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OECD/NEA BURNUP CREDIT CRITICALITY BENCHMARKS PHASE IIIA: CRITICALITY CALCULATIONS OF BWR SPENT FUEL ASSEMBLIES IN STORAGE AND TRANSPORT

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OECD/NEA Burnup Credit Criticality Benchmarks Phase IIIA: Criticality Calculations of BWR Spent Fuel Assemblies in Storage and Transport

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The report describes the final results of Phase IIIA Benchmarks conducted by the Burnup Credit Criticality Calculation Working Group under the auspices of the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA). The benchmarks are intended to confirm the predictive capability of the current computer code and data library combinations for the neutron multiplication factor (k_{eff}) of a layer of irradiated BWR fuel assembly array model. In total 22 benchmark problems are proposed for calculations of k_{eff}. The effects of following parameters are investigated: cooling time, inclusion/exclusion of FP nuclides and axial burnup profile, and inclusion of axial profile of void fraction or constant void fractions during burnup. Axial profiles of fractional fission rates are further requested for five cases out of the 22 problems. Twenty-one sets of results are presented, contributed by 17 institutes from 9 countries. The relative dispersion of k_{eff} values calculated by the participants from the mean value is almost within the band of ±1 %Δk/k. The deviations from the averaged calculated fission rate profiles are found to be within ±5 % for most cases.

Keywords: Burnup Credit, BWR Fuel, Void, Benchmark, Criticality, OECD, NEA, Burnup Profile, Actinide, Fission Product

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OECD/NEA 燃焼度クレジット臨界計算ベンチマーク問題 IIIA：貯蔵及び輸送用 BWR 使用済燃料集合体の臨界計算

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この報告書は、経済開発機構原子力機関（OECD/NEA）の後援の下に燃焼度クレジット臨界計算ワーキンググループで実施されたベンチマーク問題 IIIA の最終結果を示す。ベンチマーク問題は、1 層の沸騰水型原子炉使用済燃料集合体配列モデルの中性子増倍率（k_{eff}）に対する、現行の計算コード及びデータ・ライブラリの組合せが持つ予測能力を確認することを意図している。全部で 22 題の k_{eff}を計算するベンチマーク問題を提案している。以下のパラメタの効果を検討する：冷却期間、FP 核種考慮の有無、軸方向燃焼度分布考慮の有無、燃焼時の軸方向ボイド率分布または一定ボイド率の考慮。22 題のうち 5 題の問題については、軸方向の核分裂割合の分布も要求している。9 ヶ国からなる 17 機関の寄与により、21 組の結果を提示している。参加者により計算された k_{eff}値の平均値からの相対的な広がりは、ほぼ±1%Δk/k の幅に収まっている。また、核分裂割合分布計算値の平均からのずれは、多くの場合において±5 パーセントの範囲内にあるとの結果を得ている。

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1. Introduction

Reactivity of thermal reactor fuels typically decreases as the fuel burnup proceeds. Essentially, this arises from a decrease in concentration of fissile nuclides in fuel and an increase in the concentration of fission products (FPs) that absorb neutrons. As direct measurements of reactivity are not practical at present, reliable computational tools are crucial for nuclear criticality safety evaluation of irradiated fuels in storage and transportation[1,2].

The Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA) formed a Burnup Credit Criticality Calculation Working Group in 1991 and has proceeded with a series of related benchmark calculations[3,4]. From 1991 to 1995, its main activity was calculational benchmarking for PWR irradiated fuels. In Phase I, separate criticality and burnup calculations were proceeded for an infinite array of PWR fuel rods with an axially infinite length. The criticality calculations were intended to study the differences of k_{eff} results among participants in modelling actinides and FP nuclides (Phase IA)[5]; the burnup calculations were planned to provide a comparison of different code systems for the predictive capability for spent fuel isotopic concentrations (Phase IB)[6]. Phase II benchmarks studied end effects by assuming an axial profile of burnup for PWR fuel rods. Phase IIA[7] dealt with an infinite array of fuel rods with axial symmetry about the mid-plane. The subject of Phase IIB[8] was a finite number of fuel assemblies (21) in a fuel cask. The configuration was slightly axially asymmetric about the mid-plane due to different axial boundary conditions.

Benchmark calculations for irradiated BWR fuels started in 1996 as a new series, Phase III. The existence of voids during burnup is an important difference from the irradiated PWR fuels studied in earlier Phases. This paper presents the results of Phase IIIA Benchmarks for criticality calculations of irradiated BWR fuel assemblies. Phase IIIB Benchmarks, which are in progress, are designed to compare the predictive capability of depletion codes.

Chapter 2 of this report gives an overview of the benchmark problems. Chapter 3 summarizes participants and their analysis methods, followed by Chapter 4 which presents the results of their calculations with some discussion. Some general conclusions are drawn in the final chapter.

2. Overview of Benchmark Problems

An infinite array of an irradiated BWR fuel assembly was specified for Phase IIIA criticality benchmark calculations. The assembly was modelled based on the so-called "STEP-II-typed assemblies," (Japanese 2nd-step fuel) for average exposure of 40 GWd/tU. **Figure 2.1** shows a conceptual drawing of the assembly, i.e., the left-hand-side is its horizontal cross section, and the right-hand-side is a vertical cross section of a fuel rod and a water rod. Each fuel assembly consisted of an eight-by-eight fuel rod array immersed in water, in which array a thick (3.2-cm-diameter) water rod was centered replacing the central 2 × 2 rods, and a channel box, which was surrounded by an 8.46-mm-thick water reflector, i.e., half the water gap. The reflective boundary condition was imposed outside a 15.24 cm × 15.24 cm square cell of the fuel assembly. All the fresh fuel rods were assumed to be identical for simplicity (See Appendix III for this modelling). Although a BWR fresh fuel rod (and also fuel assembly) may contain several levels of uranium enrichment, we only distinguished the net fuel region (3.8 wt% enriched), and the top and bottom blanket regions (0.71 wt% enriched).

The total length of the fuel rods including the blanket regions was about 370 cm. The fuel rod was divided into 9 axial regions to allow modelling of axial variations in atomic number densities. Axially homogeneous fuel burnup and/or void fractions during burnup were assumed for some cases. In such cases, the atomic number densities provided were averaged over the entire fuel region obtained by fuel burnup calculations. Precise specifications are given in Appendix I, including the atomic number densities, which were supplied as a part of the specifications.

