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Investigation of the Core Neutronics Analysis Conditions for Evaluation of Burn-up Nuclear Characteristics of the Next-generation Fast Reactors

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Since next-generation fast reactors aim to achieve a higher core discharge burn-up than conventional reactors do, core neutronics design methods must be refined. Therefore, a suitable analysis condition is required for the analysis of burn-up nuclear characteristics to accomplish sufficient estimation accuracy while maintaining a low computational cost. We investigated the effect of the analysis conditions on the accuracy of estimation of the burn-up nuclear characteristics of next-generation fast reactors in terms of neutron energy groups, neutron transport theory, and spatial mesh. This study treated the following burn-up nuclear characteristics: criticality, burn-up reactivity, control rod worth, breeding ratio, assembly-wise power distribution, maximum linear heat rate, sodium void reactivity, and Doppler coefficient for the equilibrium operation cycle. As a result, it was found that the following conditions were the most suitable: 18-energy-group structure, 6 spatial meshes per assembly with diffusion approximation. Additionally, these conditions should apply to correction factors for energy group structure, spatial mesh and transport effects.

Keywords: Next-generation Fast Reactors, Core Neutronics Design, Burn-up Nuclear Characteristics, Analysis Condition, Correction Factor

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次世代高速炉の核設計における燃焼核特性評価の解析条件の検討

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(2023 年 1 月 27 日受理)

次世代高速炉は、従来炉よりも高い炉心取出燃焼度を目指しているため、炉心核設計の高度化が求められる。そのため、燃焼核特性解析では、計算コストを抑えつつ十分な計算精度が得られる適切な解析条件が必要とされる。そこで、次世代高速炉の燃焼核特性の計算精度に及ぼす解析条件の影響を、中性子エネルギー群、中性子輸送理論、空間メッシュに着目して調査した。本検討では燃焼核特性として、平衡サイクルにおける臨界性、燃焼反応度、制御棒価値、増殖比、集合体単位の出力分布、最大線出力、ナトリウムボイド反応度、ドップラー係数を取り扱った。検討の結果、エネルギー群を 18 群とし、拡散近似を用いて 1 集合体あたり 6 メッシュ分割して、エネルギー群、空間メッシュ、輸送効果の補正係数を適用することが最適であることが分かった。

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1. Introduction

Sustainable availability of electricity is required due to the risk of exhaustion of natural recourse in the future. Further, energy resources that do not emit greenhouse gases should also be investigated to reduce global warming. Fast reactors (FRs) have the potential to cope with these issues because FRs can breed plutonium, while maximizing the utilization of uranium resources, and can generate electricity without emitting greenhouse gases. Therefore, many countries are investigating and developing FR technology.

In Japan, the Feasibility Study on Commercialized Fast Reactor Cycle Systems [1)] (FS) was conducted during fiscal years 1999-2005. The FS selected the FR cycle systems that could best satisfy the developmental targets. Based on the FS results, Japan adopted a combination of a sodium-cooled FR with oxide fuel and a fuel cycle system which consists of advanced aqueous reprocessing and simplified pelletizing fuel fabrication.

The FS was followed by the Fast Reactor Cycle Development (FaCT) project, which was conducted during fiscal years 2006-2010. This project substantiated the developmental targets in terms of the technical specifications and performance of the FR cycle systems. The FaCT project developed the concept of the next-generation fast reactor, which is called the Japan Sodium-cooled Fast Reactor (JSFR). In particular, for the enhancement of economy in nuclear power generation, the whole core discharge burn-up (including core and blanket) of JSFR was determined to achieve a burn-up of greater than 80 GWd/t. Therefore, the core discharge burn-up of JSFR aimed at approximately 150 GWd/t. Such a high discharge burn-up ensures that the JSFR remains competitive against next-generation light-water reactors.

Since a high core discharge burn-up tends to increase the uncertainty of the core burn-up calculations, it may require more control rods and a higher coolant flow rate that may deteriorate the core performance. Therefore, the JSFR core burn-up calculation requires refinement of analysis conditions. However, the more refined the analysis conditions adopted are, the longer the computation time may take. In this study, we investigated the effect of the analysis conditions on the accuracy of the calculations to evaluate the burn-up nuclear characteristics of the next-generation fast reactors. Subsequently, we identified suitable analysis conditions that can accomplish sufficient accuracy, while enabling low-cost core burn-up calculations.

A part of this study was presented in the previous paper [3)]. This paper includes further discussion about the mechanism behind the difference on low-cost core burn-up

calculation, as well as the applicability of the representative correction to various fresh fuel nuclide compositions.

2. Core specifications and analysis conditions

In this section, we describe the specifications of an evaluated core and define the analysis conditions for surveying low-cost calculations.

2.1 JSFR core specifications

We selected a demonstration-scale 750-MWe JSFR core that was designed in the FaCT project. Figure 2.1-1 depicts the layout of the core, and Table 2.1-1 lists the main design parameters of the reactor. This core has a long operation cycle length of 18 months. The core height is 100 cm. The core is surrounded by the upper and lower blankets having thicknesses of 20 and 25 cm, respectively, and one-layer of radial blanket. The fuel form is mixed uranium-plutonium oxide. The whole core average discharge burn-up aims to achieve greater than 80 GWd/t, as mentioned above.

2.2 Burn-up nuclear characteristics

The following important burn-up nuclear characteristics were calculated: criticality (k_{eff}), burn-up reactivity (BuR), control rod worth (CRW), breeding ratio (BR), assembly-wise power distribution (APD), maximum linear heat rate (MLHR), sodium void reactivity (SVR), and Doppler coefficient (DC). The CRW was calculated as the total worth of all the coarse control rods. These nuclear characteristics were calculated for the equilibrium operation cycle. In this study, we considered the 13th cycle as an equilibrium operation cycle.

With the exception of SVR and DC, all characteristics were calculated both at the beginning and the end of the equilibrium cycle (BOEC and EOEC, respectively). Since SVR and DC were found to be conservative when using the EOEC results, they were obtained by using snapshot calculations at EOEC only. The snapshot calculations for these reactivity coefficients were usually performed by 70 or greater neutron energy groups. To ensure conservativeness, all control rods were totally withdrawn from the core during the snapshot calculations at EOEC. In the SVR calculation, the sodium in the fuel pin bundle part was voided, but that of the outside of wrapper tube remained. In the DC calculation, the fuel temperature was raised by 500 °C.

