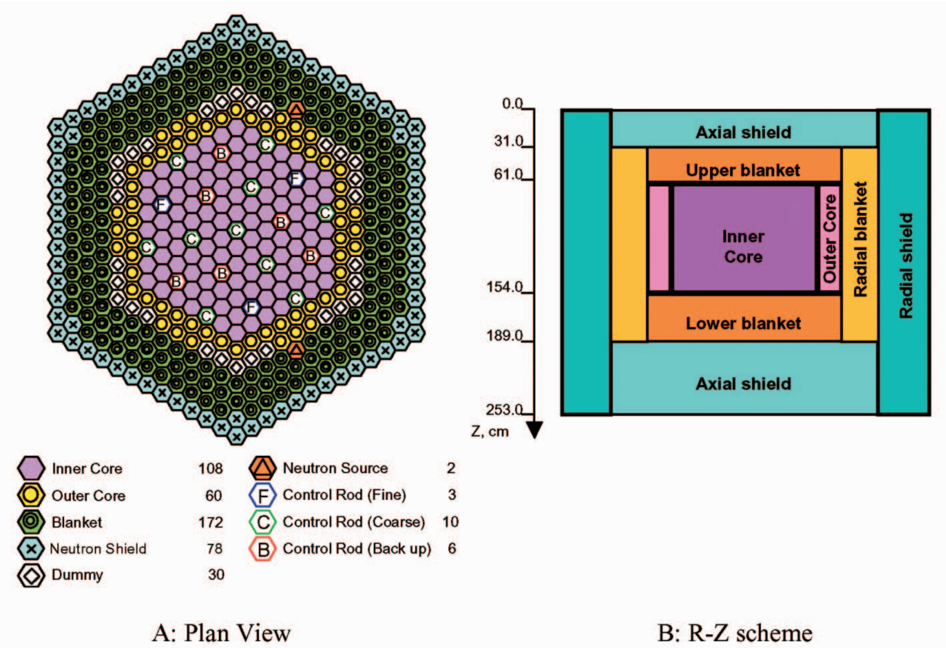

Fig. 2　Layout of FBR“ MONJU ”ICCR model

hours for S4 calculations in 70 energy-group approximation or S8 calculations in 18 energy-group approximation on a UNIX engineering work station (Sun Blade 2000: 64 bit architecture Ultra SPARC III Cu 900 MHz.).

(3) Three MONJU high-leakage models have been introduced for simulation of extremely increased neutron leakage-models ICCS, ICCS1 and ICCS2. The model ICCS (Simplified) considers only inner and outer core in plane with all axial blankets and shields in axial direction (Fig. 3A), so far this numerical model represents the case of increased radial leakage. The model ICCS1 and ICCS2 are the same as ICCS in plane, but in axial direction the ICCS1 considers the case with removed axial shielding (Fig. 3B), while in the ICCS2 the axial blankets are also removed (Fig. 3C) and the model considers only the core region.