In total 22 benchmark problems were specified, and the calculated neutron multiplication factor k_{eff} was to be reported for each problem. Parameters and problem numbers are summarized in **Table 2.1**. The basic parameters were cooling time (1 year or 5 years), inclusion or exclusion of FPs, modelling of an axial burnup profile or uniform profile, assumption of a void fraction profile or just a constant fraction (40 % or 70 %), and the mean fuel burnup (0, 20, 30 and 40 GWd/tU). **Table 2.2** lists actinides and FP nuclides to be included in the benchmark calculations. These nuclides are common to the Phase IIA Benchmarks, which treated an infinite array of irradiated PWR fuel.

In addition to the neutron multiplication factor, fractional fission rates were requested for 9 axial regions for 5 cases, #1, #5, #6, #7 and #14, which are shown in **Table 2.1** in bold type. As an example, number density distributions of ^{235}U for the 5 cases are displayed in **Fig. 2.2**. Void fraction distributions are compared in **Fig. 2.3** for Cases #7 and #14. These two figures are included to provide the reader with images of the profiles specified in the benchmark problems. Note that the void fraction profile for Case #7 gives the averaged values for all the four cycles tabulated in **Table A2** of Appendix I.

Table 2.1 Parameters and problem numbers (indicated with #)

Cooling Time [y]	Fission Products	Burnup Profile	Void Profile	Mean Burnup [GWd/tU]			
				0	20	30	40
1	Yes	Yes	Yes	**#1**	#2	#3	#4
5	Yes	Yes	Yes		**#5**	**#6**	**#7**
		No	Yes		N/A	N/A	#8
			40%(uniform)		#9	#10	#11
			70%(uniform)		#12	#13	**#14**
	No	Yes	Yes		N/A	N/A	#15
		No	Yes		N/A	N/A	#16
			40%(uniform)		#17	#18	#19
			70%(uniform)		#20	#21	#22

Note: Fractional fission rates are also requested for the **bold** cases.

Table 2.2 The actinides and FP nuclides specified to be included in calculation

Actinides (12 nuclides in total)	^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{237}Np
Fission Products (15 nuclides in total)	^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{143}Nd, ^{145}Nd, ^{153}Eu, ^{155}Gd

