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EFFECTS OF THE USED CALCULATION METHOD AND CROSS-SECTION SET UPON SAFETY PHYSICS PARAMETERS OF A GAS-COOLED FAST BREEDER REACTOR

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Effects of the used Calculation Method and Cross-section Set upon Safety Physics Parameters of a Gas-cooled Fast Breeder Reactor

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Effects of the used calculation method and cross section set upon safety phisics parameters of a gas-cooled fast breeder reactor have been studied with the GCFR pin reference core "GBR-4". The methods used to examine nuclear characteristics of the reference core design are ordinary diffusion theory, collision probability, and Monte Carlo, and the cross section sets are JAERI-FAST set-version II and RCBN set-version III, the latest libraries in JAERI. In conclusion, the ordinary diffusion treatment is applicable to a pin-type gas-cooled fast reactor, but cannot predict well the axial streaming of neutrons. There is significant difference in the neutron streaming between the collision probability and the Monte Carlo treatments. Effects of the cross section set used on the reactivity calculated is relatively small, which, however, increases with fuel burnup.

ガス冷却高速増殖炉の安全性に関連した炉物理パラメータに及ぼす異なる計算手法ならびに異なる炉定数セットの影響について

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計算手法の相異，ならびに使用する炉定数セットの相異が，ガス冷却高速炉の炉物理パラメータに如何なる影響を及ぼすかについて，定量的な研究を行なった。対象とした原子炉はピン型燃料を用いる最新の設計である〝GBR－4〟である。又，この研究のために選んだ計算手法は，(1)通常の拡散理論に基ずく手法，(2)衝突確率法に基ずく手法，および(3)モンテカルロ理論に基ずく手法であり，使用した炉定数セットは(1)JAERI－FAST－Vevsion Ⅱ および(2)RCBN－Vevsion Ⅲ であって，いずれも原研に於ける最新のライブラリィである。

結果の概要は以下の通りであった。

- 通常の拡散計算はピン型燃料のガス冷却高速炉に対して充分適用可能である。但し，
- 炉心軸方向の中性子の流れに対しては正確な記述をすることはできないことが判った。
- 中性子の流れについては，衝突確率法による計算とモンテカルロ法による計算の結果の間にも，無視出来ない程度の不一致がみられた。
- 炉定数セットの相異が初期臨界時の反応度計算に及ぼす影響は僅かであった。但し，燃焼特性に及ぼす影響は僅かとはいえず，核燃料物質の質量変化，反応度変化に対する両セットによる結果の不一致は燃焼の進行と共に拡大されて行くことが判った。

Contents

1. INTRODUCTION

Physics parameters of a gas-cooled fast breeder reactor have been calculated using different calculation methods and different cross-section sets. The purpose of this investigation is to obtain more accurate information about suitable calculation methods and cross-section sets to be applied to designing of GCFR.

For this purpose the present GCFR pin reference core "GBR-4" was selected as our reference GCFR design.

The methods employed to investigate characteristics of criticality of the reference design are as follows;

-the first is an ordinary diffusion calculation method. The diffusion code "CITATION" was used for this purpose using the latest two nuclear data sets.

-the second is the collision probability method. The code "PIGEON" [1] together with the anisotropic diffusion code, MOD-CITAION, was used for this calculation. PIGEON calculates the effective group constants including anisotropic diffusion coefficients considering the heterogeneity of cells in a fast reactor by collision probability method. MOD-CITAION had been developed to use these effective group constants.

-the third is the Monte Carlo calculation method, which is quite suitable to investigate effects of neutron streaming on physics characteristics of reactors, especially of GCFR. The Monte Carlo code, "TIMOC-J" [2], had been developed basing on TIMOC [3], and used for this calculation.

The results of these different calculations were compared and discussed. The detail of the results will be described in the Section 3.

The cross-section sets employed in this investigation are the latest two libraries, both of which were produced in JAERI. The one is JAERI-FAST set-version II [4], and the other is RCBN-version III. The latter was produced from the ENDF/B file, Version III.

Nuclear safety parameters such as Doppler coefficient, thermal expansion coefficient, void coefficient and the kinetics parameters such as prompt neutron life time and effective delayed neutron fraction, have been calculated using above two different libraries. The void coefficient in a loss of coolant accident was evaluated taking heterogeneity effect into consideration.

Burnup characteristics have been evaluated using the two-dimensional multigroup diffusion burnup code, "APOLLO" [5].

2. MAIN DESIGN PARAMETERS AND CALCULATION PROCEDURES

Most of the design parameters of our reference design was taken from the °GBR-4 SAFETY WORKING DOCUMENT [Brussels, July 1974], and listed in Table 2.1.

The first nuclear calculation that we had carried out was to determine the nuclide concentrations of three enrichment core zones. The calculation procedures were as follows;

-with the given geometrical data, volume fractions and selected materials, plutonium enrichment searches were performed in one-dimensional geometry, using the one dimensional diffusion theory code "ANDROMEDA" [6] in R-Z geometry with 25 energy groups. The target of this calculation was to obtain a rough estimation of the ratio of plutonium enrichments in the three core zones, which could give the nearly same maximum power densities in each zones.

-the second step was to calculate in more detail the ratio of plutonium enrichments in two dimensional R-Z geometry, in order to ensure flattening of the power distribution. This calculation was done with the two dimensional diffusion theory burnup code, "APOLLO". For the two-dimensional calculations the 25 energy groups given by the data libraries were collapsed into six groups.

-the final step was two dimensional burnup calculations which gave the necessary burnup excess reactivities for a chosen refuelling scheme. Iterations between a zone averaged plutonium enrichment and the target eigenvalue were carried out until convergence was reached. The target eigenvalue was so selected as to maintain the reactor to be still critical at the end of equilibrium cycle.

-from the resulting region average plutonium enrichment the final nuclide concentrations in each zone were obtained and used for the succeeding calculations.

The two dimensional R-Z calculation model is illustrated by Fig. 2.1. The reactor is composed of 256 core fuel assemblies, 13 control rod assemblies and 126 radial blanket fuel assemblies. The blankets are surrounded by steel reflectors.

For nuclear calculations, some simplifications were made for the calculation model:

-control rod absorbers were eliminated.

-the space of a control rod assembly within the active core zone was occupied only by the rod-follower material.

-the space of a control rod assembly within the axial blanket zone was occupied only by the alumina reflector material. This assumption might overestimate the neutron scattering effect in these area in comparison with that in actual situation.

Table 2.1

Main Parameters for Reference Design

Parameter		
Core and blanket geometry		
(1) Core and axial blanket		
Fuel pin diameter ;		
-clad outside	mm	7.7
-clad inside	mm	7.0
Fuel pin pitch	mm	11.65
Number of fuel pins/assembly		328
Core height	mm	1400
Axial blanket height upper	mm	600
lower	mm	600
Assembly pitch	mm	226
Number of fuel assemblies		252
Number of control assemblies		13
Core equivalent diameter	mm	3863
Core volume	m^3	16.41
Core volume fractions ;		
(all control rods are pull out)		
-fuel	%	27.3
-clad	%	6.0
-structure	%	5.9
-coolant	%	60.8
(2) Radial blanket		
Fuel pin diameter ;		
-clad outside	mm	12.3
-clad inside	mm	11.4
Number of pins/assembly		217

Table 2.1 (continued)

Parameter		
Number of assemblies		126
Equivalent outer diameter	mm	4693
Blanket volume fractions ;		
-fuel	%	52.0
-clad	%	10.3
-structure	%	5.3
-coolant	%	32.4
<u>Fuel and coolant element data</u>		
(1) Core fuel (fresh fuel)		
material		$(U,Pu)O_2$
smeared density	% T.D.	85
O/M ratio		1.98
enrich ment (PuO_2 / $(U,Pu)O_2$) ;		
-first zone	w/o	14.8
-second zone	w/o	19.1
-third zone	w/o	23.1
isotope ratio	a/o	
Pu239/Pu240/Pu241/Pu242 ; 65./20./13./2.		
U235/U238 ;0.72/99.28		
effective density (at 260°C)	g/cm^3	9.3
(2) Blanket fuel (fresh fuel)		
material		UO_2
smeared density	% T.D.	93.0
O/M ratio		1.98
isotope ratio	a/o	
U235/U238 ; 0.72/99.28		
effective density (at 260°C)	g/cm^3	10.14
(3) Clad		
material		SUS-316L
element ratio	w/o	
Cr/Mn/Fe/Ni/Mo ; 18.0/2.0/67.5/10.0/2.5		
density (at 260°C)	g/cm^3	7.87

Table 2.1 (continued)

Parameter		
(4) Structure		
material		SUS-405
element ratio	w/o	
Cr/Mn/Fe ; 13.0/1.0/86.0		
density	g/cm^3	7.87
(5) Coolant		
material		He
pressure	Kg/cm^3	90
specific volume	cm^3/g	128.9
specific weight	g/cm^3	7.76×10^{-3}

Volume ratio of the assemblies

(1) Core fuel assembly

fuel/clad/structure/coolant (%) ; 28.71/6.31/5.94/59.04

(2) Blanket fuel assembly

fuel/clad/structure/coolant (%) ;

-axial 28.71/6.31/5.94/59.04

-radial 52.0 /10.3/5.3 /32.4

(3) Control region (follower)

structure/coolant (%) ; 5.07 /94.93

(4) Reflector

structure/coolant (%) ;