2.2. Calculation flow

In cross-section (XS) preparation, both codes

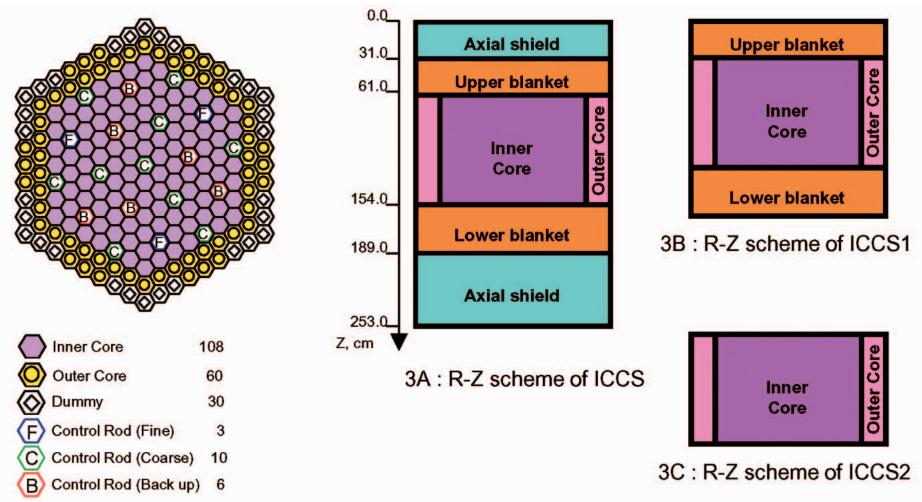

Fig. 3　FBR“ MONJU ”ICCS model

NSHEX and GMVP follow one and the same calculation flow-Fig. 4. Transport calculations by the code NSHEX have been conducted in S_4 discrete-ordinate approximation for different MONJU ICC models under convergence criteria of $5*10^{-5}$ for k_{eff} and $5*10^{-4}$ for neutron fluxes in fissionable material regions. Monte Carlo calculations by the code GMVP have been conducted for the basic model ICCB by 30 million-history particle simulations with the statistical error of $\pm 8*10^{-5}$ for k_{eff} (1σ). The used 70-, 18- and 7-group energy structure is shown in Fig. 5.

(1) Deduction of 70 energy-group macroscopic effective XS

The cell calculation code SLAROM[11] has been used to deduce macroscopic effective XS by homogeneous cell model in a 70 energy-group structure from the JFS-3-J3.2R nuclear data library[12]-a set of multi-group infinite diluted cross-sections, fission spectrum and a table of self-shielding factors, produced by the processing code system NJOY-TIMS[13]. Flux-weighted effective microscopic total and elastic scattering cross-sections $\sigma_{t,g}(\sigma_0)$ and $\sigma_{el,g}(\sigma_0)$ are obtained based on infinitely diluted cross-sections $\sigma(\infty)$ and flux-weighted self-shielding factors $f_{i,g}(\sigma_0)$, $(i = f, c, el, in, n2n)$ as:

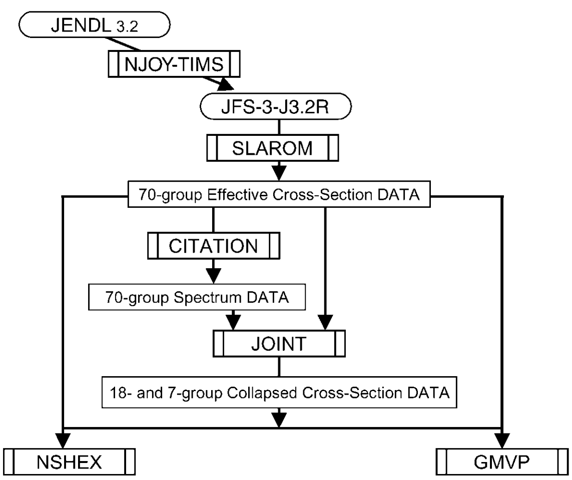

Fig. 4 Calculation flow in transport analyses

$$\sigma_{t,g}(\sigma_0) = f_{f,g}(\sigma_0)\sigma_{f,g}(\infty) + f_{c,g}(\sigma_0)\sigma_{c,g}(\infty) + $$
$$+ f_{el,g}(\sigma_0)\sigma_{el,g}(\infty) + f_{in,g}(\sigma_0)\sigma_{in,g}(\infty) + f_{n2n,g}(\sigma_0)\sigma_{n2n,g}(\infty)$$

and (1)