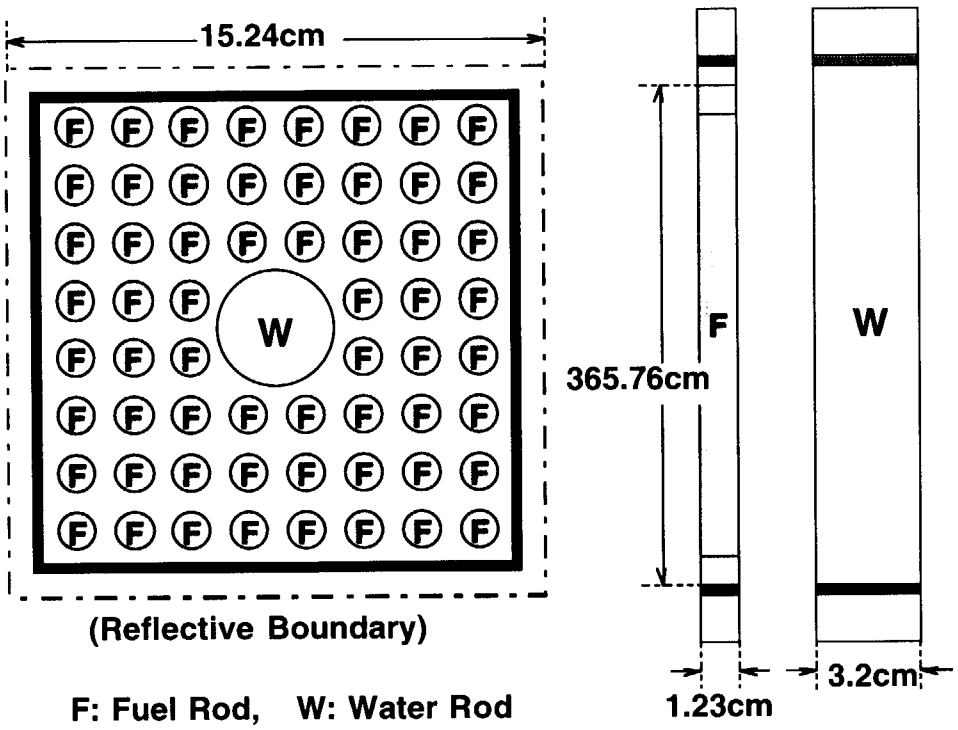

Fig. 2.1 A model of a BWR fuel assembly

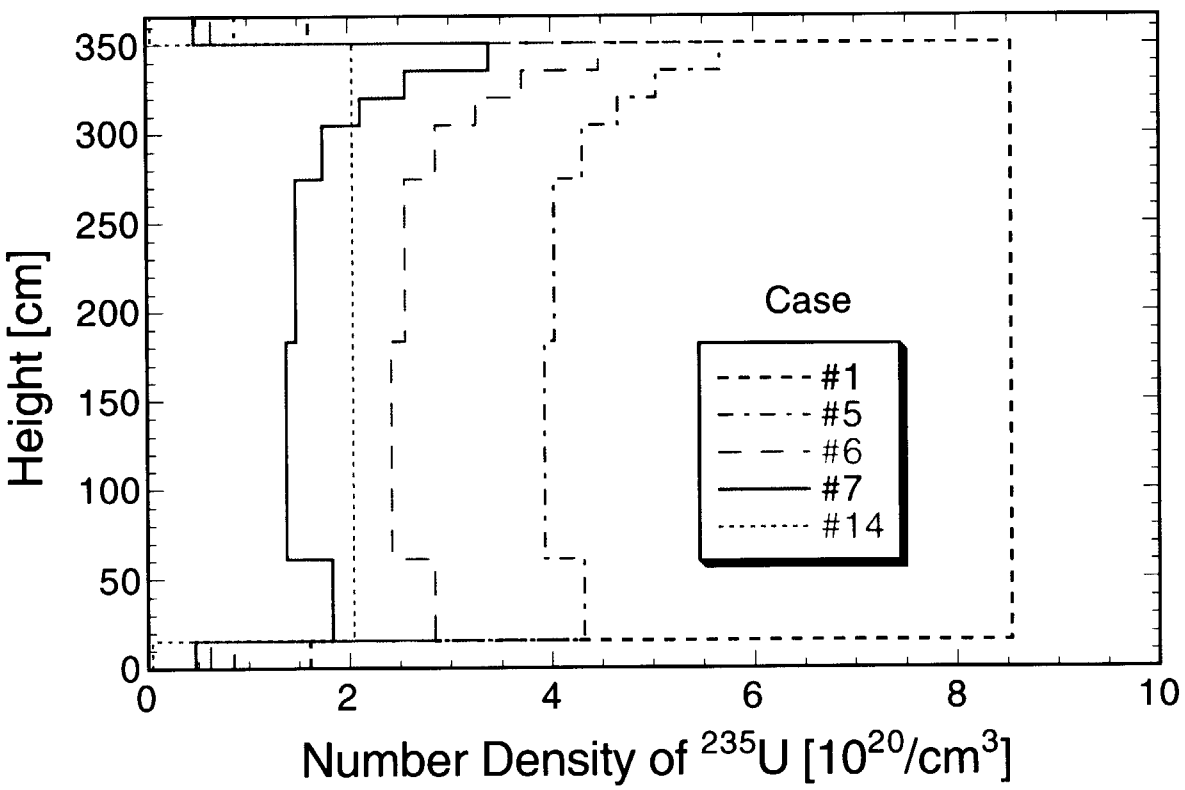

Fig. 2.2 Number density distributions of ^{235}U for Cases #1, #5, #6, #7 and #14

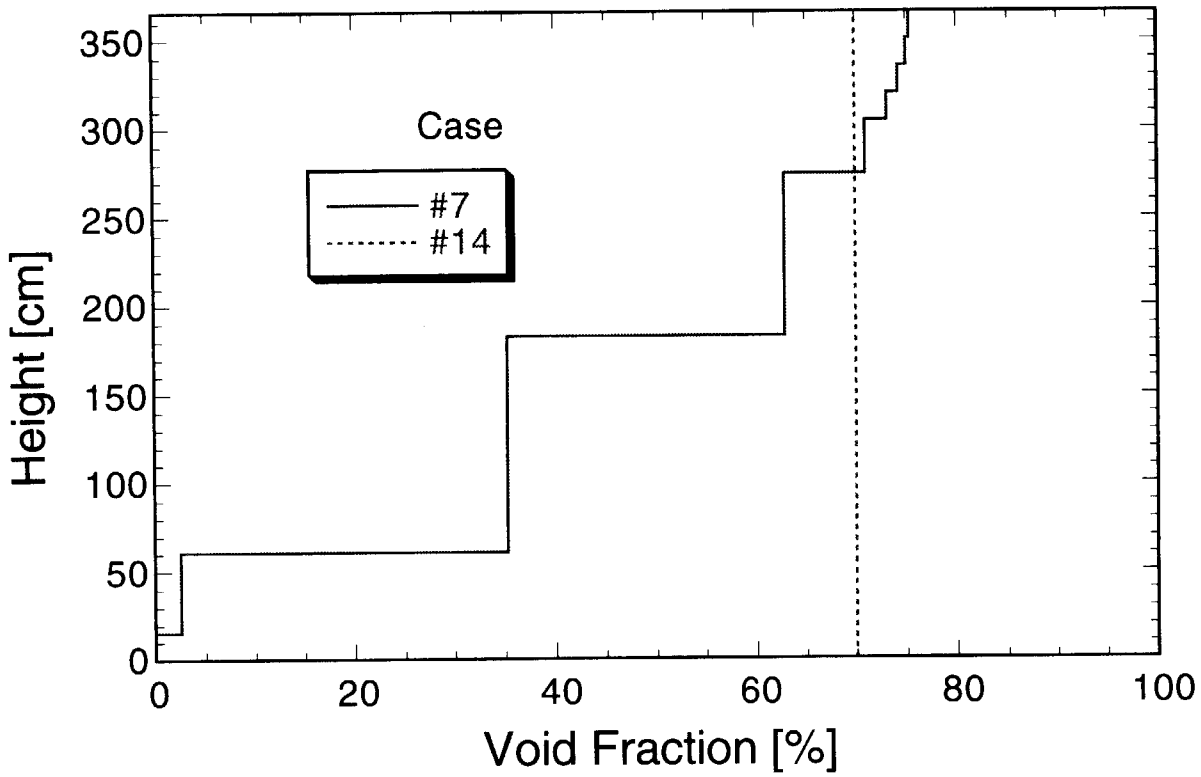

Fig. 2.3 Void fraction distributions for Cases #7 and #14

3. Participants and Calculation Methods

A copy of the problem specification as given in Appendix I was distributed to the members of OECD/NEA Burnup Credit Working Group in January 1996. The deadline for submitting calculation results was set in the specification as the end of March 1996, which was, however, postponed later until the end of May 1996.

Table 3.1 lists the final participation in the Phase IIIA series of benchmarks. It includes names of the contact person, his/her affiliation, the code used and the evaluated nuclear data files that the calculations were based on. Further details of the codes and libraries used by each participant may be found in Appendix II, which collects the final results submitted from the participants. Note that **ID** in **Table 3.1** identifies the participants throughout this report, either in uppercase (**A**, **B**, **C**, ...) or in lowercase (**a**, **b**, **c**, ...). **Table 3.2** arranges all the participating institutes according to their countries. It shows that 21 results in total were submitted from 17 institutes of 9 countries. Almost a half of the total number of participants came from two countries: Japan and U.K.

The evaluated nuclear data files for the main actinides that the calculations are based on can be used to categorize the participants. **Table 3.3** indicates that 9 were based on ENDF (ENDF-B4, -B5 and -B6), 8 on JEF (JEF-1 and -2.2), 2 on JENDL (JENDL-3.2), and the remaining 2 were based on UKNDL. Although various combinations of nuclear data libraries and criticality codes are possible, the combinations shown in **Table 3.3** are believed to cover most of the current computation systems typically used, or to be used, for licensing criticality safety of BWR spent fuel assemblies in storage and transport. **Table 3.4** summarizes the criticality codes applied to the calculation. Both the probabilistic methods (Monte Carlo methods) and deterministic methods were applied. The table indicates that KENO-V.a is popular worldwide and MONK, in various versions, is supported mainly in the U.K.