-axial 60.0/40.0

-radial 95.0/5.0

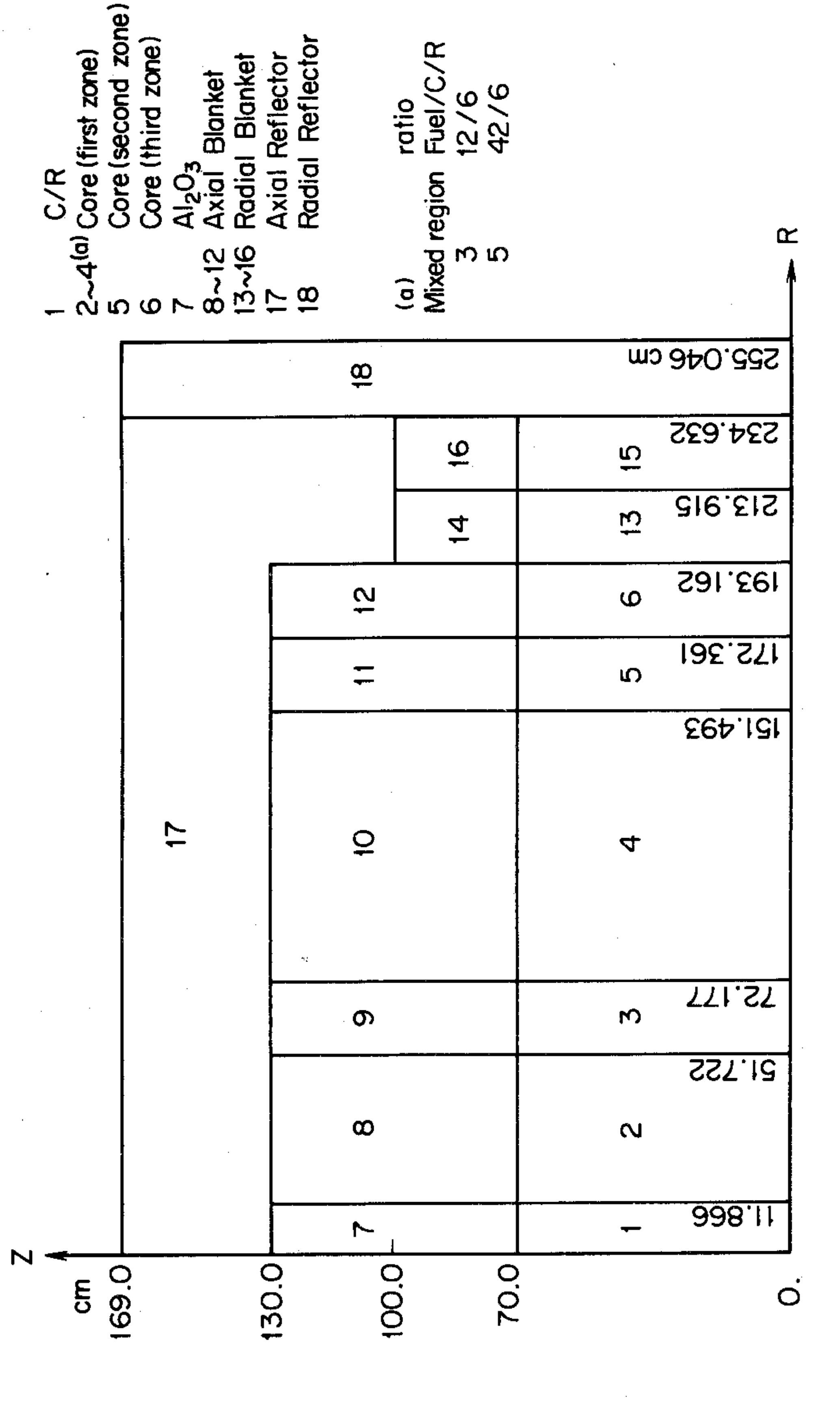

Fig. 2.1 R－Z Calculation Model

3. CRITICALITY CHARACTERISTICS

3.1 Ordinary diffusion treatment

The calculation procedure in the ordinary diffusion treatment was straightforward. It was identical with that used for LMFBR's. Criticality calculations were done with the two dimensional diffusion theory code "CITATION", using the JAERI-FAST set and RCBN set, of 25 energy groups. Both sets have the same energy group structure. The calculation model employed is a R-Z geometry, as shown in Fig. 2.1.

Comparison of the calculated results with these different cross-section sets are given in Table 3.1. Neutron spectra averaged over the core excluding the central control rod region is illustrated in Fig. 3.1.

From the table, it can be concluded that JAERI-FAST set estimates the criticality of the reactor slightly less reactive than RCBN set does. As shown later, this seemed to be brought from the larger absorption cross-section and the smaller fission cross-section of heavy nuclides of JAERI-FAST set in comparison with those of RCBN set.

The figure of the neutron spectra represents good agreement in general, but also represents relatively large discripancies in the two energy levels around the peak of the neutron spectra. The reason of these discripancies are now under investigation.

The result of this ordinary diffusion calculation will be discussed in detail in Section 3.3, comparing with the result of the Monte Carlo calculations.

3.2 Collision Probability Treatment

It has been pointed out that the axial leakage of neutrons was underestimated if the ordinary diffusion treatment was applied for gas-cooled fast reactor.

In order to solve this problem, we had developed the code "PIGEON". It is programmed to be able to calculate the effective group constants including anisotropic diffusion coefficients of a unit cell in the fuel or the control rod assembly of a fast reactor, by using the collision probability method. The anisotropic diffusion coefficients is calculated basing on the Benoist's formula.

Heterogeneity effects of fuel pins of the reference design were evaluated by this code. Criticality calculations were performed in two imensional R-Z geometry and 25 energy groups, using the diffusion theory code "Mod. CITATION", which had been produced from the original CITATION to use the Benoist's anisotropic diffusion coefficients.

As results of direct diffusion calculations using homogeneous and heterogeneous cross-sections, the heterogeneity effect including the anisotropic neutron diffusion effect was -0.47 %Δk/k.

The heterogeneity effects on neutron balance are listed below:

	Homogeneous	Heterogeneous	Difference
k_{eff}	1.0622	1.0572	-0.47%
Production	1.0	1.0	
Absorption	0.8966	0.8978	+0.13%
Radial Leakage	0.0139	0.0143	+2.88%
Axial Leakage	0.0309	0.0338	+9.39%

This result shows the axial streaming of neutrons is dominant in the heterogeneity effect. The other interesting result is that the radial leakage of neutrons is also underestimated by about 3 % in the ordinary diffusion treatment.

However, it must be said that the heterogeneity effects in the control rod followers and the alumina reflectors have not been taken into account, since these zones were treated as homogeneous regions. The axial streaming of the neutrons in the overall reactor will, therefore, become much more, and consequently the heterogeneity effect will also become large.

3.3 Monte Carlo Treatment and Discussion

The method most widely used to calculate nuclear characteristics of a gas cooled fast reactor may be based on diffusion theory approximation. It sometimes gives, however, a poor-approximated solution for a system of a complicated geometry. For such a system, the approximation could be improved by using a detailed three dimensional, multigroup diffusion theory treatment with some correction calculations as those discribed in the previous section. It is, however, quite difficult to perform such vast calculations even with the current large capacity, high speed computers.

On the other hand, the current Monte Carlo code can treat complicated general geometries with fairly short computer time. Then we tried to apply the Monte Carlo method to the GBR-4 system, in order to check the accuracy of the results obtained by the diffusion theory method previously described.

Fig. 3.2 shows a calculation model of the GBR-4 system. The neutron multiplication factor, the mean generation time, the stationary energy- and region-dependent fluxes, and the transmission ratios among geometrical regions have been calculated by the TIMOC-J, Monte Carlo neutron transport code. Although the generality of the geometry routine in this code allows an exact specification for the geometry of the GBR-4 system, we simplified it for saving the computer time.

Each of 13 control rod channels in the core was treated as a separated region as shown in Fig. 3.2. All of the control rods, the positions of which were slightly rotated for symmetry, were assumed to be withdrawn from the core. The outer boundaries of the core zone 1, 2 and 3 and the blanket were modified to have cylindrical surfaces conserving the volume of each zone. All of reflector regions were omitted.

The nuclear cross-sections employed were the group averaged cross-sections contained in the JAERI-FAST set with 25 energy groups. Results of the Monte Carlo calculations and the comparison with those of the diffusion theory method are given in Table 3.2.

The neutron multiplication factor obtained by the diffusion method well agrees with the result of the Monte Carlo method. The prompt neutron life time, however, is 13 % less than the generation time, which coincides with the prompt neutron life time if the multiplication factor closes to unity. The figure in the table for the life time was obtained with k_{eff} = 1.0711 by a one-dimensional diffusion code, ANDROMEDA, with a cylindrical geometry using the axial buckling obtained by the two-dimensional (r,z) diffusion calculation. This discrepancy seems to occur because of the inaccurate treatment of the geometry in the one-dimensional diffusion code.

The total leakage rate by the diffusion method is 28 % higher than that by the Monte Carlo method. It must be noted that the leakage from the core to the axial blanket through the central control rod channel is 2.7 times larger than the core averaged leakage per unit assembly in case of the diffusion calculation, while that is only 1.6 times larger in case of the Monte Carlo calculation. This shows that the diffusion coefficient of the control rod region might be overestimated in the diffusion method.

The diffusion theory estimated a little higher power ratio in the core zone 1, and a little lower power ratios in the core zone 2 and 3, in comparison with those by the Monte Carlo method, but the differences were at most 3.4 %.

Region averaged neutron spectra in the core and in the axial blanket are shown in Fig. 3.3 and Fig. 3.4, respectively. It seems that the discrepancy in the spectra between the Monte Carlo method and the diffusion method occurs from the different treatment on the elastic scattering.

Fig. 3.5 shows the production and the destruction time distributions which can be used in the safety consideration about the reactor kinetics.

3.4 Discussion on Heterogeneity and Streaming Effects

Heterogeneity and streaming effects of the system having the heterogeneous pin geometry have been studied by the Monte Carlo method.

To evaluate these effects, we chose a typical hexagonal unit cell of a fuel pin which is axially composed of three parts, i.e., the top and the bottom axial blankets and the core fuel. The Monte Carlo calculations were performed on the unit cell under the axial symmetry condition with the boundary conditions of the perfect reflections at all of the cell boundaries except the top boundary. Similar calculations were also carried out on the radially homogenized unit cell in order to get information about the heterogeneity effects. The results of both calculations are given in Table 3.3.

Although the number of the secondary histories computed for the heterogeneous system is not sufficiently large to make a precise comparison between the heterogeneous system and the homogeneous system, this table shows several interesting features; that is, the homogeneous calculation seems

- to underestimate the neutron transmission between the core and the blanket,
- to underestimate the neutron leakages out of the core and out of the blanket, and
- to overestimate the volume- and energy-integrated flux, the power generation and the absorption in the core, or conversely to underestimate them in the axial blanket.