2.3 Design envelope for fresh-fuel nuclide composition

The JSFR core has a design scope for fresh-fuel nuclide composition which varies from the FR deployment phase to the FR's multi-recycle equilibrium phase. The design envelope is based on the correlations among the important burn-up nuclear

characteristics [4] such as BuR, SVR, DC. For example, Fig. 2.3-1 depicts the linear correlation between BuR and SVR. The highest BuR (or lowest SVR) was obtained using a fuel containing high-fissile plutonium (referred to as U-Pu high fissile fuel). Conversely, the lowest BuR (or highest SVR) was obtained using a fuel containing degraded plutonium and minor actinides (referred to as transuranic (TRU) low fissile fuel). In this study, the analysis conditions for burn-up nuclear characteristics were inclusively investigated using these two fuel compositions.

Table 2.3-1 shows the composition of U-Pu high fissile and TRU low fissile fuel used in the JSFR core neutronics design [5,6] together with the burn-up nuclear characteristics evaluated in this study. The breeding ratios were approximately 1.1. The whole core average discharge burn-ups exceeded 80 GWd/t. The BuR, APD, and MLHR for the U-Pu high fissile fuel core were higher than those for the TRU low fissile fuel core. On the other hand, the TRU low fissile fuel core had larger SVR, smaller absolute value of DC, and smaller CRW than those of the U-Pu high fissile fuel core.

2.4 Analysis conditions for surveying the low-cost calculations

The following treatments often affect analysis results: simplifying the cell model to perform the cell calculations, coarsening the neutron energy group and spatial mesh, and applying diffusion approximation to the neutron transport calculations. In terms of the cell calculations, it was confirmed that the one-dimensional (1-D) heterogeneous cell model worked correctly on the fuel and control rod [7,8]. Additionally, the computation time hardly increased even when a 1-D heterogeneous cell model was adopted for the cell calculations instead of the homogeneous cell model. Therefore, we adopted a 1-D heterogeneous cell model for the cell calculations of the fuel and control rod. Then, we considered two low-cost treatments: for the neutron energy group, and for the spatial mesh and neutron transport.

The cost of calculations decreased using the foregoing low-cost treatments. To measure the calculation accuracy and cost, we defined the referential detailed conditions and low-cost calculation conditions for each treatment, as depicted in Table 2.4-1 and Table 2.4-2, respectively.

Regarding the treatment of the neutron energy group, the core burn-up calculation using 175-group cross sections (175G) was adopted as the referential detailed condition, where a part of the 175G (below 50 keV) was produced by a hyper-fine group structure cell calculation [9,10]. The low-cost calculations adopted the following energy groups: The 70G, 36G, 18G, and 7G, where G denotes groups. 70G adopts the effective cross sections that were produced by the cell calculation with the 70-group structure

constant set. The 36G, 18G, and 7G adopt the effective cross sections condensed from those of 70G with the neutron spectrum obtained by core calculation.

For the treatment of the spatial mesh and neutron transport, a combination of 24 radial meshes per assembly (corresponding to about 5-cm spatial meshes) and S_4 angular quadrature that was set using P_0 treatment in transport theory (M24T) was adopted as the referential detailed condition because it can be regarded as a transport calculation with an infinitely small spatial mesh and angular quadrature due to the cancellation of spatial mesh and angular quadrature effects [11]. Further, note that we have used the transport cross section instead of the total cross section. The low-cost calculations adopted the case of coarsening the number of meshes per assembly from 24 to 6 (M6T), the application of diffusion approximation (M24D), and both of them (M6D).

In this study, the low-cost calculations are called base calculations. Snapshot corrections are applied to all of the burn-up nuclear characteristics of the base calculations in order to bring them close to the result of the referential detailed calculation. These correction factors are typically evaluated using the snapshot calculations of the low-cost and referential detailed conditions, where a preliminary or representative fuel composition can be used. In this study, we selected the EOEC nuclide composition of 7G-M6D that yielded the lowest cost to be the representative composition.

This study estimated the degree of agreement between the burn-up nuclear characteristics of the referential detailed conditions and those of base calculations with the snapshot corrections applied. The degree of agreement was estimated by the reference accuracy; it was tentatively fixed by the cross-section induced uncertainty (1σ) of the nuclear characteristics of the 750-MWe JSFR core, which was estimated by the accumulation of cross section covariance on JENDL-4.0 [12].

The computation time is the time required to perform the following calculations: the base, snapshot for SVR and DC, and the snapshot for corrections. The base calculations contain cell, condensation, and core burn-up calculations. The cell calculations took 1.3 hours for 175G and a few minutes for 70G. All condensation calculations took a few minutes. The computation time of core burn-up calculations refer to that consumed until the end of the 13th cycle.

2.5 Calculation method

Figure 2.5-1 shows the flow chart of the core burn-up calculation. Every core burn-up calculation was performed on the versatile reactor analysis code system, named MARBLE2 [13,14], by modeling the control rod insertion and refueling. We assumed that the 750-MWe JSFR was operated at full power and that the reactor shutdown period for

refueling and regular inspection was negligible to ensure simplicity. These calculations were performed using ORPHEUS, which is a fast reactor core burn-up analysis code system in MARBLE2. ORPHEUS has neutron flux calculation solvers based on diffusion and transport theories. Various patterns of spatial mesh division are also available in this code system.

To perform the cell calculations, the cells of fuels and control rods were modeled using 1-D multi-ring models [7,8]. For the other regions, we adopted homogeneous models and used the SLAROM-UF code [9,10] with a fast reactor group constant set, UFLIB.J40 [15]. During the condensation process of the microscopic cross sections, we selected the DIF3D code [16] as a neutron flux calculation tool. To perform neutron flux calculation during condensation, we used a three-dimensional Triangle-Z geometry model.