Table 3.1 Participation in Phase IIIA benchmarks

ID	Code	Nuclear Data	Contact Person	Institute	Country
A	PHOENIX4/KENO5a	ENDF/B-6	W. Lipiec	ABB	Sweden
B	TGBLA/ALEX	ENDF/B-5,-4, JENDL-3.1	Y. Ando	Toshiba	Japan
C	CASMO-SIMULATE	ENDF/B-4	J.M. Conde	CSN	Spain
D	SRAC93/CITATION	JENDL-3.2	K. Suyama	JAERI	Japan
E	MONK7A	UKNDL, JEF-2.2	P.R. Thorne & R.L. Bowden	BNFL	U.K.
F	MCNP4A	JEF-2.2	W. Bernnat	U. of Stuttgart	Germany
G	RESMOD/KENO5a	JEF-2.2	W. Bernnat	U. of Stuttgart	Germany
H	MCNP4A	JENDL-3.2	H. Sakamoto	JAERI	Japan
I	APOLLO2+TRIMARAN2	JEF-2.2	Y.K. Lee	CE-SACLAY	France
J	SCALE4.2	ENDF/B-4,-5	I. Nojiri	PNC	Japan
K	SCALE4.1	ENDF/B-4,-5	D. Mennerdahl	EMS	Sweden
L	KENO5a/MGCL-JINS	ENDF/B-4, JENDL-3.2	S. Mitake	NUPEC	Japan
M	SCALE4.2	ENDF/B-4	S. Mitake	NUPEC	Japan
N	APOLLO1+MORET3	JEF-1, ENDF/B	A. Nouri, G. Poullot	CEA-IPSN	France
O	WIMS6+KENOVa	JEF-2.2	Th. Maldague	Belgonucleaire	Belgium
P	BOXER	JEF-1	P. Grimm	PSI	Switzerland
Q	SCALE4.3	ENDF/B-5,-6	M.D. DeHart	ORNL	U.S.A.
R	MONK7	JEF-2.2	D. Hanlon	AEA Technology	U.K.
S	VIM	ENDF/B-5,-6	R.N.Blomquist	ANL	U.S.A.
T	MONK6B	UKNDL*1, JEF-2.2	J. Stewart	DOT	U.K.
U	MONK6B	UKNDL*2, JEF-2.2	J. Stewart	DOT	U.K.

*1 With standard data adjustments.
*2 With no data adjustments.

Table 3.2 Participating countries and institutes to Phase IIIA Benchmarks

No.	Country	Institute	No. of Participants
1	Belgium	Belgonucleaire	1
2	France	CEA-IPSN, CE-SACLAY	2
3	Germany	U. of Stuttgart(2)*	2
4	Japan	JAERI(2)*, NUPEC(2)*, PNC, Toshiba	6
5	Spain	CSN	1
6	Sweden	ABB, EMS	2
7	Switzerland	PSI	1
8	U.K.	AEA Technology, BNFL, DOT(2)	4
9	U.S.A.	ANL, ORNL	2
Total	9	17(21)*	21

* If a participant applied different methods, the number of methods is shown in ( ).

Table 3.3 Evaluated nuclear data and criticality codes applied to the benchmark calculations

Evaluated Nuclear Data*1	Computer Code (**ID***2)	No. of Participants	
ENDF-B4	CASMO/SIMULATE(**C**), KENO5a(**J,K,L,M**)	5	9
ENDF-B5	KENO5a(**Q**), TGBLA/ALEX(**B**), VIM(**S**)	3	
ENDF-B6	KENO5a(**A**)	1	
JEF-1	BOXER(**P**), MORET3(**N**)	2	8
JEF-2.2	KENO5a(**G,O**), MCNP4A(**F**), MONK7(**R**), MONK7A(**E**), TRIMARAN2(**I**)	6	
JENDL-3.2	CITATION(**D**), MCNP4A(**H**)	2	2
UKNDL	MONK6B(**T,U**)	2	2
Total		21	

*1 For the main actinides.
*2 See **Table 3.1**.

Table 3.4 Criticality codes applied in the benchmark calculations

Probabilistic or Deterministic	Computer Code	Version (**ID***)	No. of Participants	
Probabilistic	KENO	5a(**A,G,J,K,L,M,O,Q**)	8	17
	MCNP	4A(**F,H**)	2	
	MONK	6B(**T,U**), 7(**R**), 7A(**E**)	4	
	MORET	3(**N**)	1	
	TRIMARAN	2(**I**)	1	
	VIM	(**S**)	1	
Deterministic	BOXER	(**P**)	1	4
	CASMO/SIMULATE	(**C**)	1	
	CITATION	(**D**)	1	
	TGBLA/ALEX	(**B**)	1	
Total			21	

* See **Table 3.1**

4. Results and Discussions

4.1 Multiplication factor

The calculation results of the neutron multiplication factor, k_{eff}, are listed in **Table 4.1**, up to four decimal points. Note that, however, some of the results were not so precise, since the typical 2σ values for the Monte Carlo calculation results were in the range of 0.07 % to 0.26 % (See the last line of **Table 4.1**). The last three columns of **Table 4.1** show the following quantities: the averaged k_{eff} (denoted as $\bar{k}_{eff,i}$), two standard deviations ($2\sigma_i$) and their ratio, $2\sigma_i^{(r)} = 2\sigma_i/\bar{k}_{eff,i}$ of all the submitted results for Case #*i*. As in the Phase IIB report[8], $\sigma_i^{(r)}$ is defined as the relative standard deviation. It is worthwhile to make the following two observations:

(1) Twice the relative standard deviation is generally less than 1 %Δk/k except for the fresh fuel case (Case #1), for which it amounts to 1.3 %Δk/k. This tendency that the fresh fuel case has the largest deviation was also observed in other criticality benchmark series Phase IIA[7] and IIB[8] that dealt with irradiated PWR fuel assemblies.

(2) The relative standard deviations for cases with FPs were generally a little smaller than those for the corresponding cases without FPs (See the table below). This tendency is common to Phase IIB, where a cask model of irradiated PWR fuel assemblies was studies.

Burnup Profile	Void Profile	Mean Burnup [GWd/tU]	With FPs		Without FPs	
			Case #	$2\sigma^{(r)}$	Case #	$2\sigma^{(r)}$
Yes	Yes	40	7	0.0079	15	0.0084
No			8	0.0073	16	0.0082
	40% (uniform)	20	9	0.0087	17	0.0104
		30	10	0.0080	18	0.0093
		40	11	0.0080	19	0.0081
	70% (uniform)	20	12	0.0089	20	0.0098
		30	13	0.0080	21	0.0090
		40	14	0.0088	22	0.0087

It would appear that the more nuclides are involved in criticality calculations of irradiated BWR fuel assemblies, the smaller the deviations among different codes and data become. This might be due to cancellation of errors between various nuclides for complicated geometry systems.