It may be concluded that these tendencies were induced mainly by the difference in the estimation of the neutron streaming effect; that is, in case of the heterogeneous system, neutrons in the core zone can easily be transmitted into the axial blanket through the coolant area occupied only by helium gas, while in case of homogeneous system their transmission is interrupted by the fuel or structure materials distributed everwhere in the unit cell.

As the conclusion in this section we can say the followings. The diffusion theory method is a sufficiently appropriate approximation to solve the Boltzmann equation for gas cooled fast reactors such as the GBR-4. For the detailed design study, however, some improvements on the diffusion coefficients and the modelling of the systems are required. The nuclear characteristics in the blanket would have an error of about 10 % due to the streaming effect if the ordinary diffusion method was used. This was deduced from the rough comparison described above, but will be the subject for a future study.

Table 3.1

COMPARISON OF NUCLEAR CHARACTERISTICS OBTAINED BY THE ORDINARY DIFFUSION METHOD USING DIFFERENT CROSS-SECTION SETS

	JAERI-FAST	RCBN	DIFFERENCE
k_{eff}	1.05758	1.06111	- 0.00353
NEUTRON BALANCE			
IN THE REACTOR			
SOURCE	0.94555	0.94232	+ 0.00323
ABSORPTION	0.92459	0.92465	- 0.00006
LEAKAGE	0.02096	0.01767	+ 0.00329
AXIAL	0.01117	0.00927	+ 0.00190
RADIAL	0.00979	0.00840	+ 0.00139
IN THE CORE			
SOURCE	0.86953	0.86501	+ 0.00452
ABSORPTION	0.66741	0.66115	+ 0.00626
LEAKAGE	0.20212	0.20386	- 0.00174
AXIAL	0.12008	0.12125	- 0.00117
RADIAL	0.08204	0.08261	- 0.00057

(Reactor Temparature; 600°K)

Table 3.2

COMPARISON OF NUCLEAR CHARACTERISTICS OBTAINED BY
THE MONTE CARLO AND THE DIFFUSION METHOD CALCULATIONS

	MONTE CARLO	DIFFUSION
(CALCULATED SECONDARY HISTORIES	116,000)	
NEUTRON MULTIPLICATION FACTOR	1.0533 $\pm$ .0018	1.0545
MEAN GENERATION TIME (MONTE CARLO) OR PROMPT NEUTRON LIFE TIME (DIFFUSION), (10^{-7}sec)	5.978 $\pm$ .037	5.23(*)
TOTAL LEAKAGE	.04900 $\pm$ .00054	.06267
CORE TO AXIAL BLANKET		
AT THE CENTRAL C/R CHANNEL	.00078	.00129
IN ALL CORE ZONES CONTAINING C/R	.1284	.1254
PER UNIT ASSEMBLY	.00049	.00048
LEAKAGE FROM AXIAL BLANKET		
AT THE CENTRAL C/R CHANNEL	.00008	.00018
IN THE OTHER CHANNELS	.0286	.0365
PER UNIT ASSEMBLY	.00011	.00014
LEAKAGE FROM RADIAL BLANKET	.0203	.0262
POWER RATIOS (%)		
CORE ZONE 1	59.3	60.5
ZONE 2	15.5	15.1
ZONE 3	18.1	17.5
RADIAL BLANKET	3.4	3.3
AXIAL BLANKET	3.6	3.6

(*) One-dimensional diffusion code, ANDROMEDA, was used.

Table 3.3

NUCLEAR CHARACTERISTICS OBTAINED BY THE MONTE CARLO METHOD IN THE CELL OF THE CORE ZONE 1

	HETEROGENEOUS	HOMOGENEOUS
(CALCULATED SECONDARY HISTORIES	8,000	72,000)
NEUTRON MULTIPLICATION FACTOR	1.146 ± .007	1.156 ± .002
MEAN GENERATION TIME (10^{-7}sec)	5.06 ± .12	5.20 ± .04
MEAN DESTRUCTION TIME(10^{-7}sec)	8.47 ± .13	8.68 ± .05
TRANSMISSION FROM CORE TO BLANKET	.533	.505
FROM BLANKET TO CORE	.393	.374
LEAKAGE FROM CORE	.140	.130
FROM BLANKET	.0396± .0019	.0302± .0005
MEAN NUMBER OF SCATTERING EVENTS	40.0 ± .2	40.11 ± .08
POWER RATIOS (%) CORE	95.4	95.9
BLANKET	4.6	4.1
FLUX INTEGRATED IN CORE	210.7	214.7
IN BLANKET	47.4	43.9
ABSORPTION CORE	.815	.834
BLANKET	.147	.138
TOTAL	.962 ± .008	.972 ± .003

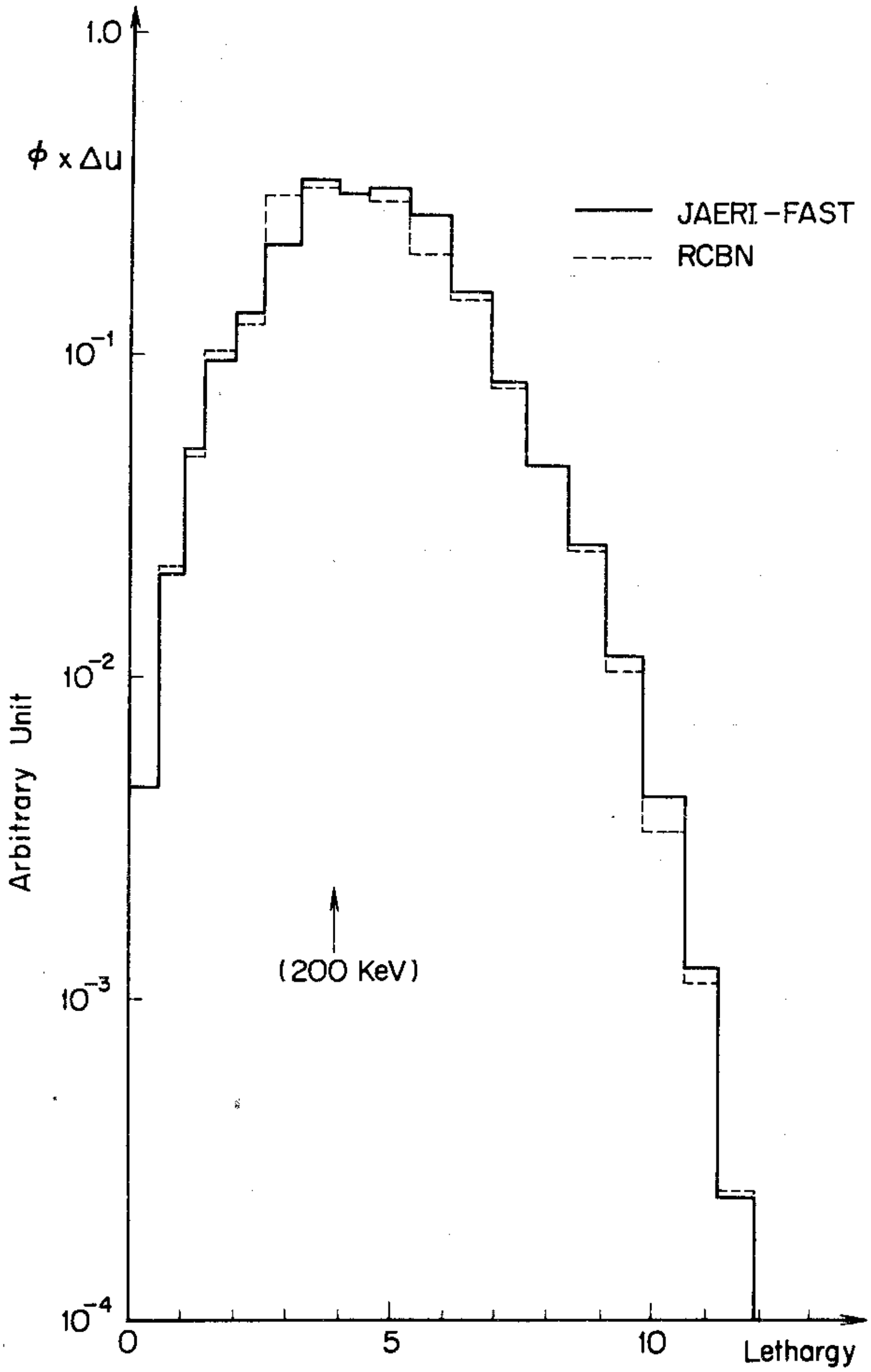

Fig. 3.1 Neutron Spectra (Core Average)