To perform the core calculations, we adopted a three-dimensional Triangle-Z geometry model similar to the methodology used for condensation. We also selected DIF3D as a diffusion solver with the option of a finite-difference method. However, ORPHEUS had no transport solver in the three-dimensional Triangle-Z geometry model. Therefore, we attached the MINISTRI code [17] to obtain the transport effect. MINISTRI is an S_N transport calculation code based on a finite-difference method.

To perform the burn-up calculations, we used a “BURNUP” solver that is developed into MARBLE2. Every burn-up calculation adopted a set of zones in a three-dimensional Hexagonal-Z geometry for storing the burn-up nuclide compositions.

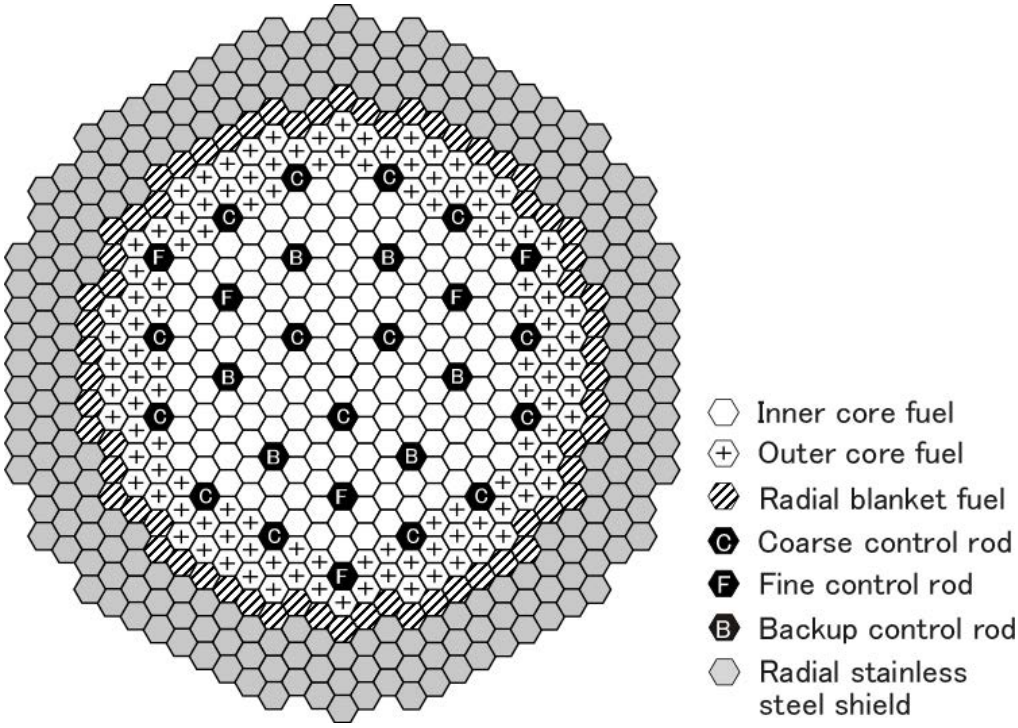


Fig. 2.1-1　Core layout of the 750-MWe JSFR

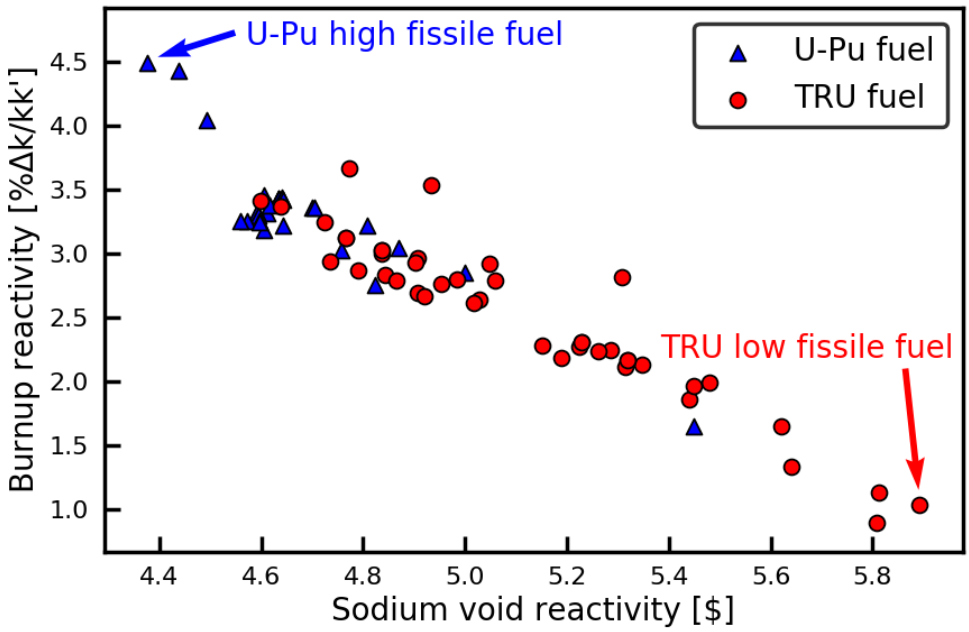


Fig. 2.3-1　Correlation of sodium void reactivity and burn-up reactivity [4)]