Figure 4.1 displays the $\bar{k}_{eff,i}$ value for all the 22 cases, showing an overall range from 0.95 to 1.40. **Table 4.2** shows the relative deviation in % of $k_{eff,i}^{(x)}$ value for Participant **X** from the averaged value $\bar{k}_{eff,i}$ over all the participants. Note that small letters are used in the

symbolic forms for participant identification instead of capital letters to remind the readers that they are related to deviations. The relative deviation is denoted by $\delta_i^{(x)}$ for each case i in % and Participant **X**, which is defined as

$$\delta_i^{(x)} = 100\left(\frac{k_{eff,i}^{(x)}}{\bar{k}_{eff,i}} - 1\right).$$

The last two rows of **Table 4.2a** show the average $\bar{\delta}^{(x)}$ and two standard deviations $2\sigma^{(x)}$, respectively, of relative deviations $\delta_i^{(x)}$ with respect to all the cases i for Participant **X**. The average $\bar{\delta}^{(x)}$ indicates a weighted difference in k_{eff} of Participant **X** and the averaged one. The two standard deviation value is an index that reflects the consistency of calculations. In other words, the smaller this value is, the more consistent the calculation is with the average. For plotting convenience, all the results are divided into three groups. The first group, which consists of **d**, **g**, **h**, **j**, **l**, **q** and **s**, satisfies $|\bar{\delta}^{(x)}| < 0.3$ and $2\sigma^{(x)} < 0.3$. The second group, which consists of **a**, **f**, **k**, **m**, **n**, **p** and **r**, satisfies $|\bar{\delta}^{(x)}| < 0.4$. The others, namely **b**, **c**, **e**, **i**, **o**, **t** and **u**, are categorized into the third group. The relative deviations for each group are shown in **Figures 4.2a**, **4.2b** and **4.2c**, respectively. From these figures it is clear that the relative deviations are almost all within the band of ±1 %Δk/k.

Further observations can be made for the individual results from some participants: **b** is systematically low (from 0.6 to 1.6 %) relative to the average; **c** is systematically high (from 0.1 to 1.1 %) except for Case #1 (fresh fuel), which is 0.2 % lower than the average; **e** and **t** are exceptionally 0.9 % high for Case #1; **n** is always high, especially for Cases #14 and #22. Some results have clear tendencies with respect to the fuel burnup. As the fuel burnup proceeds, $\Delta k_{eff}/\bar{k}_{eff}$ increases (for **b** and **c**) or decreases (for **o** and **t**).

We also calculated the correlation coefficients between every pair of relative deviations. The correlation coefficient between the results **x** and **y** was defined as:

$$\rho_{\mathbf{x},\mathbf{y}} = \frac{1}{\sigma^{(x)} \cdot \sigma^{(y)}} \cdot \frac{1}{n} \sum_{i=1}^{n} \left(\delta_i^{(x)} - \bar{\delta}^{(x)}\right)\left(\delta_i^{(y)} - \bar{\delta}^{(y)}\right),$$

where n = 22 in this case. The results are summarized in **Table 4.2b**. Using the correlation coefficients, the results for k_{eff} are categorized into three correlated groups: Group A: **a**, **e**, **f**, **g**, **h**, **i**, **o** and **t**; Group B: **b**, **c**, **d**, **k**, **m**, **p**, **q** and **s**; and Group Γ: **j**, **l**, **n** and **r** (**u** is difficult to be categorized into any of these groups). Group A is mainly comprised of JEF-2.2 users. Group B is mainly consisted of ENDF/B-4 and -5 users. Group Γ is a mixture in the aspect of the evaluated nuclear data.

4.2 Fission rates

Regionwise fission rates calculated by the participants are summarized in **Table 4.3**.

Regions 1 and 9 correspond to the bottom and top blankets, respectively. Note that the required quantity in the specification is defined as [see the note at the bottom of "5. Requested Information and Results" in Appendix I]:

$$[F.F.D.]_n = \frac{\iint dEd\bar{r}_n \Sigma_f(\bar{r}_n, E)\Phi_n(\bar{r}_n, E)}{\sum_{i=1}^{9} \iint dEd\bar{r}_i \Sigma_f(\bar{r}_i, E)\Phi_i(\bar{r}_i, E)}.$$

This quantity was called "fractional fission density for Region n" in the specification. However, this report calls it "fractional fission rates for Region n", considering the nature of the defined quantity.

The averaged profile of all the fractional fission rates submitted from participants are shown in **Fig. 4.3a**. It is observed that the fractional fission rates of upper regions increase as the fuel burnup proceeds from 20 GWd/tU (Case 5) to 40 GWd/tU (Case 7). This tendency is more clearly shown in **Fig. 4.3b** that shows the normalized fission densities (see also Fig. 1 of Appendix IV.2, which collects axial distributions of normalized fission densities for all the 22 cases calculated with KENO-V.a code combined with SCALE 27 and MGCL-JINS libraries). It is worthwhile to point out that the fractional fission rates for Case 14 (40 GWd/tU, 5 y cooling, FPs included, uniform burnup, 70% uniform void) and Case 1 (fresh fuel case) are very similar, and they are almost not distinguishable in **Fig. 4.3b**. This may be because the nuclides are assumed to be distributed uniformly in the whole fuel region for both cases.

Table 4.4 shows differences of fractional fission rates from the average, in which the last two columns provide the maximum and minimum deviations. The differences are displayed also as **Figs. 4.4a** and **4.4b,** where bars (**I**) indicate deviations from the average, and triangles (▽) show the maxima or minima. A general trend is observed that the differences decrease as the burnup increases (from Case 5 to Case 7). This trend on fractional fission rates may be closely related to the fact that the deviations in k_{eff} among participants decrease as the burnup proceeds. For example, **f** and **n** have relatively large deviations in the aspect of fractional fission rates as shown in **Fig. 4.4b**, their relative deviations from the average of k_{eff} are also profound, -0.57% and 0.93%, respectively (see **Table 4.2**). However, the correlation is not always the rule: **t** has significant deviations in fractional fission rates for Case 7 (**Fig. 4.4b**), while its relative deviation from the average of k_{eff} is only 0.04%.