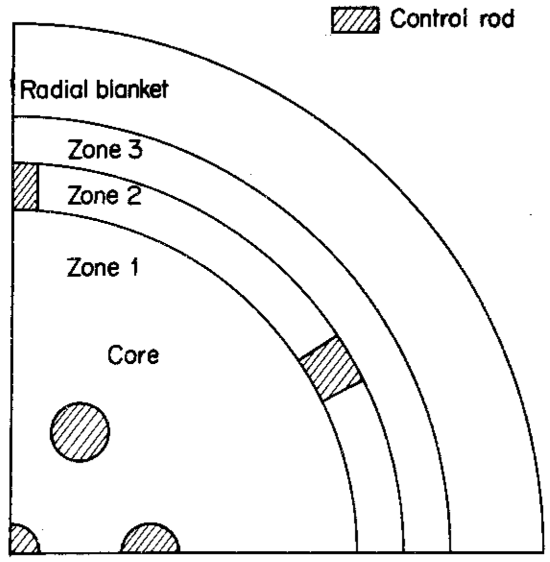

Fig. 3.2 Horizontal Section, Used in the Monte Carlo Calculations

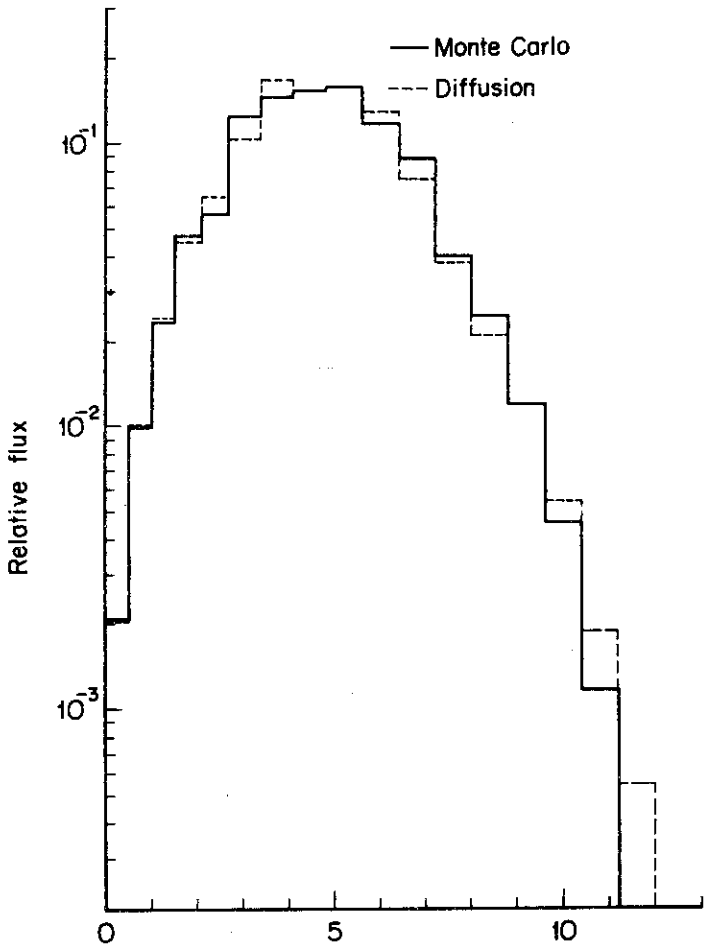

Fig. 3.3 Core Average Spectra, $\phi(E) \cdot \Delta E$

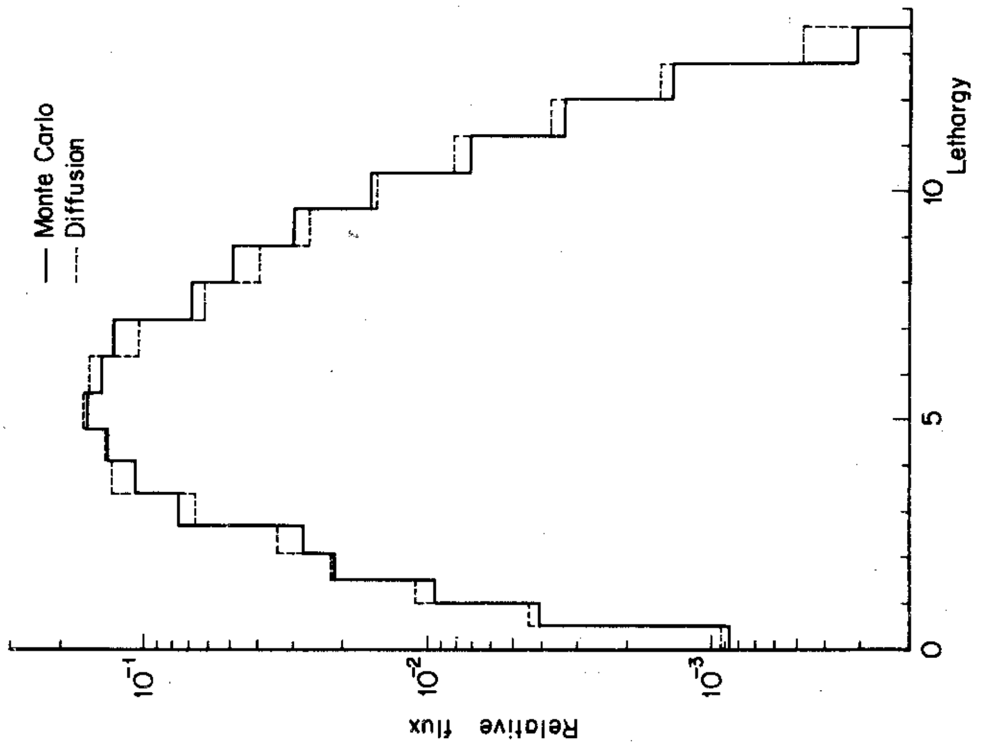

Fig.3.4 Axial Blanket Average Spectra, $\phi(E)\cdot\Delta E$

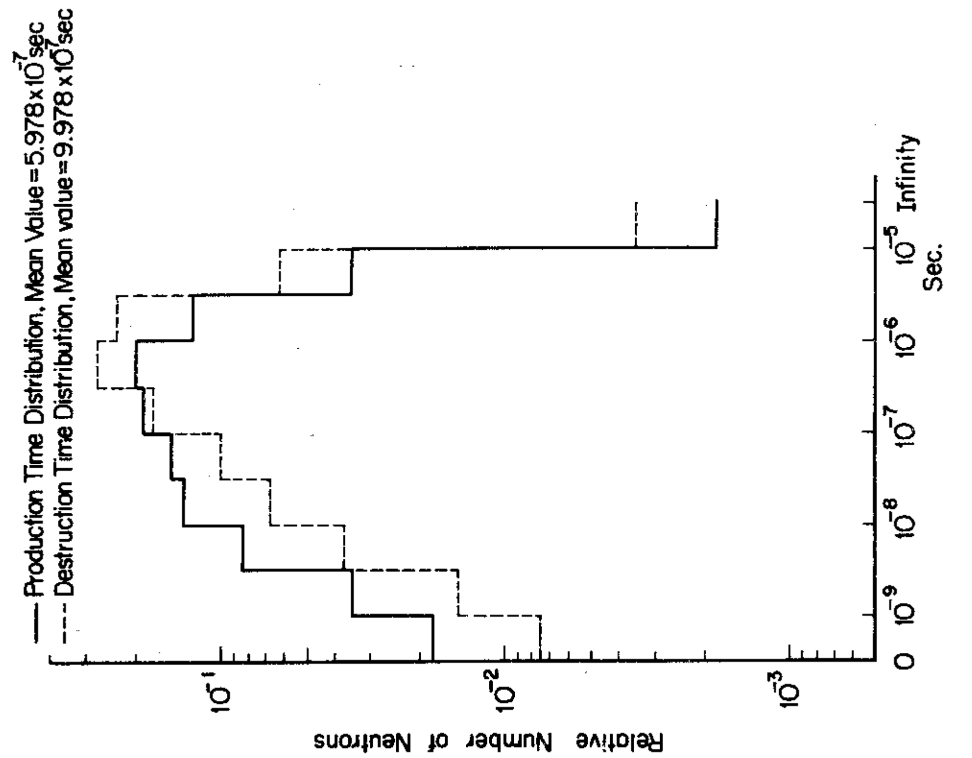

Fig. 3.5 Production and Destruction Time Distributions, Pt or Dt(T)ΔT

4. NUCLEAR SAFETY PARAMETERS

4.1 Doppler Coefficient

Two-dimensional R-Z diffusion calculations have been performed with JAERI-FAST and RCBN sets for four temperatures, i.e., 300°K, 600°K, 1100°K and 1500°K to get Doppler coefficient.

The temperature dependence in reactivity is shown in Fig. 4.1, and the numerical results are shown in Table 4.1. RCBN set gives higher Doppler coefficients in all the temperature range than JAERI-FAST set does. The discrepancy is about 10 % at the reactor operation temperature (∿1100°K), and decreases with increasing the reactor temperature.

4.2 Thermal Expansion Coefficient

Two dimensional R-Z diffusion calculations have been performed with different core heights which simulate axial core expansion. Assumptions of the analysis are as follows:

- only the axial expansion has been considered and the no radial expansion;
- the axial expansion for the reactor were 1 % and 5 % ($\Delta L/L$);
- the reactor temperature was fixed on 600°K;
- changing the atomic densities have been done to keep the fuel and structure masses constant.

The results are listed in Table 4.2. As shown in this table, RCBN set estimates higher expansion coefficients than JAERI-FAST set does, but the discrepancies are considered to be small.

4.3 Void Coefficient

The loss of coolant reactivity was evaluated by voiding the reactor model entirely and recomputing the eigen-value. The calculations were done in two dimensional cylinder geometry and 25 energy groups. The heterogeneity effects of fuel pin have been taken into account.

The total coolant void effect on reactivity for the design considered is 0.26 % $\Delta K/K$. The void effect on neutron balance is listed below;

	Normal	Void	Difference
k_{eff}	1.0572	1.0599	+0.26 %
Production	1.0	1.0	
Absorption	0.8978	0.8929	-0.55 %
Radial Leakage	0.0143	0.0147	+2.80 %
Axial Leakage	0.0338	0.0359	+6.21 %

It can be seen from the above table that the positive void effect is dominated by the reduced absorption in the voided core.