$$\sigma_{el,g}(\sigma_0) = f_{el,g}(\sigma_0)\sigma_{el,g}(\infty),$$

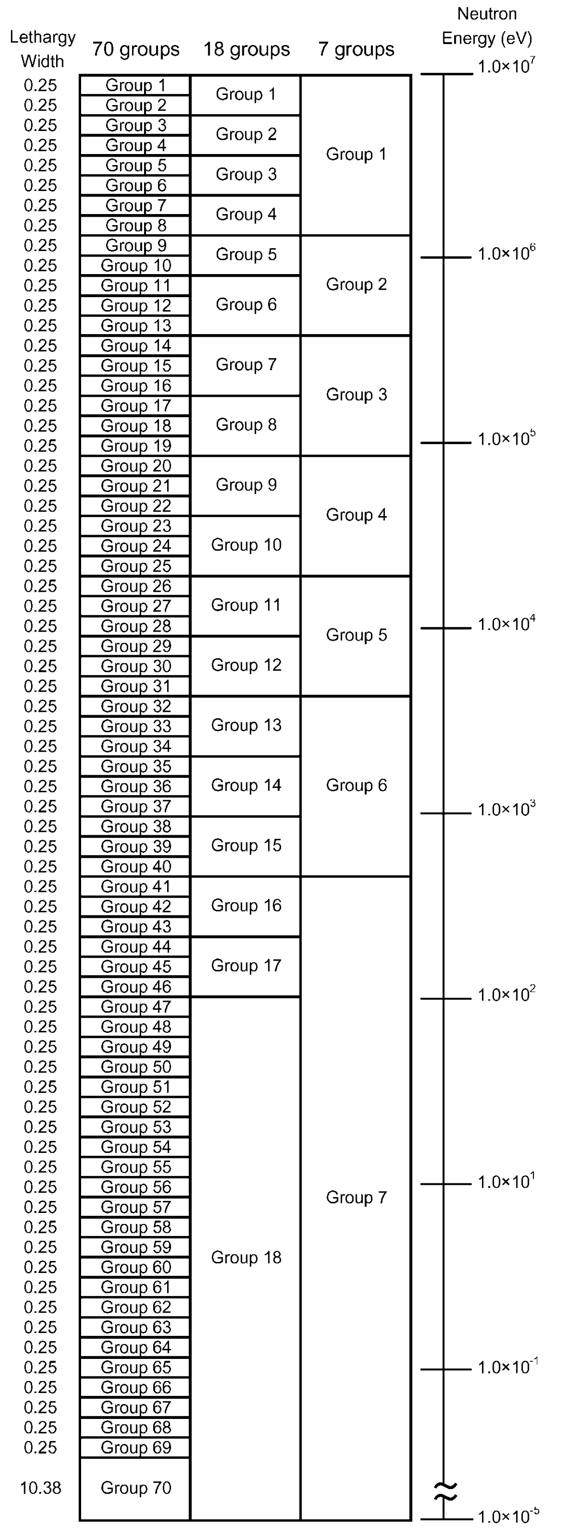

Fig. 5 Description of energy - group structures

where σ_0 is the background cross-section.

Current-weighted total and elastic scattering microscopic cross-sections are calculated respectively by:

$$\sigma^*_{t,g}(\sigma_0) = f^*_{t,g}(\sigma_0)[\sigma_{f,g}(\infty) + \sigma_{c,g}(\infty) + \sigma_{el,g}(\infty) + \sigma_{in,g}(\infty) + \sigma_{n2n,g}]$$
$$\sigma^*_{el,g}(\sigma_0) = \sigma_{el,g}(\sigma_0) + [\sigma^*_{t,g}(\sigma_0) - \sigma_{t,g}(\sigma_0)] \quad (2)$$

The scattering anisotropy is considered by extended P1 transport correction:

$$\sigma_{tr,g}(\sigma_0) = \sigma_{t,g}(\sigma_0) - \mu\sigma_{el,g}(\sigma_0) \quad (3)$$

and

$$\sigma^*_{tr,g}(\sigma_0) = \sigma^*_{t,g}(\sigma_0) - \mu\sigma^*_{el,g}(\sigma_0), \quad (4)$$

where μ is the average cosine of the scattering angle.

Finally, the cell averaged macroscopic 70 energy-group XS of the kind i for each material composition are calculated based on the corresponding microscopic cross-sections and the given atomic number densities : N_j :

$$\Sigma_{i,g} = \sum_j N_j \sigma_{i,g,j}(\sigma_0) \quad (5)$$

$$\Sigma^*_{i,g} = \sum_j N_j \sigma^*_{i,g,j}(\sigma_0) \quad (6)$$

In this study two sets of 70 energy-group XS have been used: S1 (flux-weighted) and S2 (current-weighted). The S1 XS have been applied for some tentative analyses. The current-weighted S2 XS have been deduced and applied in the transport analysis for evaluation of the group-collapsing effect by different XS collapsing methods.