4.3 End effect

The end effect is defined as the difference between k_{eff}'s with and without a burnup profile:

$\Delta k_{eff}^{(BP)} = k_{eff}$(with a burnup profile) - k_{eff}(without a burnup profile).

The 2nd and 3rd columns of **Table 4.5** show the end effect deduced from the k_{eff} results submitted by participants. The columns treat the cases of 40 GWd/tU, 5 year cooling and considering the void profile, both with and without FPs. They indicate that the end effect becomes positive (from 0.7 to 1.3 %Δk depending on participants) when we consider FPs, whereas it is negative (from -0.8 to -0.1 %Δk) when FPs are disregarded.

Only incomplete discussions can be made, since the calculation cases are limited in this series of benchmarks. K. Suyama, H. Sakamoto and Y. Ando calculated all the cases based on the complete combination of parameters, which is given in Appendix V.1. Table 4.7 and Figure 4.1 of Appendix V.1 show the end effect calculated with a combination of MCNP4A and JENDL-3.2. Because the k_{eff} results of this combination agree well with the averaged k_{eff} results (see column **h** in **Table 4.2**), the full results should provide us with useful information. They show that the end effect increases as the fuel burnup proceeds, which is consistent with the findings of Phases IIA and IIB, both of which evaluated the end effect for irradiated PWR fuels.

4.4 Void fraction effect

The void fraction effect is defined analogous to the end effect as the difference between k_{eff} considering a void profile during burnup and considering just a constant void fraction:

$$\Delta k_{eff}^{(VF)} = k_{eff}(\text{with void profile}) - k_{eff}(\text{with a constant void fraction}).$$

The relevant k_{eff} results from the participants are shown from the 4th to 7th columns in **Table 4.5** for the cases of 40 GWd/tU, 5 year cooling and axially uniform burnup. The void effect results in a decrease as the uniform void fraction increases. For 40% uniform void fraction, the void fraction effect is from 2.9 to 3.3 %Δk, and becomes negative (from -1.5 to -0.7 %Δk) when the uniform void fraction increases to 70%. This tendency is almost insensitive to the presence of FPs. The results shown in Table 4.8 of Appendix V.1 are consistent with our limited cases. However, that Table further indicates that the void fraction effect is still positive even for 70% uniform void fraction (40 GWd/tU and 5 year cooling) when the burnup profile is considered. This fact is interpreted that the top end region, where the void fraction is more than 70% (see **Fig. 2.3**), becomes more important for the cases with a burnup profile than for the cases without burnup profiles.

4.5 Effect of cooling time

The effect of cooling time in this report deals with difference in k_{eff} for different values of cooling time, 1 and 5 years:

$$\Delta k_{eff}^{(CT)} = k_{eff}(1 \text{ year cooling}) - k_{eff}(5 \text{ year cooling}).$$

The differences in k_{eff} results from the participants are shown from the 8th to 10th columns of **Table 4.5**. As the cooling time becomes longer, in this case from 1 year to 5 years, k_{eff} decreases. The difference becomes larger as the burnup proceeds, and it amounts to 2.6 to 3.1%Δk at 40 GWd/tU. The averaged results shown in **Fig. 4.5** confirm the above-mentioned tendency. The relevant cases are considering fission products, burnup and void profiles. **Table 4.6** of Appendix V.1 shows the same trend for other combinations of parameters, even though the effect is less pronounced when the FPs are disregarded.