4.4 Kinetic Parameters

The effective delayed neutron fraction and the prompt neutron life time of the design considered were calculated by ANDROMEDA code with one-dimensional cylinder model and 25 energy groups. Two different cross-section sets, JAERI-FAST and RCBN, were used. The resulting values are listed below:

	JAERI-FAST	RCBN
Prompt Neutron Life Time (10^{-7} s)	5.2	4.9
Effective Delayed Neutron Fraction (%)	0.39	0.39

Table 4.1 COMPARISON OF DOPPLER COEFFICIENTS CALCULATED WITH JAERI-FAST AND RCBN SETS

Doppler Coefficients

TEMPERATURE CHANGE °K	% Δk/k		Δk/k/°C × 10^6		
	JAERI-FAST	RCBN	JAERI-FAST	RCBN	DIFFERENCE
300 - 600	-0.550	-0.623	-18.3	-20.8	-12.0 %
600 - 1100	-0.515	-0.576	-10.3	-11.5	-10.4 %
1100 - 1500	-0.246	-0.265	- 6.16	- 6.63	- 7.1 %

Neutron Balance in the Core

	JAERI-FAST			RCBN		
TEMPERATURE	300°K	600°K	1500°K	300°K	600°K	1500°K
SOURCE	0.8634	0.8695	0.8782	0.8580	0.8651	0.8746
ABSORPTION	0.6632	0.6674	0.6734	0.6563	0.6612	0.6677
LEAKAGE	0.2002	0.2021	0.2048	0.2017	0.2039	0.2069

Table 4.2 COMPARISON OF AXIAL EXPANSION EFFECTS CALCULATED WITH JAERI-FAST AND RCBN SETS

Reactivity Change

AXIAL EXPANSION ΔL/L	Δk/k (%)		DIFFERENCE
	JAERI-FAST	RCBN	
1 %	-0.107	-0.117	-8.5 %
5 %	-0.595	-0.610	-2.5 %

Neutron Balance in the Core Zone

ΔL/L	JAERI-FAST			RCBN		
	0 %	1 %	5 %	0 %	1 %	5 %
SOURCE	0.8695	0.8702	0.8731	0.8651	0.8657	0.8686
ABSORPTION	0.6674	0.6672	0.6660	0.6612	0.6609	0.6596
LEAKAGE	0.2021	0.2030	0.2071	0.2039	0.2048	0.2090

(Reactor Temperature; 600°K)

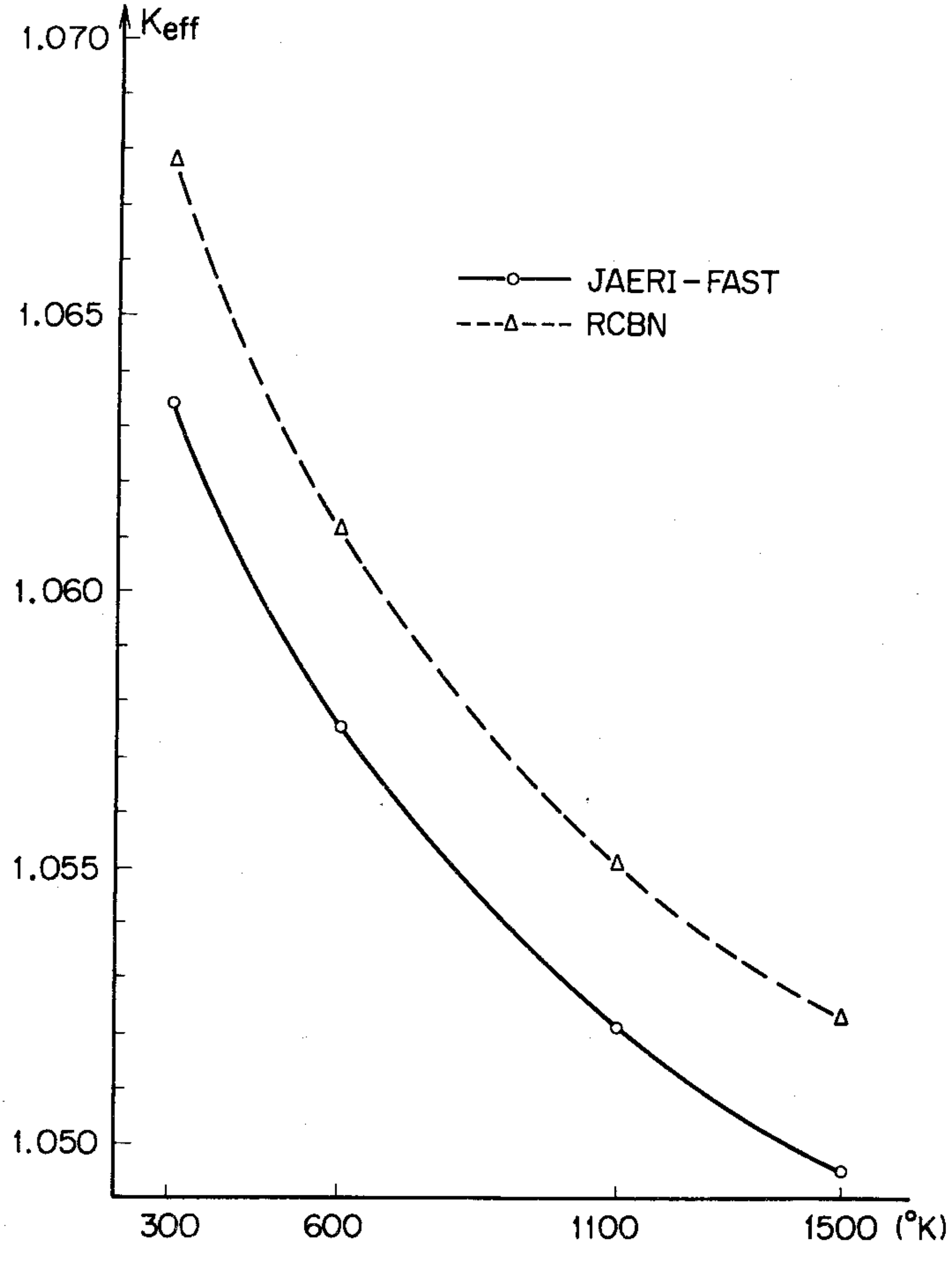

Fig. 4.1 Doppler Effect

5. BURNUP CHARACTERISTICS

5.1 Reactivity Change

The purpose of burnup calculations in our present study was to see effect of different cross-section sets on burnup characteristics. Then a simple calculation model was chosen, as illustrated in Fig. 5.1. In this 'Burnup Model-1', both the axial reflector and the radial reflector were eliminated in order to save computer time.

The reactivity changes from the beginning of the life (BQL) to the end of the equilibrium cycle 4 (EOC-4) are illustrated in Figures 5.2 - 5.4. In these figures, reactivity swings due to the radial blanket refuelling are also illustrated.

These figures indicate that the tendencies of reactivity changes given by the both sets are similar, although the discrepancies in k_{eff} gradually increase with the burnup progression. These discrepancies are enlarged after the radial blankets were entirely refuelled.

By the way, as indicated in these figures, the multiplication constant at the beginning of life of the reference design is tremendously large, and also at the end of equilibrium cycle-1, k_{eff} exceeds 1.02 in case of JAERI-FAST set. These higher effective multiplications lead to the lower breeding characteristics.

The causes of these too higher multiplication constant were as follows;

-the fuel volume fraction in the core, we used, was taken from the "GBR 4 SAFETY WORKING DOCUMENT", but adjusted for the 13 control rod followers. It was made clear later that this adjustment was not necessary.

-Uranium in the core and the blanket was selected as natural one. Then the inventory of Uranium 235 became too large.

In order to correct these points, we selected another calculation model, that is, BURNUP-MODEL-2. In this model, the following data were chosen;

* Volume fraction of the core fuel assembly:

fuel	27.3 %
cladding	6.0 %
structure	5.9 %
coolant	60.8 %

* Isotope ratio in Uranium of the fresh fuel:

	U-235	U-238
core	0.1 %	99.9 %
blanket	0.2 %	99.8 %

* R-Z calculation model:

Steel reflectors were considered as shown in Fig. 2.1.

With the modified calculation model, a part of burnup calculation has been carried out again. The reactivity change is shown in Fig. 5.5, and in Table 5.1. In this case, k_{eff} at the end of equilibrium cycle became less than unity, although at the beginning of life k_{eff} was about 1.048 in case in JAERI-FAST set. This gives the large reactivity swing in one cycle, i.e., about 3 %ΔK/K/cycle.

5.2 Power Swing

Specific power distributions obtained by JAERI-FAST set are illustrated in Fig. 5.6 and Fig. 5.7, and the average specific powers in each zone are listed in Table 5.2. As indicated in this table, the difference between the two sets was quite small in the power distribution.

Tables 5.3 and 5.4 show the power fraction changes.

5.3 Mass Balance

Table 5.5 and Table 5.6 show the initial core inventory, and Table 5.7 and Table 5.8 indicate the mass balance in one cycle.

As shown in Table 5.8, differences between two sets are relatively large from the view point of the mass change. These differences bring the difference in the breeding ratio as indicated in Table 5.9. In general, the breeding ratios are estimated higher by about 2 % in RCBN set.

Table 5.1

REACTIVITY SWING
(BURNUP MODEL-2)

	FULL POWER YEARS	% Δk/k/CYCLE		$/CYCLE	
		JFS	RCBN	JFS	RCBN
BOL → 1 Yr	0 → 1	3.25	2.76	8.3	7.1
	1* → 2	2.83	2.37	7.3	6.1
BOC → EOC	2* → 3	2.68	2.25	6.9	5.8

* Core is refuelled.

Table 5.2

AVERAGE SPECIFIC POWERS (BURNUP MODEL-1)

		CORE									BLANKET	
		ZONE 1			ZONE 2			ZONE 3			RAD.	AX.
	BURNUP YEARS AT THE BEGINNING STAGE	0	1	2	0	1	2	0	1	2		
JAERI FAST SET	BOL		240			217			194		9	7
	AFTER BURNUP (1 Yr.)		235			209			188		13	11
	BOC-1	243	231	221	221	207	194	201	187	176	16	11
	EOC-1	236	225	216	212	199	188	194	182	172	20	14
	RELOAD OF R.B.	250	238	227	225	210	198	201	187	176	9	11
	AFTER BURNUP (1 Yr.)	244	233	224	214	201	190	192	180	170	13	15
RCBN SET	BOL		240			216			194		9	8
	AFTER BURNUP (1Yr.)		235			208			187		13	10
	BOC-1	243	231	221	219	205	193	199	186	175	16	11
	EOC-1	236	226	217	210	197	187	192	180	170	20	15
	RELOAD OF R.B.	250	238	228	224	210	197	200	187	175	9	11
	AFTER BURNUP (1 Yr.)	244	234	224	212	199	187	190	178	168	13	15

Table 5.3

POWER DISTRIBUTION (JAERI-FAST SET)
(BURNUP MODEL-1)

	FULL POWER YEARS	POWER FRACTION (%)						POWER PEAKING FACTOR IN CORE	
		CORE		RADIAL BLANKET		AXIAL BLANKET			
BOL	0	94.7		2.7		2.6		1.38	
	1	92.2		3.9		3.9		1.37	
	1*	92.7		3.8		3.5		1.42	
	2	90.2		5.0		4.8		1.41	
BOC-1	2*	91.3		4.9		3.8		1.36	
EOC-1	3	88.7		6.1		5.2		1.36	
BOC-2	3*	90.3	93.3#	5.9	2.8#	3.8	3.9#	1.42	1.44#
EOC-2	4	87.8	90.7	7.1	4.0	5.1	5.3	1.41	1.45
BOC-3	4*	89.4	92.2	6.8	3.8	3.8	4.0	1.40	1.45
EOC-3	5	86.9	89.6	8.1	5.1	5.0	5.3	1.39	1.44

* Core is refuelled.
# Radial blanket is entirely refuelled.