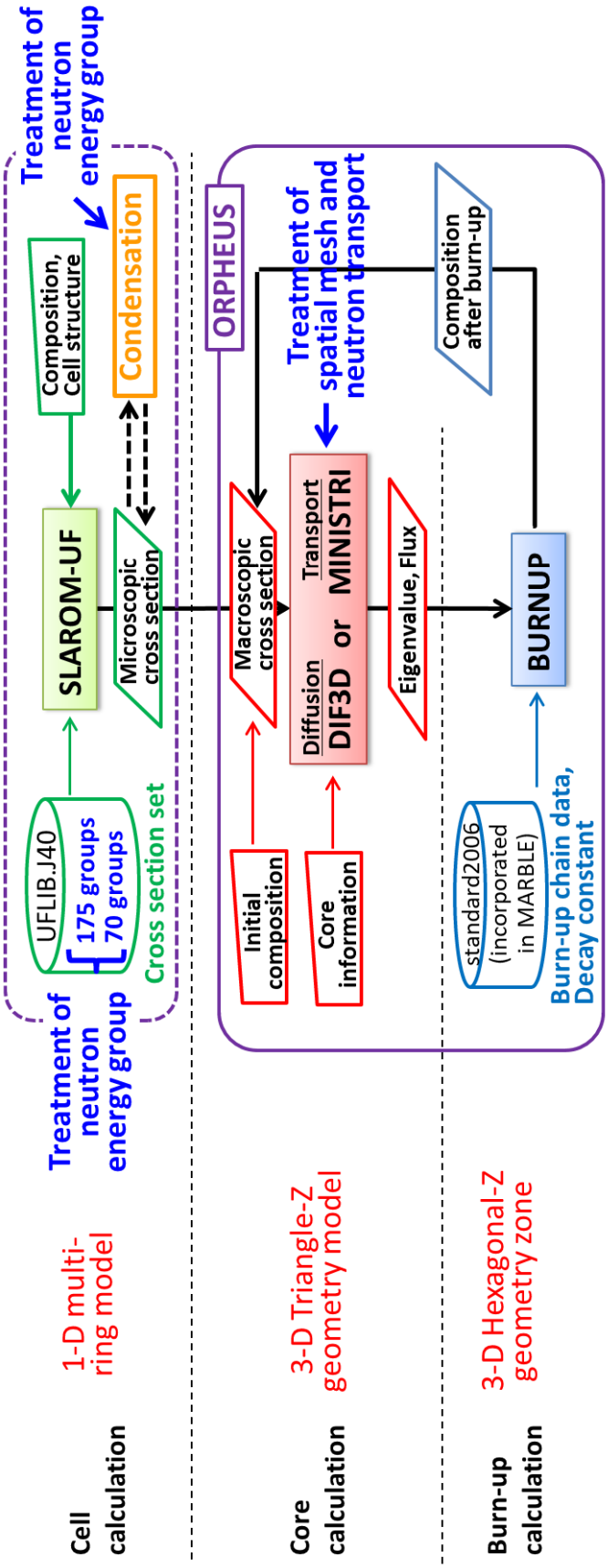


Fig. 2.5-1 Flow chart of core burn-up calculation

Table 2.1-1 Main design parameters of the 750-MWe JSFR

Item	Value
Plant parameters	
Power output [MW] (electric / thermal)	750 / 1765
Coolant temperature [°C] (outlet / inlet)	550 / 395
Operation cycle length [month]	18
Core specifications	
Core height [cm]	100
Axial blanket thickness [cm] (upper / lower)	20 / 25
Refueling batch (core / blanket)	6 / 6
Fuel specifications	
Fuel form	Oxide
Fuel smear density [%TD]	82

Table 2.3-1 Fuel specifications and nuclear characteristics of the 750-MWe JSFR

Item	U-Pu high fissile fuel core	TRU low fissile fuel core
TRU composition ratio [wt%]		
^{238}Pu/^{239}Pu/^{240}Pu/	2.5 / 52.4 / 26.2/	1.7 / 46.7 / 23.6/
^{241}Pu/^{242}Pu/^{241}Am	10.3 / 7.6 / 1.0	2.0 / 6.7 / 11.5/
^{237}Np/^{243}Am/^{244}Cm	0.0 / 0.0 / 0.0	6.2 / 1.4 / 0.2
Core characteristics		
Pu enrichment [wt%]		
(inner core / outer core)	18.1 / 24.3	18.5 / 24.4
Breeding ratio	1.09	1.13
Average discharge burn-up [GWd/t]		
(core / whole core)	150 / 84	157 / 86
Maximum linear heat rate [W/cm]	411	400
Burn-up reactivity [%Δk/kk’]	2.7	1.3
Sodium void reactivity [$]	4.7	5.6
Doppler coefficient [Tdk/dT]	-6.0×10^{-3}	-4.5×10^{-3}

Table 2.4-1 Calculations for the treatment of neutron energy group

	Detailed condition	**Low-cost conditions**			
Case name	**175G (Corresponding to the continuous energy)**	**70G**	**36G**	**18G**	**7G**
Neutron energy group	175 groups	70 groups	36 groups	18 groups	7 groups
Group structure	Modified VITAMIN-J-175 type	JFS-3 type	Condensed from JFS-3 type	Condensed from JFS-3 type	Condensed from JFS-3 type
	Hyper-fine group (cell calculation only) (E_n < 50 keV)	Lethargy width: 0.25	Lethargy width: 0.50	Lethargy width: 0.50 (E_n > 820 keV), 0.75 (E_n < 820 keV)	Lethargy width: 1.00 (E_n > 820 keV), 2.25 (100 eV < E_n< 820 keV)

Table 2.4-2 Calculations for the treatment of spatial mesh and neutron transport

	Detailed condition	**Low-cost conditions**		
Case name	**M24T (Corresponding to the infinite small spatial mesh and S_N angular quadrature)**	**M6T**	**M24D**	**M6D**
Radial meshes / assembly	24	6	24	6
Flux calculation theory	Transport ($S_4 P_0$)	←	Diffusion	←

3. Analysis results

In this section, we discuss the calculation accuracy and computation time of the low-cost treatment of the neutron energy group along with that of the spatial mesh and neutron transport.

3.1 Treatment of neutron energy group

Figure 3.1-1(a) and (b) depict the results of the burn-up nuclear characteristics for the low-cost treatment of the neutron energy group for the U-Pu high fissile and TRU low fissile fuel cores, respectively. The maximum difference from the result of the referential detailed conditions is compared with the corresponding reference accuracy for each characteristic. Note that both the maximum difference and reference accuracy for BuR are represented in absolute values.

Most of the results that were obtained for low-cost calculation and referential detailed conditions showed agreement within the reference accuracy. A few nuclear characteristics exceeded the reference accuracy of 1σ. k_{eff} by 7G and APD at the blanket by 18G were still within the reference accuracy of 2σ. However, APD at the blanket by 7G significantly exceeded the reference accuracy.