(2) XS collapsing

The collapsing of the macroscopic effective XS into few energy groups (18 or 7) has been performed by the code JOINT[14]:

$$\Sigma_{i,G} = \frac{\sum_{g\in G} \Sigma_{i,g}\Phi_g}{\sum_{g\in G}\Phi_g}, \text{ (G=18, 7)}, \quad (7)$$

where the 70 energy-group collapsing fluxes Φ_g (integrated over the energy in the group g) have been calculated by the 2D RZ option of the diffusion code CITATION[15]. An additional collapsing procedure has been incorporated into the code JOINT for the transport XS:

$$\Sigma^*_{tr,G} = \frac{\sum_{g\in G} \Sigma^*_{tr,g} D^*_g \Phi_g}{\sum_{g\in G} D^*_g \Phi_g}, \text{ (G=18, 7)} \quad (8)$$

where current-weighted diffusion coefficients $D^*_g = 1/3\Sigma^*_{tr,g}$ are used. The collapsing method (8), known as current-weighted method, aims at preserving the neutron mean free path $\lambda_{tr,g}$ and consequently at preserving the leakage term in the neutron balance equation in the collapsed energy-group structure[16]:

$$\lambda_{tr,G} = \frac{\sum_{g\in G} \lambda_{tr,g}\Phi_g}{\sum_{g\in G}\Phi_g} = \frac{\sum_{g\in G} D^*_g \Phi_g}{\sum_{g\in G} \Sigma^*_{tr,G} D^*_g \Phi_g} = \frac{1}{\Sigma^*_{tr,G}},$$

Both flux-weighted and current-weighted collapsing methods have been used in the transport calculations. For simplicity they are noted as J1 (flux-weighted) and J2 (current-weighted) method hereafter. The collapsing method J1 has been used for both S1 and S2 XS, while the collapsing method J2 has been applied for condensation of the S2 70 energy-group XS.

3. Results

3.1 Energy-group approximation effect

The energy-group approximation effect has been studied in different core models for both J1 and J2 collapsing methods. The results are presented as relative percentage differences %Δ k/k^{70}, *i.e.*

$$(k^g_{eff} - k^{70}_{eff}) / k^{70}_{eff} * 100\%, \text{ g = 18, 7.}$$

It has been realized that the collapsing method J1 causes significant overestimation of k_{eff}, proportional to the degree of the collapsing as shown in Table 1. The group collapsing effect estimated by GMVP is independent of the SLAROM conditions S1 or S2. For the S1 XS the deterministic results are in a good agreement with the Monte-Carlo results, while for the S2 XS the group-collapsing effect is slightly smaller. The group-collapsing

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Table 1. The energy - group collapsing effect for flux - weighted transport XS

Code	SLAROM	Model	% Δ *k*/*k* ^{70}	
			70→18	70→7
GMVP*	S1	ICCB	0.19	0.35
GMVP*	S2	ICCB	0.19	0.35
NSHEX	S1	ICCR	0.18	0.34
NSHEX	S2	ICCR	0.13	0.27
NSHEX	S1	ICCS	0.25	0.46
NSHEX	S2	ICCS	0.19	0.37

*) 1σ = ±8pcm

effect is more significant in the model ICCS, where an increased neutron leakage takes place. It can be supposed that the transport XS in 18- and 7-group structures are overestimated in some of the energy groups, which causes an underestimation of the neutron leakage in these groups and results in an overestimation of k_{eff}. The results lead to the conclusion that the collapsing method J1 is not appropriate in transport analysis, as it cannot provide proper condensation of the transport XS.

Then the current-weighted method J2 has been applied to the transport XS collapsing by the code JOINT. The 70 energy-group macroscopic XS have been deduced by SLAROM S2 method. One item that should be pointed out is the reformulation of the self-scattering XS based on the current-weighted transport XS. This reformulation is applied to NSHEX analysis in XS processing by the code JOINT. An analogical algorithm for the self-scattering XS reformulation has been additionally applied to GMVP data processing, otherwise the scattering XS exceeds the transport XS and the calculations would fail.

Figure 6 shows the Monte Carlo and the deterministic results by the collapsing method J2. The GMVP analysis produces results that include smaller collapsing effect-below -0.05%. By NSHEX whole core model ICCR, the effect has been estimated to be -0.12% for 18 and -0.19% for 7 energy groups. The results for k_{eff} of the MONJU ICC increased-leakage models are more significantly underestimated in 18 and 7 energy-group analysis as shown in Table 2. In all cases, the energy-group collapsing effect by the NSHEX analysis is proportional to the degree of the collapsing and obviously follows different energy-group dependency than the GMVP results.