Table 5.4

POWER DISTRIBUTION (JAERI-FAST SET)
(BURNUP MODEL-2)

	FULL POWER YEARS	POWER FRACTION (%)			POWER PEAKING FACTOR IN CORE
		CORE	RADIAL BLANKET	AXIAL BLANKET	
BOL	0	96.2	1.9	1.9	1.43
	1	92.3	3.5	4.2	1.42
	1*	93.1	3.5	3.4	1.46
	2	89.3	5.1	5.6	1.44
BOC-1	2*	90.9	5.0	4.1	1.39
EOC-1	3	87.2	6.6	6.2	1.37

* Core is refuelled.

Table 5.5

IN-PILE HEAVY METAL INVENTORY (BOL)
(BURNUP MODEL-1)

(kg)

ISOTOPE	CORE	RAD. BLANKET	AX. BLANKET	TOTAL
Pu-239	4,144	0	0	4,144
Pu-240	1,280	0	0	1,280
Pu-241	836	0	0	836
Pu-242	129	0	0	129
U -235	217	369	244	830
U -238	30,314	51,500	34,121	115,935
Pu-fissile	4,980	0	0	4,980
Pu-fertile	1,410	0	0	1,410
Pu-total	6,390	0	0	6,390
U -total	30,531	51,869	34,365	116,765
Fissile	5,197	369	244	5,810
Fertile	31,723	51,500	34,121	117,345
Heavy M.	36,920	51,869	34,365	123,154

Table 5.6

IN-PILE HEAVY METAL INVENTORY (BOL)
(BURNUP MODEL-2)

(kg)

ISOTOPE	CORE	RAD. BLANKET	AX. BLANKET	TOTAL
Pu-239	3,887	0	0	3,887
Pu-240	1,201	0	0	1,201
Pu-241	784	0	0	784
Pu-242	121	0	0	121
U -235	29	102	65	196
U -238	29,082	51,767	32,613	113,462
Pu-fissile	4,671	0	0	4,671
Pu-fertile	1,322	0	0	1,322
Pu-total	5,993	0	0	5,993
U -total	29,111	51,869	32,678	113,658
Fissile	4,699	102	65	4,866
Fertile	30,404	51,767	32,613	114,784
Heavy M.	35,103	51,869	32,678	119,651

Table 5.7

MASS BALANCE IN ONE CYCLE

(BOL ⟶ 1 FULL POWER YEAR, BURNUP MODEL-1)

JAERI-FAST SET (kg)

ISOTOPE	CORE	RAD. BLANKET	AX. BLANKET	TOTAL
Pu-239	- 77.0	+ 230.2	+ 242.8	+ 396.0
Pu-240	- 72.2	+ 2.2	+ 2.6	- 67.4
Pu-241	- 165.8	+ 0.0	+ 0.0	- 165.8
Pu-242	+ 21.7	+ 0.0	+ 0.0	+ 21.7
U -235	- 43.0	- 16.8	- 17.3	- 77.1
U -236	+ 9.8	+ 4.7	+ 4.8	+ 19.3
U -238	- 821.2	- 252.1	- 264.9	- 1,338.2
Pu-fissile	- 242.8	+ 230.2	+ 242.8	+ 230.2
Pu-fertile	- 50.5	+ 2.2	+ 2.6	- 45.7
Pu-total	- 293.3	+ 232.4	+ 245.4	+ 184.5
U -total	- 854.4	- 264.2	- 277.4	- 1,396.0
Fissile	- 285.8	+ 213.4	+ 225.5	+ 153.1
Fertile	- 861.9	- 245.2	- 257.5	- 1,364.6
Heavy M.	- 1,147.7	- 31.8	- 32.0	- 1,211.5

RCBN SET

Pu-239	- 75.8	+ 233.0	+ 247.5	+ 404.7
Pu-240	- 65.1	+ 2.2	+ 2.7	- 60.2
Pu-241	- 163.1	+ 0.0	+ 0.0	- 163.1
Pu-242	+ 18.0	+ 0.0	+ 0.0	+ 18.0
U -235	- 41.5	- 16.4	- 17.0	- 74.9
U -236	+ 8.8	+ 4.1	+ 4.3	+ 17.2
U -238	- 826.2	- 255.7	- 270.9	- 1,352.8
Pu-fissile	- 238.9	+ 233.0	+ 247.5	+ 241.6
Pu-fertile	- 47.1	+ 2.2	+ 2.7	- 42.2
Pu-total	- 286.0	+ 235.2	+ 250.2	+ 199.4
U- total	- 858.9	- 268.0	- 283.6	- 1,410.5
Fissile	- 280.4	+ 216.6	+ 230.5	+ 166.7
Fertile	- 864.5	- 249.4	- 263.9	- 1,377.8
Heavy M.	- 1,144.9	- 32.8	- 33.4	- 1,211.1

Table 5.8

MASS BALANCE IN ONE CYCLE

(BOL ⟶ 1 FULL POWER YEAR, BURNUP MODEL-2)

JAERI-FAST SET (kg)

ISOTOPE	CORE	RAD. BLANKET	AX. BLANKET	TOTAL	ERROR*
Pu-239	- 64.3	+ 279.0	+ 306.0	+ 520.7	- 3.4
Pu-240	- 72.9	+ 3.2	+ 5.0	- 64.7	+14.1
Pu-241	- 169.0	+ 0.0	+ 0.0	- 169.0	+ 1.8
Pu-242	+ 22.8	+ 0.0	+ 0.0	+ 22.8	+20.6
U -235	- 6.3	- 5.8	- 6.2	- 18.3	0.0
U -236	+ 1.4	+ 1.7	+ 1.8	+ 4.9	+11.0
U -238	- 882.6	- 304.0	- 335.5	- 1,522.1	- 1.7
Pu-fissile	- 233.3	+ 279.0	+ 306.0	- 351.7	- 5.8
Pu-fertile	- 50.1	+ 3.2	+ 5.0	- 41.9	+10.8
Pu-total	- 283.4	+ 282.2	+ 311.0	+ 309.8	- 7.7
U -total	- 887.5	- 308.1	- 339.9	- 1,535.5	- 1.7
Fissile	- 239.6	+ 273.2	+ 299.8	+ 333.4	- 6.1
Fertile	- 931.3	- 299.1	- 328.7	- 1,559.1	- 1.4
Heavy M.	- 1,170.9	- 25.9	- 28.9	- 1,225.7	- 0.0

RCBN SET

ISOTOPE	CORE	RAD. BLANKET	AX. BLANKET	TOTAL	ERROR*
Pu-239	- 63.4	+ 286.5	+ 316.2	+ 539.3	
Pu-240	- 65.6	+ 3.4	+ 5.5	- 56.7	
Pu-241	- 166.1	+ 0.0	+ 0.1	- 166.0	
Pu-242	+ 18.9	+ 0.0	+ 0.0	+ 18.9	
U -235	- 6.1	- 5.9	- 6.3	- 18.3	
U -236	+ 1.3	+ 1.5	+ 1.6	+ 4.4	
U -238	- 887.1	- 312.7	- 348.0	- 1,547.8	
Pu-fissile	- 229.5	+ 286.5	+ 316.3	+ 373.3	
Pu-fertile	- 46.7	+ 3.4	+ 5.5	- 37.8	
Pu-total	- 276.2	+ 289.9	+ 321.8	+ 335.5	
U -total	- 891.9	- 317.1	- 352.7	- 1,561.7	
Fissile	- 235.6	+ 280.6	+ 310.0	+ 355.0	
Fertile	- 932.5	- 307.8	- 340.9	- 1,581.2	
Heavy M.	- 1,168.1	- 27.2	- 30.9	- 1,226.2	

* relative value (%)

Table 5.9

BREEDING RATIO

(BURNUP MODEL-2)

	FULL POWER YEARS	JAERI-FAST SET				RCBN SET			
		CORE	RAD.B.	AX.B.	TOTAL	CORE	RAD.B.	AX.B.	TOTAL
BOL	0	0.771	0.274	0.305	1.350	0.774	0.285	0.320	1.379
	1	0.754	0.283	0.312	1.348	0.754	0.291	0.325	1.370
	1*	0.756	0.281	0.311	1.348	0.757	0.290	0.325	1.372
	2	0.736	0.290	0.314	1.341	0.734	0.298	0.327	1.359
BOC-1	2*	0.739	0.287	0.309	1.335	0.739	0.296	0.322	1.357
EOC-1	3	0.719	0.297	0.312	1.328	0.716	0.304	0.323	1.344

* Core is refuelled.