To understand the reason why APD at the blanket by 7G significantly exceeded the reference accuracy, we investigated plutonium-239 inventory in a blanket assembly, because plutonium-239 mainly causes fission in the blanket region and affects APD there. Figure 3.1-2 depicts the results of plutonium-239 inventory in a blanket assembly. This figure indicates that plutonium-239 inventory decreased as neutron energy group coarsened. In terms of 7G, plutonium-239 inventory was 4% lower than that obtained under the referential detailed condition. The same tendency was also seen for the TRU low fissile fuel core. As shown in Fig. 3.1-3, the discrepancy of APD at the blanket is proportional to the discrepancy of plutonium-239 inventory. The low-cost burn-up calculations with coarsened neutron energy group cannot properly estimate the neutron spectrum change during burnup, which decreases the capture reaction of uranium-238 and the resulting production of plutonium-239.

If the snapshot corrections are almost equivalent on any fresh-fuel TRU compositions, we can use the snapshot correction obtained with the representative fuel composition. Table 3.1-1 shows the maximum snapshot corrections and their variations by changing fresh-fuel TRU compositions. According to Table 3.1-1, if calculations adopt an 18-group or more detailed structure, snapshot corrections become almost equivalent. Hence the snapshot corrections obtained with the representative fuel composition with

an 18-group or more detailed structure are universally effective for the burn-up nuclear characteristics of any fresh-fuel TRU composition in the range discussed in this study.

The computation time and calculation accuracy in Table 3.1-2 were assessed by verifying that they were within the range of the reference accuracy to ensure proper treatment of the neutron energy group. The computation time of the base calculation was found to decrease significantly and yield approximately similar values for 36G, 18G, and 7G. This was mainly due to the decrease in the computation time of the core burn-up calculation. However, 7G did not satisfy the target accuracy. Because the snapshot corrections need to be calculated only once with the representative fuel composition, 18G was estimated to be the best case among 70G, 36G, 18G, and 7G. The 18G enables the fastest calculation while meeting the calculation accuracy.

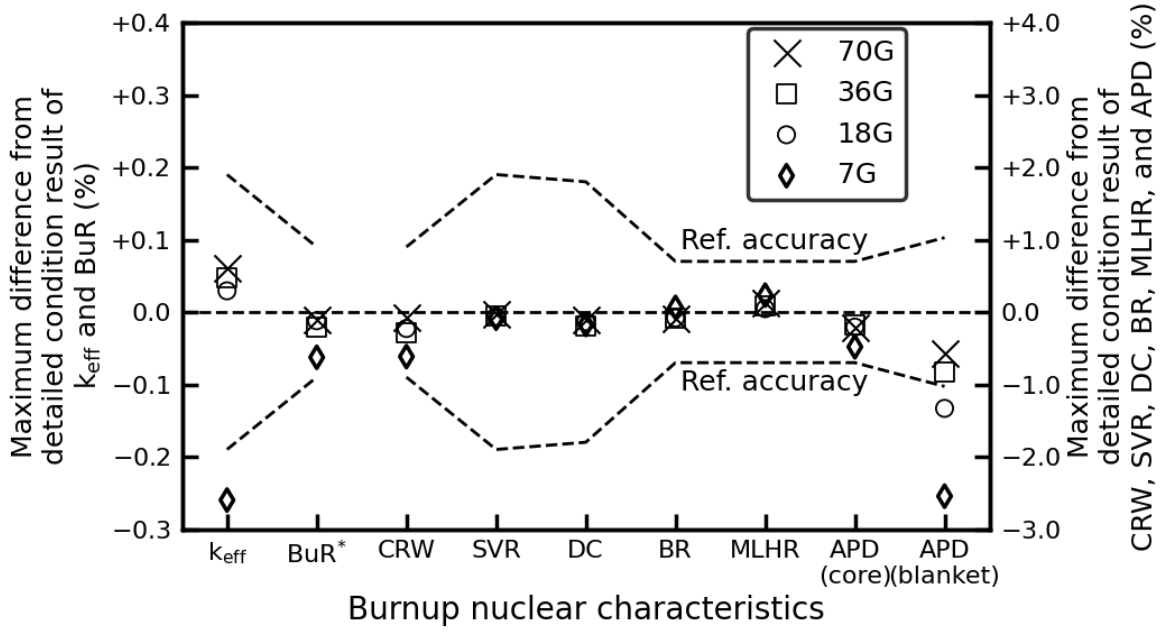
3.2 Treatment of the spatial mesh and neutron transport

Figure 3.2-1 shows the results of the low-cost treatment of spatial mesh and neutron transport for burn-up nuclear characteristics. For APDs, the snapshot corrections at BOEC and EOEC were calculated by modeling respective control rod insertion patterns. The agreement with the result under referential detailed condition became satisfactory for the case of M6D, which is the coarsest. On the other hand, those of certain nuclear characteristics for M24D exceeded the reference accuracy. In terms of the treatment of spatial mesh and neutron transport, neutron flux distribution was strongly affected by the amount of fissile nuclides in a fuel assembly. In M6D, regarding this effect, coarsening mesh on diffusion theory counteracted applying diffusion approximation. On the other hand, M24D has only the effect of applying diffusion approximation. Therefore, the difference of M24D became large.