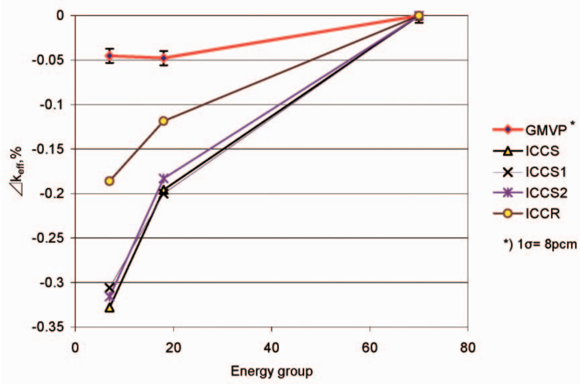

Fig.6 Energy - group collapsing effect by current - weighted transport XS

Table 2. The energy - group collapsing effect for current - weighted transport XS

Code	SLAROM	Model	% Δ *k*/*k* ^{70}	
			70→18	70→7
GMVP*	S2	ICCB	-0.05	-0.05
NSHEX	S2	ICCR	-0.12	-0.19
NSHEX	S2	ICCS	-0.20	-0.33
NSHEX	S2	ICCS1	-0.20	-0.31
NSHEX	S2	ICCS2	-0.19	-0.32

*) 1σ = ±8pcm

3.2 Application of the NSHEX to MONJU criticality analysis

The criticality analysis by the code NSHEX for MONJU ICC produces results that are in good agreement with the Monte Carlo results, taking into account the complexity of the solved problem-MONJU initial critical core in whole core simulation, where the dummy fuel sub-assemblies introduce a significant heterogeneity. The calculations have been conducted from 7 to 70 energy-group approximations and from simplified partial-core models to the whole core simulation. The wide range of solved problems allows some practical conclusions to be extracted that are important for the implementation of the code in the standard criticality analysis of a large fast breeder reactor core like FBR MONJU.

The deterministic analysis by the 3D code NSHEX has been successfully applied for evaluation of effects and dependencies like the energy-group approximation dependency of k_{eff}. In such an analysis the relative change of k_{eff} is more important than the absolute value. However, if the exact value of k_{eff} is to be determined, for example for the estimation of the transport effect correction or for the comparisons between NSHEX and the Monte-Carlo results, some numerical difficulties appear. The common origin of these difficulties is associated with the convergency of the NSHEX nodal-equivalent method in whole core simulation. To attain fluxes to be converged in 18 and 70 energy-group analyses, the calculations have to be conducted with additional modifications on the convergency criteria. Then the influence of these additional conditions in the final results has to be evaluated. For example, in case of simulating the whole core problem by the model ICCR, the neutron fluxes converge in all regions only in 7 energy-group approximation. In 18 and 70 energy-group analysis the fluxes converge under restricted convergency criteria that have been applied only for the inner core (IC) and the outer core (OC). Then the difference between k_{eff} found at the different convergency criteria has been estimated from the 7 energy-group results. Figure 7 shows the convergence rates in 7 energy-group analyses. In case C1 the convergency criteria are applied to fluxes in IC and OC, while in case C2 the convergence judgment takes the fluxes in all regions into account. It can be noticed that the fluxes in the IC and OC converge faster than in the other regions, so the value of k_{eff} found under the convergency criteria C1 is somewhat underestimated. By model ICCR this underestimation has been evaluated to be 100 pcm for 7 energy-group results. The same correction has been then applied to the 18 and 70 energy-group results, assuming that the influence of the convergency criteria modification correction is the same in these energy-group approximations.

The relative percentage differences between NSHEX and GMVP results are sown in Fig. 8 where two types of corrections have been separately applied to the deterministic results: correction for model ICCR against the model ICCB (used by GMVP) and correction for the restricted application of the convergency criteria. Depending on the applied corrections, the relative differences between the NSHEX and the GMVP results vary from -0.11 to 0.03% Δk (70 energy-group) and from -0.25 to -0.11 % Δk (7 energy-group). The differences are comparable with these found by the verification tests, where the disagreements between NSHEX and GMVP results are in the range of 0.02 to 0.10% Δk for KNK-II benchmark and of 0.19 to 0.32 % Δk for the large assembly-size KNK-II model.