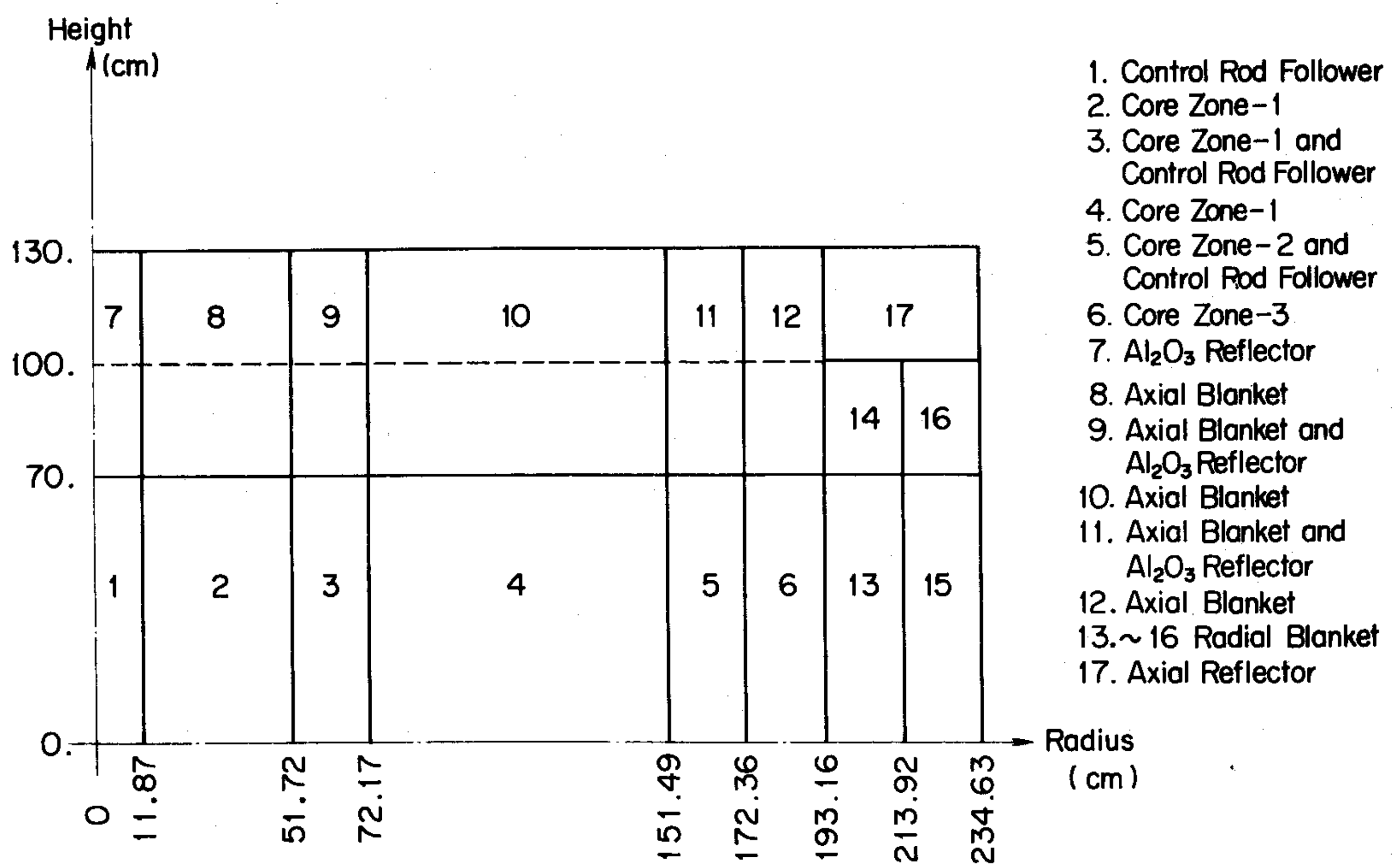

Fig. 5.1 R-Z MODEL-1 FOR Burnup Calculation

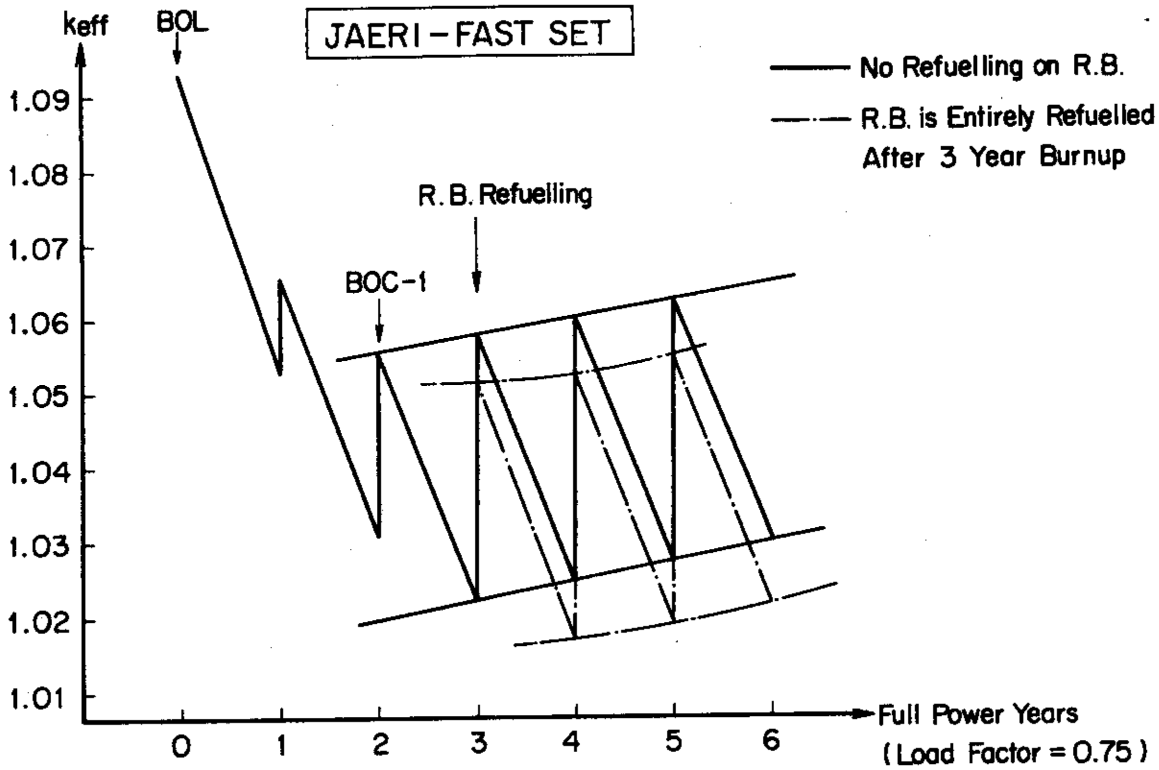

Fig. 5.2 keff versus Time (Burnup Model-1)

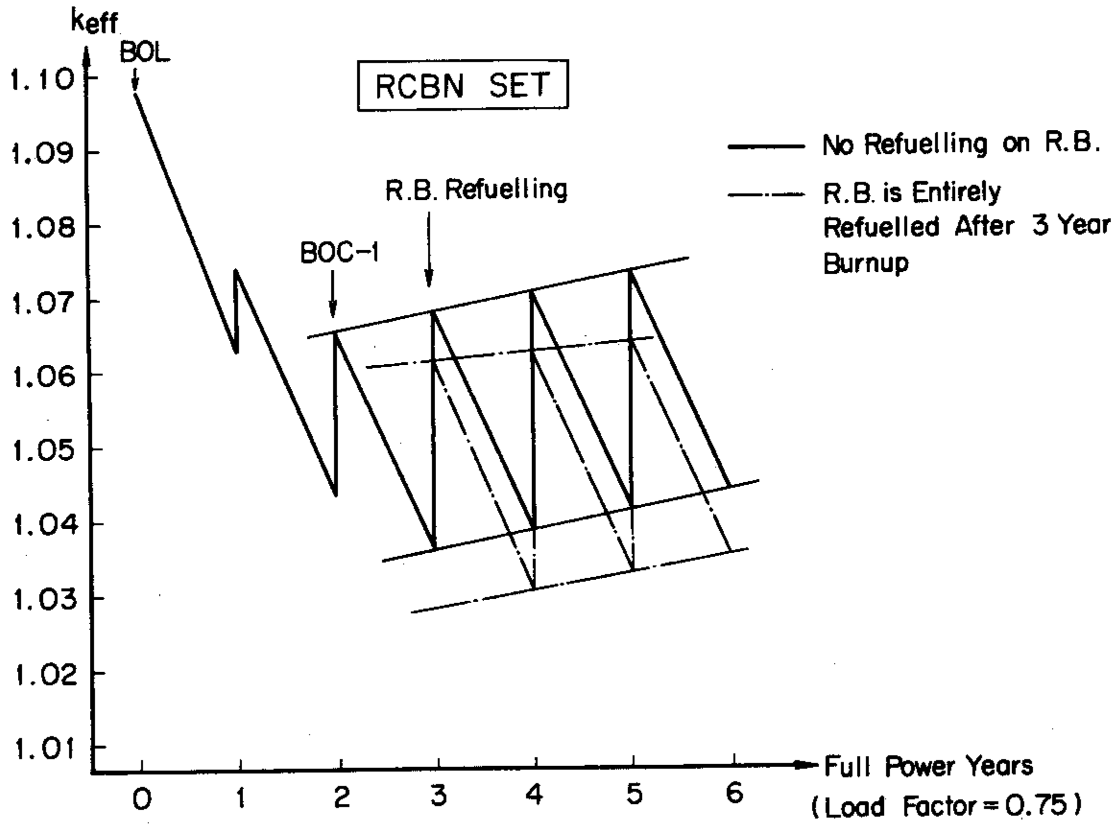

Fig. 5.3 keff versus Time (Burnup Model-1)