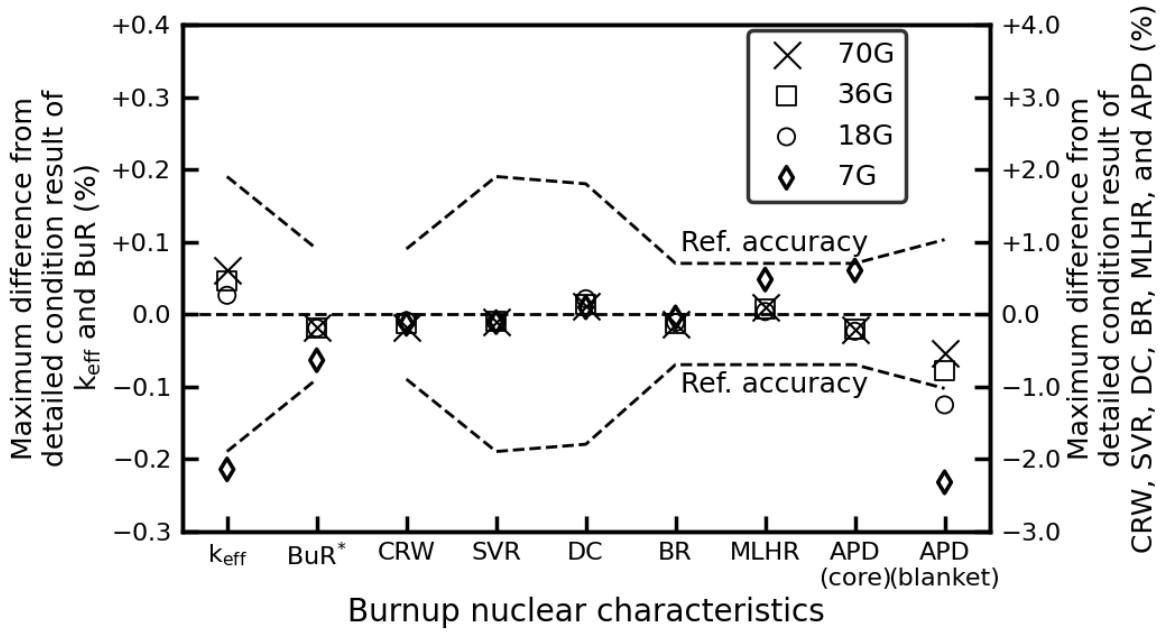
As for the treatment of neutron energy, we investigated the variation of the snapshot corrections by changing fresh fuel TRU compositions for the universal use of the representative snapshot correction to any fresh fuel TRU composition. Table 3.2-1 shows the maximum snapshot corrections and their variations by changing fresh fuel TRU compositions. According to Table 3.2-1, in terms of spatial mesh and neutron transport, the snapshot corrections were almost equivalent regardless of changes in fresh fuel TRU composition. Hence the snapshot corrections are universally effective for the burn-up nuclear characteristics of any fresh fuel TRU composition assumed in this study.

Table 3.2-2 shows the computation time and the adequacy of calculation accuracy, regarding the treatment of spatial mesh and neutron transport. Note that the neutron transport calculations were performed with 12 parallel threads for M24T and

M6T. In spite of applying diffusion approximation, the total computation time of M24D did not significantly differ from that of M24T. Furthermore, M24D did not meet the target of calculation accuracy. On the other hand, in the cases with 6 meshes per assembly, diffusion approximation made the computation time shorter. Consequently, M6D needed the shortest total computation time. This case also met the calculation accuracy target. Considering the computation time and calculation accuracy, M6D was preferable for calculation of the burn-up nuclear characteristics. M6D reduced the total computation time of M24T by 27%.



(a) U-Pu high fissile fuel core



(b) TRU low fissile fuel core

Fig. 3.1-1 Maximum difference of low-cost calculation (Treatment of neutron energy group)

* The absolute difference in the unit of %Δk/kk'

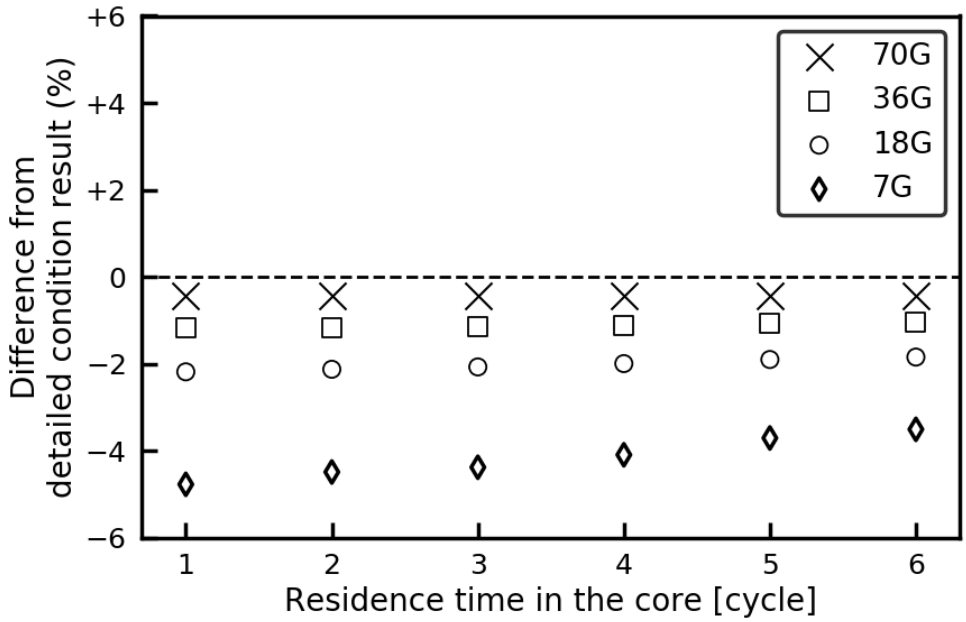


Fig. 3.1-2 ^{239}Pu inventory discrepancy from the detailed condition result in a blanket assembly (U-Pu high fissile fuel core)