Table 4.1 The results of the neutron multiplication factor from participants

Case #	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	$\bar{k}_{eff,i}$ *2	$2\sigma_i$ *2	$2\sigma_i/\bar{k}_{eff,i}$
1	1.4112	1.3806	1.4007	1.3963	1.4162	1.4045	1.4067	1.4082	1.4119	1.3979	1.3906	1.3989	1.3902	1.4095	1.4137	1.4058	1.4038	1.4057	1.4032	1.4159	1.3919	1.4030	0.0186	0.0133
2	1.1944	1.1815	1.2022	1.1928	1.1974	1.1918	1.1949	1.1940	1.1925	1.1919	1.1881	1.1918	1.1876	1.1985	1.2002	1.1991	1.1982	1.1924	1.1957	1.1997	1.1901	1.1940	0.0098	0.0082
3	1.1134	1.1008	1.1204	1.1091	1.1139	1.1076	1.1107	1.1115	1.1083	1.1049	1.1075	1.1082	1.1047	1.1140	1.1159	1.1161	1.1149	1.1088	1.1132	1.1151	1.1067	1.1107	0.0094	0.0085
4	1.0278	1.0197	1.0377	1.0250	1.0282	1.0250	1.0270	1.0263	1.0241	1.0247	1.0241	1.0256	1.0209	1.0309	1.0285	1.0322	1.0294	1.0275	1.0309	1.0283	1.0247	1.0271	0.0079	0.0077
5	1.1850	1.1694	1.1897	1.1802	1.1858	1.1786	1.1823	1.1836	1.1806	1.1759	1.1770	1.1794	1.1742	1.1845	1.1876	1.1867	1.1852	1.1783	1.1842	1.1896	1.1751	1.1816	0.0108	0.0091
6	1.0930	1.0810	1.0998	1.0886	1.0896	1.0858	1.0902	1.0908	1.0856	1.0881	1.0885	1.0883	1.0853	1.0950	1.0945	1.0957	1.0949	1.0889	1.0958	1.0952	1.0869	1.0905	0.0092	0.0084
7	0.9981	0.9924	1.0090	0.9963	0.9991	0.9943	0.9984	0.9988	0.9961	0.9956	0.9936	0.9970	0.9952	0.9998	1.0013	1.0042	1.0034	0.9991	0.9993	0.9987	0.9936	0.9983	0.0079	0.0079
8	0.9906	0.9827	0.9995	0.9873	0.9878	0.9854	0.9880	0.9888	0.9884	0.9870	0.9865	0.9877	0.9859	0.9927	0.9887	0.9936	0.9918	0.9896	0.9911	0.9887	0.9842	0.9889	0.0073	0.0073
9	1.1905	1.1710	1.1928	1.1822	1.1900	1.1832	1.1852	1.1872	1.1882	1.1843	1.1802	1.1838	1.1799	1.1901	1.1910	1.1887	1.1885	1.1843	1.1852	1.1904	1.1799	1.1856	0.0103	0.0087
10	1.0735	1.0602	1.0803	1.0695	1.0738	1.0671	1.0698	1.0710	1.0733	1.0689	1.0667	1.0706	1.0658	1.0751	1.0734	1.0744	1.0740	1.0703	1.0714	1.0732	1.0662	1.0709	0.0085	0.0080
11	0.9598	0.9517	0.9683	0.9567	0.9593	0.9529	0.9564	0.9582	0.9588	0.9579	0.9537	0.9579	0.9560	0.9621	0.9580	0.9622	0.9617	0.9580	0.9604	0.9589	0.9521	0.9581	0.0077	0.0080
12	1.2000	1.1820	1.2038	1.1944	1.2029	1.1960	1.1964	1.1988	1.2007	1.1947	1.1895	1.1959	1.1890	1.2021	1.2001	1.1997	1.1997	1.1959	1.1978	1.1996	1.1903	1.1966	0.0107	0.0089
13	1.0967	1.0840	1.1042	1.0933	1.0983	1.0909	1.0941	1.0965	1.0973	1.0929	1.0900	1.0942	1.0907	1.0992	1.0970	1.0985	1.0983	1.0947	1.0956	1.0964	1.0885	1.0948	0.0088	0.0080
14	0.9994	0.9911	1.0082	0.9966	1.0001	0.9927	0.9971	0.9982	0.9982	0.9975	0.9952	0.9973	0.9948	1.0077	0.9983	1.0021	1.0008	0.9994	1.0011	0.9993	0.9912	0.9984	0.0088	0.0088
15	1.1007	1.0872	1.1035	1.0983	1.1015	1.0966	1.0991	1.1005	1.0964	1.0989	1.0971	1.1000	1.0941	1.1066	1.1047	1.1062	1.1045	1.1002	1.1021	1.1068	1.0999	1.1002	0.0092	0.0084
16	1.1057	1.0897	1.1068	1.1020	1.1055	1.1004	1.1031	1.1050	1.1018	1.1029	1.1004	1.1044	1.0985	1.1079	1.1083	1.1089	1.1069	1.1080	1.1079	1.1087	1.1024	1.1041	0.0090	0.0082
17	1.2609	1.2374	1.2579	1.2523	1.2643	1.2530	1.2564	1.2584	1.2616	1.2539	1.2484	1.2555	1.2480	1.2625	1.2638	1.2600	1.2590	1.2577	1.2580	1.2641	1.2510	1.2564	0.0130	0.0104
18	1.1684	1.1479	1.1673	1.1630	1.1703	1.1627	1.1629	1.1661	1.1673	1.1616	1.1593	1.1656	1.1592	1.1718	1.1707	1.1689	1.1676	1.1666	1.1669	1.1713	1.1623	1.1651	0.0109	0.0093
19	1.0737	1.0571	1.0738	1.0695	1.0753	1.0673	1.0696	1.0724	1.0726	1.0705	1.0685	1.0714	1.0685	1.0760	1.0746	1.0755	1.0734	1.0757	1.0746	1.0760	1.0698	1.0717	0.0087	0.0081
20	1.2733	1.2499	1.2705	1.2663	1.2766	1.2662	1.2689	1.2716	1.2714	1.2662	1.2615	1.2687	1.2606	1.2738	1.2763	1.2728	1.2711	1.2701	1.2708	1.2759	1.2644	1.2689	0.0124	0.0098
21	1.1949	1.1731	1.1928	1.1885	1.1975	1.1889	1.1895	1.1922	1.1926	1.1900	1.1850	1.1910	1.1852	1.1958	1.1949	1.1946	1.1932	1.1927	1.1935	1.1968	1.1890	1.1910	0.0107	0.0090
22	1.1154	1.0982	1.1158	1.1116	1.1174	1.1095	1.1119	1.1142	1.1155	1.1128	1.1099	1.1141	1.1096	1.1231	1.1168	1.1178	1.1168	1.1172	1.1153	1.1163	1.1126	1.1139	0.0097	0.0087
2σ *1	0.0013	---	---	---	0.0012	0.0017	0.0009	0.0007	0.0026	0.0014	0.0010	0.0011	0.0010	0.0009	0.002	---	0.0010	0.0010	0.0015	0.0015	0.0015	---	---	---

*1 A typical value for twice the standard deviation of Monte Carlo calculation

*2 $\bar{k}_{eff,i} = \frac{1}{N}\sum_{x=A}^{U} k_{eff,i}^{(x)}$, $\sigma_i = \sqrt{\sum_{x=A}^{U}\left(k_{eff,i}^{(x)} - \bar{k}_{eff,i}\right)^2 \Big/ (N-1)}$, N=21 (the number of participants)

Table 4.2a Relative deviations in % from the average of k_{eff} obtained by all the participants