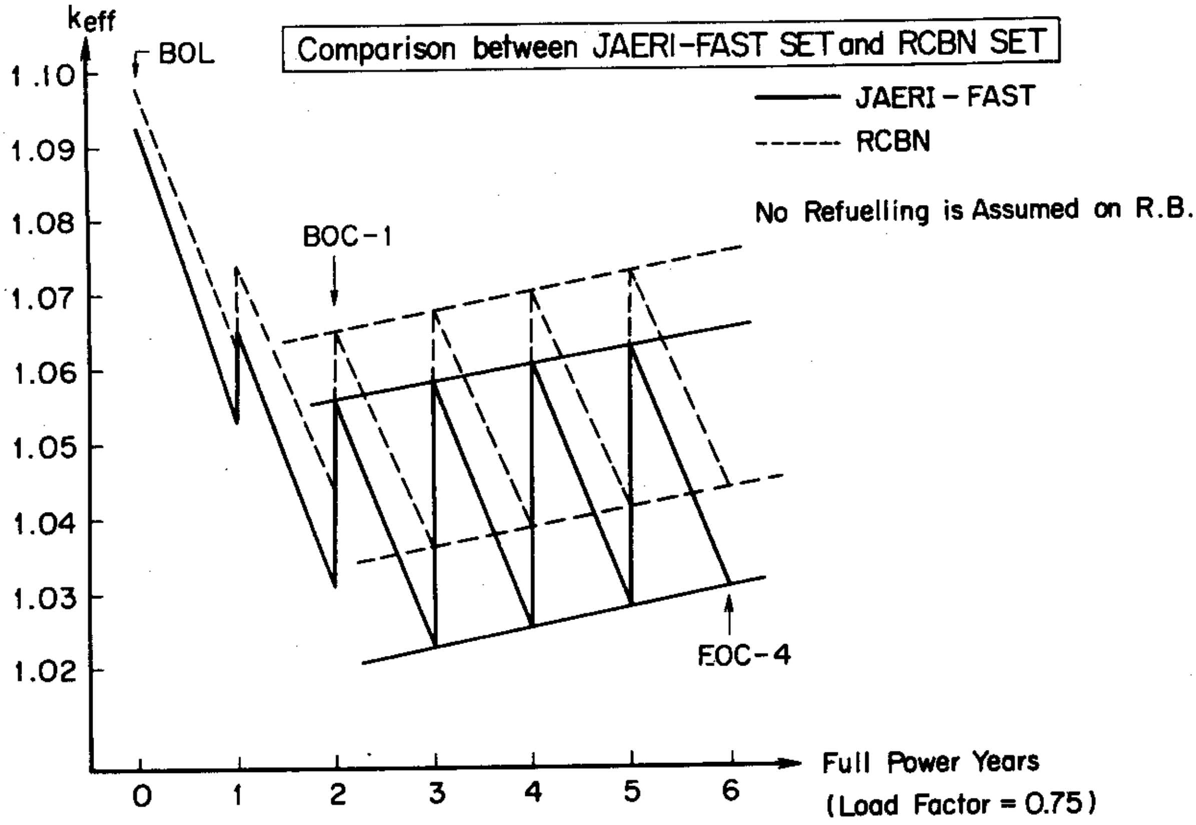

Fig. 5.4 k_{eff} versus Time (Burnup Model-1)

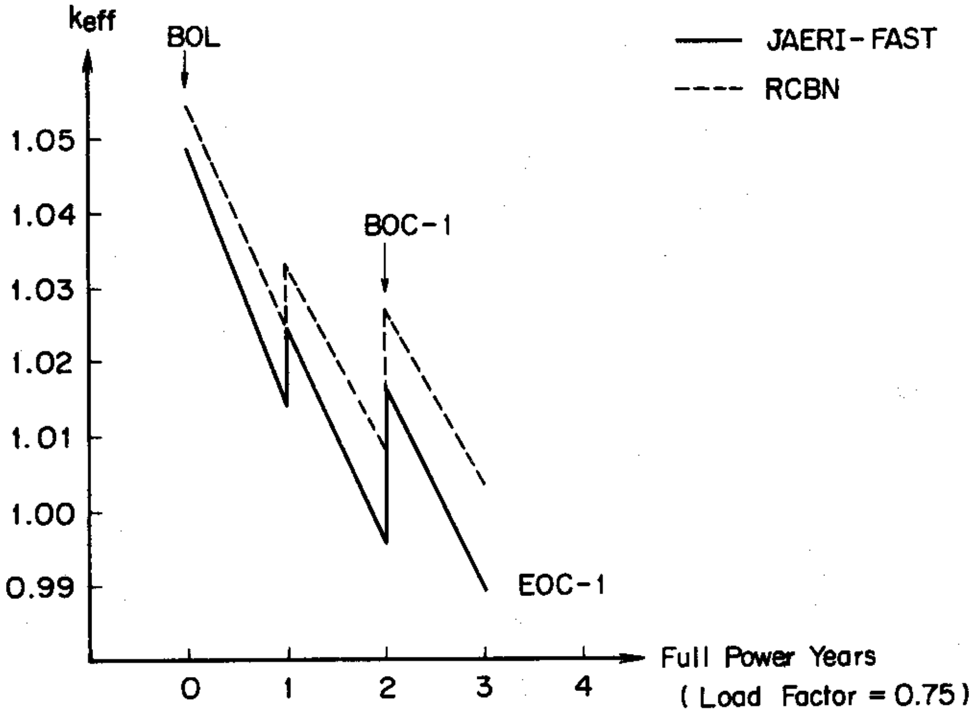

Fig. 5.5 k_{eff} versus Time (Burnup Model-2)

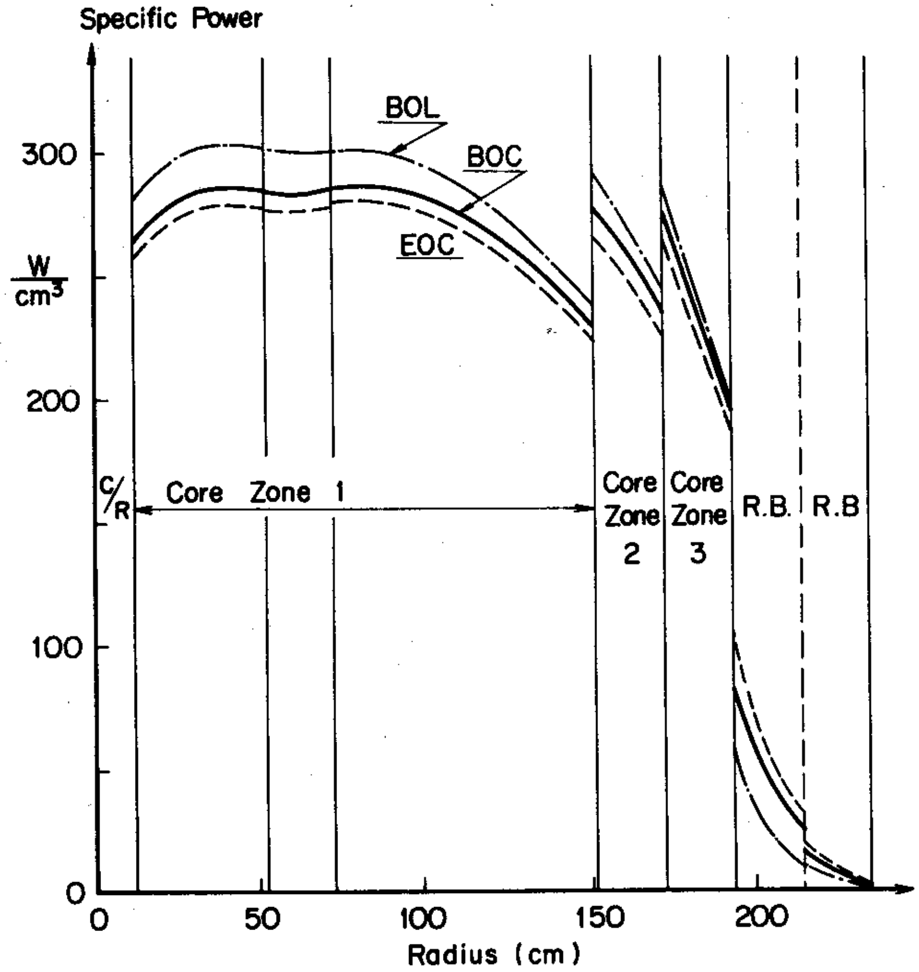

Fig. 5.6 Power Swing in Midplane
(R-Z Calculation)
(Burnup Model - 1)

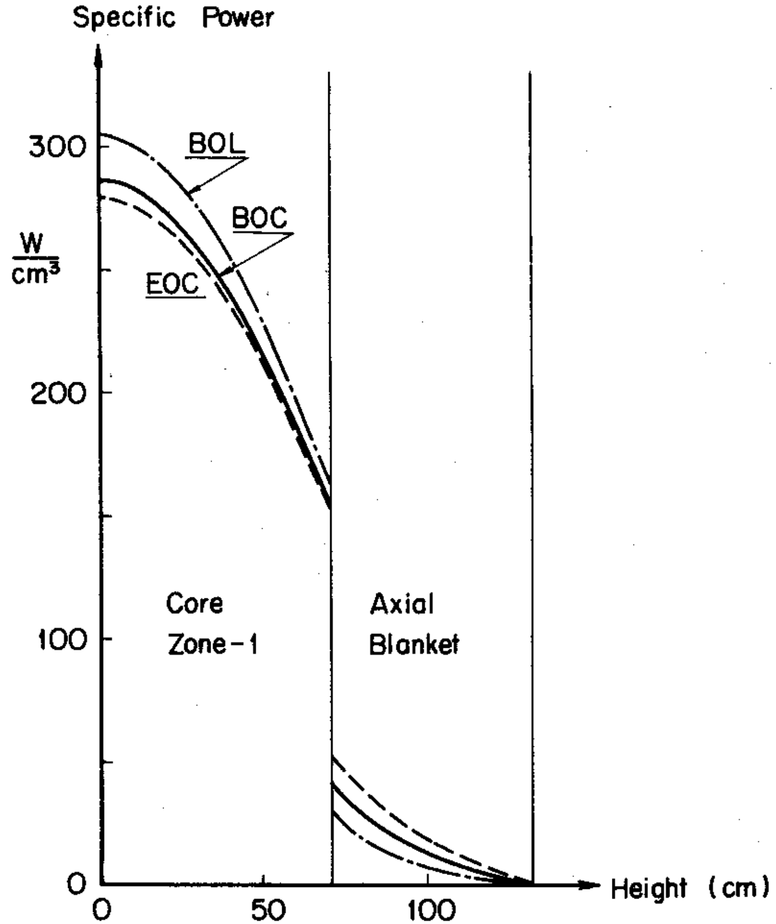

Fig. 5.7 Power Swing Along Core Axis
in Core Zone-1 (R= 41.76cm)
(Burnup Model - 1)

6. CONCLUSION

In summary the following conclusions may be drawn;

-the ordinary diffusion treatment can be applicable to the pin-type gas cooled fast reactor;

-however, about the axial streaming of the neutrons, further investigation must be continued in order to get the exact prediction, since the results of the Monte Carlo treatment and the collision probability treatment are not always in agreement.

-the difference between JAERI-FAST set and RCBN set is small in the reactivity calculation (0.3 %), but it has a tendency to increase with burnup progression. This suggests that the review on absorption cross-sections of heavy nuclides must be needed.

Acknowledgement

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Acknowledgement

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