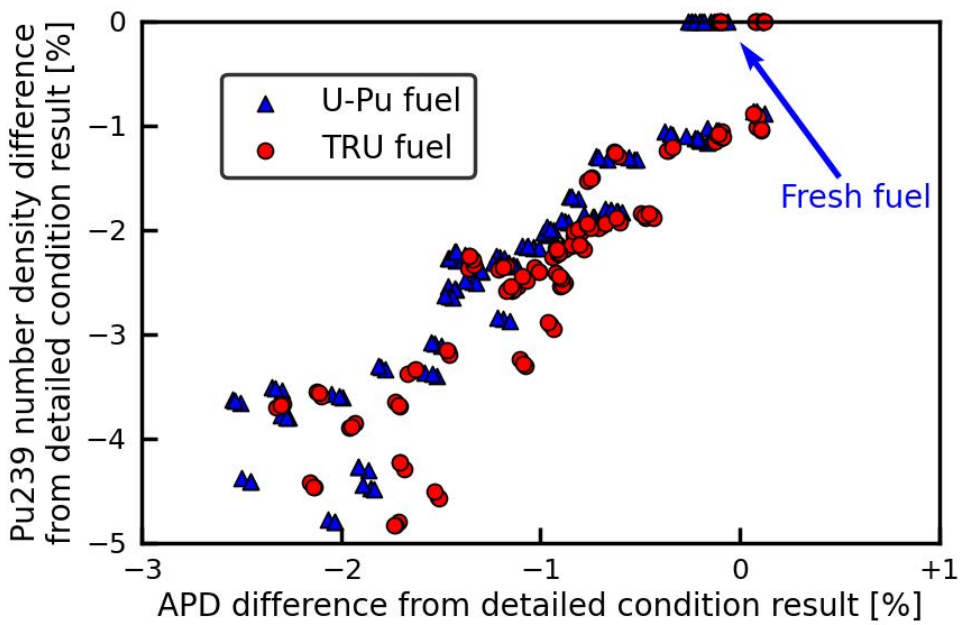
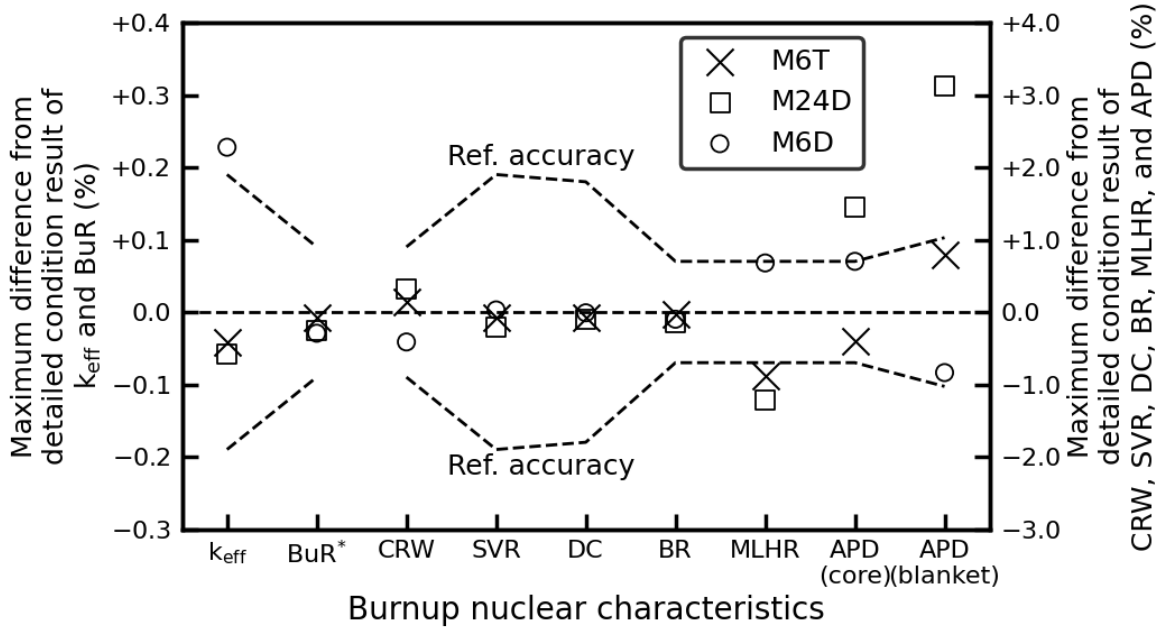
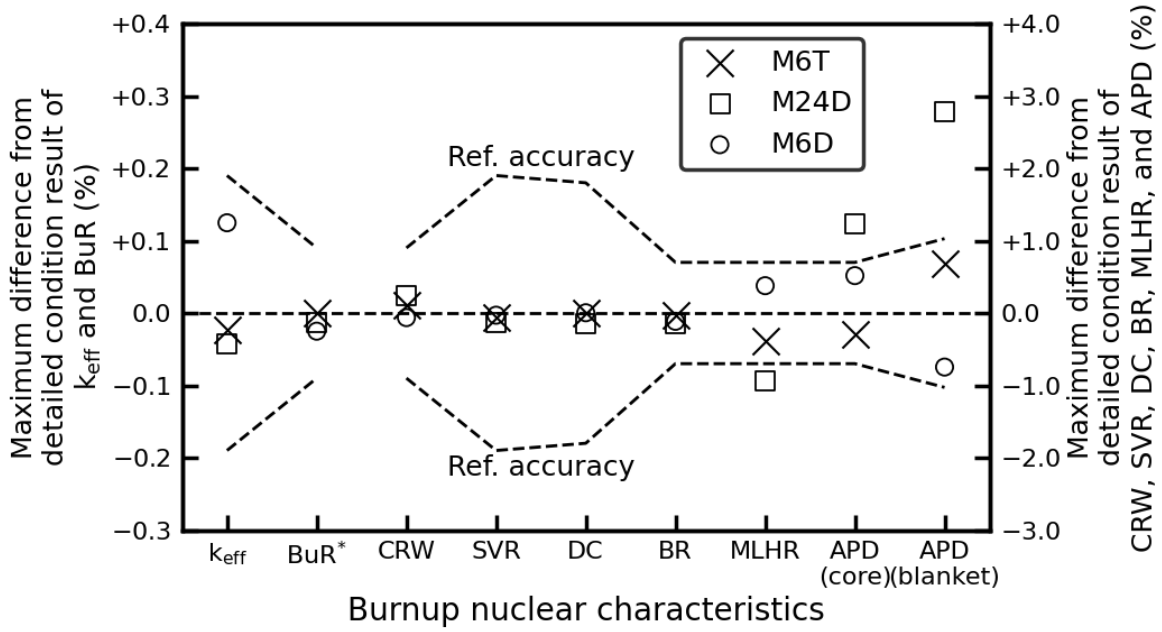


Fig. 3.1-3 Correlation of APD and ^{239}Pu inventory difference from the detailed condition result on blanket assembly (7G)