The most important difference between GMVP and NSHEX results is that they show a different energy-group approximation dependency of k_{eff}. While in the Monte Carlo analysis the XS collapsing effect (for current-weighted transport XS) has been found to be relatively small; below 0.05% Δk as shown in Fig.6, the effect is still significant; al-

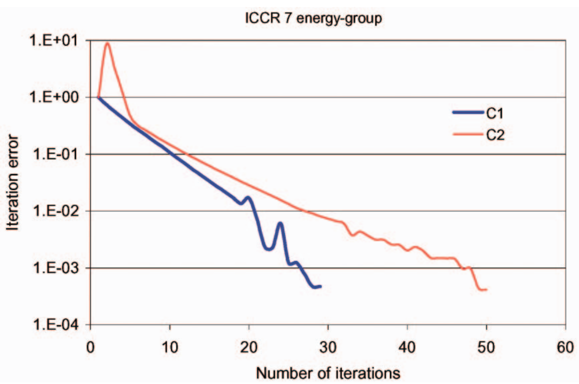

Fig.7 Convergence rate under different judgment criteria

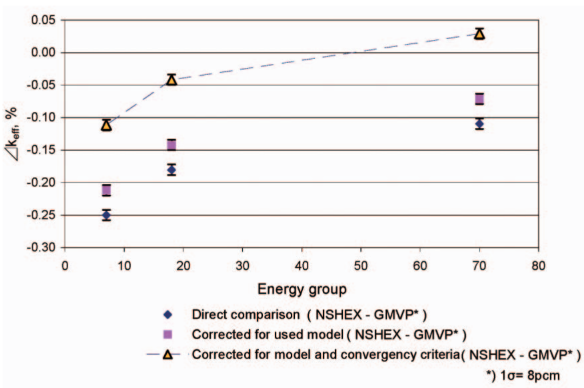

Fig.8 Comparison between GMVP and NSHEX results for MONJU ICC

most 0.2% Δ *k* for the few-group deterministic analysis. However, the few-group calculations are of prime practical importance due to their better numerical efficiency, hence this energy-group dependency should be investigated carefully. It has been clarified that the effect is proportional to the neutron leakage by the different models. For NSHEX some implication has been found that the neutron leakage is not satisfactorily simulated by the 7- and 18-group analysis especially in the increased-leakage MONJU ICC 3D core models. The application of such a models therefore demands an improvement in the description of the neutron leakage by introduction of more precise boundary conditions or by another XS collapsing algorithm that can provide independency from the energy-group approximation for whole core models and also for the simplified core models as well.

4. Conclusions

MONJU ICC criticality problem has been solved by deterministic and Monte Carlo transport methods. The 3D hexagonal transport code NSHEX has been applied for wide range of 3D core models. The analysis of up to 70 energy-groups under various appropriate conditions has been carried out. The results and the dependencies have been investigated, and a new finding was obtained. The NSHEX results have been compared with the corresponding GMVP results.

The energy-group approximation effect in the few-group analysis has been estimated to be -(0.05±0.01)% $\Delta k/k^{70}$ by the Monte-Carlo calculations. Although the amount is not so large, it is several times larger than that of the diffusion calculations. And these results are based on the current-weighted method for the transport XS. Much larger energy-group dependency was identified by the flux-weighted collapsing method. The energy-group approximation effect of k_{eff} has been found by NSHEX calculations to be several times larger than that of the Monte Carlo analysis and proportional to the degree of the XS collapsing. Moreover it has been confirmed that the influence is more significant for the simplified core models with an increased neutron leakage. This implies that the neutron leakage is not satisfactorily simulated by the 7- and 18-group analysis and there is room for improvement in the present collapsing algorithm and/or in the boundary conditions that are incorporated in the code NSHEX.

The NSHEX results are in good agreement with the GMVP results, especially without collapsing, for MONJU initial critical core criticality analysis. Hence the applicability of the code NSHEX to the MONJU whole core analysis has been confirmed. However, some difficulties associated with the convergence of the nodal method and with the estimation of the neutron leakage still remain and have to be resolved for full implementation of the code in FBR core analysis.

The wide range of solved problems by the code NSHEX allows some practical conclusions to be extracted that are important for the implementation of the code in the standard criticality analysis. First of all, if the analysis is carried out in a few-group approximation, the results have to be corrected with the corresponding energy-group approximation effect correction. Second, the energy-group approximation effect correction is not universal-it depends on the core model, especially the leakage term. The simplified core models have to be used with an increased attention in few-group approximations. It can be even recommended such a simplified simulations to be solved in 70 energy-group approximation. Otherwise an energy-group collapsing effect correction that corresponds to the increased-leakage model has to be applied.

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