(a) U-Pu high fissile fuel core



(b) TRU low fissile fuel core

Fig. 3.2-1 Maximum difference of low-cost calculation
(Treatment of spatial mesh and neutron transport)
* The absolute difference in the unit of %Δk/kk'

Table 3.1-1 Maximum snapshot correction factors and maximum snapshot correction difference by changing fuel compositions (Treatment of neutron energy group)

	Maximum snapshot correction factors				Maximum Snapshot correction difference by changing fuel composition				Ref. accuracy
	70G→175G	36G→175G	18G→175G	7G→175G	70G→175G	36G→175G	18G→175G	7G→175G	
k_{eff}	1.0003	1.0002	1.0002	0.9978	0.0003	0.0002	0.0003	0.0020	**0.0019**
CRW	0.9996	0.9929	0.9869	0.9209	0.0006	0.0014	0.0030	0.0145	**0.0090**
SVR	1.0130	—	—	—	0.0013	—	—	—	**0.0190**
DC	1.0187	—	—	—	0.0016	—	—	—	**0.0180**
BR	0.9980	0.9988	0.9996	1.0026	0.0006	0.0005	0.0005	0.0008	**0.0070**
MLHR	0.9984	0.9971	0.9958	0.9910	0.0009	0.0014	0.0014	0.0047	**0.0070**
APD (core)	0.9980	1.0026	1.0050	1.0085	0.0014	0.0016	0.0015	0.0045	**0.0070**
APD (blanket)	1.0035	1.0086	1.0161	1.0328	0.0013	0.0016	0.0017	0.0037	**0.0103**

Table 3.1-2 Computation time and calculation accuracy (Treatment of neutron energy group)

	U-Pu high fissile					TRU low fissile				
	175G	70G	36G	18G	7G	175G	70G	36G	18G	7G
Base calculation [h]	17.0	4.6	3.0	2.7	2.6	13.4	4.0	3.0	2.8	2.6
SVR and DC calculation [h]	3.5	0.4	0.4	0.4	0.4	2.7	0.4	0.4	0.4	0.4
Snapshot correction [h]	—	4.7	4.7	4.6	4.6	—	5.1	5.0	5.0	5.0
Total [h]	20.5	9.8	8.1	7.8	7.6	16.2	9.5	8.4	8.2	8.0
Adequacy of calculation accuracy	OK	OK	OK	OK	NG	OK	OK	OK	OK	NG

Table 3.2-1 Maximum snapshot correction factors and maximum snapshot correction difference by changing fuel compositions (Treatment of spatial mesh and neutron transport)

	Maximum snapshot correction factors			Maximum Snapshot correction difference by changing fuel composition			Ref. accuracy
	M6T→ M24T	M24D→ M24T	M6D→ M24T	M6T→ M24T	M24D→ M24T	M6D→ M24T	
k_{eff}	1.0003	1.0033	0.9977	0.0001	0.0003	0.0002	**0.0019**
CRW	0.9881	0.9876	1.0528	0.0008	0.0010	0.0018	**0.0090**
SVR	0.9985	1.0113	1.0118	0.0007	0.0025	0.0036	**0.0190**
DC	0.9987	0.9927	0.9952	0.0008	0.0018	0.0018	**0.0180**
BR	0.9993	0.9967	0.9967	0.0002	0.0006	0.0008	**0.0070**
MLHR	1.0169	0.9854	1.0222	0.0029	0.0038	0.0020	**0.0070**
APD (core)	1.0066	1.0212	0.9852	0.0028	0.0055	0.0036	**0.0070**
APD (blanket)	0.9914	0.9771	1.0205	0.0021	0.0061	0.0050	**0.0103**

Table 3.2-2 Computation time and calculation accuracy (Treatment of spatial mesh and neutron transport)

	U-Pu high fissile				TRU low fissile			
	M24T	M6T	M24D	M6D	M24T	M6T	M24D	M6D
Base calculation [h]	9.5	3.3	5.1	2.6	9.4	3.3	5.1	2.6
SVR and DC calculation [h]	4.6	1.3	0.9	0.4	4.7	1.3	1.0	0.4
Snapshot correction [h]	—	7.7	7.6	6.7	—	8.6	8.2	7.2
Total [h]	14.1	12.4	13.6	9.7	14.1	13.2	14.2	10.2
Adequacy of calculation accuracy	OK	OK	NG	OK	OK	OK	NG	OK

4. Conclusions

This study investigated the impacts of core burn-up calculation conditions on calculation accuracy and computation time for the next-generation fast reactors. We took two low-cost treatments concerning analysis conditions and the methods for core burn-up calculation for the 750-MWe JSFR for the neutron energy group, and for the spatial mesh and neutron transport.

In terms of the treatment of the neutron energy group, by adopting the cross sections smaller than 70 groups, the computation time of base calculation decreased significantly in comparison with the case adopting the 175-group cross section. This was due to the decrease in the computation time of core burn-up calculation. Adopting an 18-group cross section minimized the total computation time. The snapshot corrections of this case were almost equivalent among any fresh fuel TRU compositions assumed in this study.

In terms of the treatment of spatial mesh and neutron transport, there was little difference in the total computation times for the cases with 24 meshes per assembly regardless of applying the diffusion approximation. In these cases, applying the diffusion approximation worsened accuracy and some burnup nuclear characteristics did not fall within the reference accuracy. However, in the cases with 6 meshes per assembly, the case that applied the diffusion approximation exhibited a shorter computation time and sufficient accuracy. Hence this case allowed for the fastest calculation and gave equivalent results with the referential detailed condition results within the reference accuracy. As for the treatment of the neutron energy group, the snapshot corrections of this case were almost equivalent in any fresh fuel TRU composition assumed in this study.

As a result, from the viewpoints of calculation accuracy and computation time, it has been found that the following analysis conditions are the most preferable for base calculation of burn-up nuclear characteristics among analysis conditions in this study: 18-group structure, 6 meshes per assembly and diffusion approximation. These conditions should be used with the energy group corrections, spatial mesh and transport ones, which are evaluated by the snapshot calculations. These corrections need to be calculated only once with an arbitrary fuel composition and are usable for any fresh fuel TRU composition in the range assumed in this study.

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