Case #	a	b	c	d	e	f	g	h	i	j	k	l	m	n	o	p	q	r	s	t	u
1	0.58	-1.60	-0.17	-0.48	0.94	0.11	0.26	0.37	0.63	-0.36	-0.89	-0.29	-0.91	0.46	0.76	0.20	0.06	0.19	0.01	0.92	-0.79
2	0.03	-1.05	0.68	-0.10	0.28	-0.19	0.07	0.00	-0.13	-0.18	-0.50	-0.19	-0.54	0.37	0.52	0.42	0.35	-0.14	0.14	0.47	-0.33
3	0.24	-0.90	0.87	-0.15	0.28	-0.28	0.00	0.07	-0.22	-0.53	-0.29	-0.23	-0.54	0.29	0.46	0.48	0.37	-0.18	0.22	0.39	-0.36
4	0.07	-0.72	1.03	-0.20	0.11	-0.20	-0.01	-0.08	-0.29	-0.23	-0.29	-0.14	-0.60	0.37	0.14	0.50	0.23	0.04	0.37	0.12	-0.23
5	0.29	-1.03	0.69	-0.12	0.36	-0.25	0.06	0.17	-0.08	-0.48	-0.39	-0.18	-0.62	0.25	0.51	0.43	0.31	-0.28	0.22	0.68	-0.55
6	0.22	-0.88	0.85	-0.18	-0.09	-0.44	-0.03	0.02	-0.45	-0.22	-0.19	-0.21	-0.48	0.41	0.36	0.47	0.40	-0.15	0.48	0.43	-0.33
7	-0.02	-0.59	1.08	-0.20	0.08	-0.40	0.01	0.05	-0.22	-0.27	-0.47	-0.13	-0.31	0.16	0.31	0.60	0.52	0.08	0.10	0.04	-0.47
8	0.18	-0.62	1.08	-0.16	-0.11	-0.35	-0.09	-0.01	-0.05	-0.19	-0.24	-0.12	-0.30	0.39	-0.02	0.48	0.30	0.08	0.23	-0.02	-0.47
9	0.42	-1.23	0.61	-0.28	0.38	-0.20	-0.03	0.14	0.22	-0.11	-0.45	-0.15	-0.48	0.38	0.46	0.27	0.25	-0.11	-0.03	0.41	-0.48
10	0.24	-1.00	0.88	-0.13	0.27	-0.35	-0.10	0.01	0.23	-0.18	-0.39	-0.03	-0.47	0.39	0.24	0.33	0.29	-0.05	0.05	0.22	-0.44
11	0.17	-0.67	1.06	-0.15	0.12	-0.55	-0.18	0.01	0.07	-0.03	-0.46	-0.03	-0.22	0.41	-0.01	0.42	0.37	-0.01	0.24	0.08	-0.63
12	0.28	-1.22	0.60	-0.19	0.52	-0.05	-0.02	0.18	0.34	-0.16	-0.60	-0.06	-0.64	0.46	0.29	0.26	0.26	-0.06	0.10	0.25	-0.53
13	0.17	-0.99	0.86	-0.14	0.32	-0.36	-0.07	0.15	0.23	-0.18	-0.44	-0.06	-0.38	0.40	0.20	0.34	0.32	-0.01	0.07	0.14	-0.58
14	0.10	-0.73	0.98	-0.18	0.17	-0.57	-0.13	-0.02	-0.02	-0.09	-0.32	-0.11	-0.36	0.93	-0.01	0.37	0.24	0.10	0.27	0.09	-0.72
15	0.04	-1.18	0.30	-0.18	0.12	-0.33	-0.10	0.02	-0.35	-0.12	-0.28	-0.02	-0.56	0.58	0.41	0.54	0.39	0.00	0.17	0.60	-0.03
16	0.15	-1.30	0.25	-0.19	0.13	-0.33	-0.09	0.09	-0.20	-0.10	-0.33	0.03	-0.50	0.35	0.38	0.44	0.26	0.36	0.35	0.42	-0.15
17	0.36	-1.51	0.12	-0.33	0.63	-0.27	0.00	0.16	0.42	-0.20	-0.64	-0.07	-0.67	0.49	0.59	0.29	0.21	0.10	0.13	0.61	-0.43
18	0.28	-1.48	0.19	-0.18	0.44	-0.21	-0.19	0.08	0.19	-0.30	-0.50	0.04	-0.51	0.57	0.48	0.32	0.21	0.13	0.15	0.53	-0.24
19	0.19	-1.36	0.20	-0.21	0.34	-0.41	-0.20	0.06	0.08	-0.11	-0.30	-0.03	-0.30	0.40	0.27	0.35	0.16	0.37	0.27	0.40	-0.18
20	0.35	-1.50	0.13	-0.20	0.61	-0.21	0.00	0.21	0.20	-0.21	-0.58	-0.02	-0.65	0.39	0.58	0.31	0.17	0.09	0.15	0.55	-0.35
21	0.32	-1.51	0.15	-0.21	0.54	-0.18	-0.13	0.10	0.13	-0.09	-0.51	0.00	-0.49	0.40	0.32	0.30	0.18	0.14	0.21	0.48	-0.17
22	0.14	-1.41	0.17	-0.21	0.31	-0.39	-0.18	0.03	0.14	-0.10	-0.36	0.02	-0.39	0.83	0.26	0.35	0.26	0.30	0.13	0.22	-0.12
$\bar{\delta}(x)^{*1}$	0.22	-1.11	0.57	-0.20	0.31	-0.29	-0.05	0.08	0.04	-0.20	-0.43	-0.09	-0.50	0.44	0.34	0.39	0.28	0.05	0.18	0.37	-0.39
$2\sigma^{(x)*1}$	0.28	0.65	0.79	0.16	0.49	0.30	0.21	0.20	0.54	0.25	0.32	0.18	0.31	0.34	0.41	0.20	0.20	0.34	0.24	0.48	0.40

*1 $\bar{\delta} = \frac{1}{n}\sum_{i=1}^{n}\delta_i^{(x)}$, $\delta_i^{(x)} = 100\left(\frac{k_{eff,i}^{(x)}}{\bar{k}_{eff,i}} - 1\right)$, $\sigma^{(x)} = \sqrt{\sum_{i=1}^{n}\left(\delta_i^{(x)} - \bar{\delta}^{(x)}\right)^2 \Big/ (n-1)}$, $n = 22$ (the number of cases)

Table 4.2b Correlation coefficients between each pair of relative deviation results shown in **Table 4.2a**

	a	b	c	d	e	f	g	h	i	j	k	l	m	n	o	p	q	r	s	t	u
a	1.00	-0.62	-0.54	-0.67	0.76	0.61	0.40	0.81	0.73	-0.26	-0.63	-0.22	-0.60	-0.06	0.56	-0.79	-0.68	-0.01	-0.39	0.62	-0.36
b	-0.62	1.00	0.97	0.53	-0.79	-0.56	-0.09	-0.64	-0.59	-0.04	0.56	-0.30	0.56	-0.20	-0.69	0.71	0.71	-0.42	0.34	-0.74	-0.29
c	-0.54	0.97	1.00	0.59	-0.73	-0.51	-0.11	-0.61	-0.50	-0.05	0.53	-0.28	0.55	-0.23	-0.68	0.58	0.68	-0.53	0.25	-0.77	-0.33
d	-0.67	0.53	0.59	1.00	-0.64	-0.53	-0.48	-0.65	-0.58	0.06	0.66	0.27	0.56	-0.12	-0.50	0.52	0.59	-0.38	0.34	-0.51	0.26
e	0.76	-0.79	-0.73	-0.64	1.00	0.72	0.43	0.83	0.82	-0.21	-0.87	-0.04	-0.69	0.06	0.68	-0.81	-0.72	0.16	-0.59	0.67	-0.24
f	0.61	-0.56	-0.51	-0.53	0.72	1.00	0.67	0.67	0.48	-0.38	-0.68	-0.30	-0.84	-0.25	0.68	-0.50	-0.52	-0.07	-0.44	0.61	-0.08
g	0.40	-0.09	-0.11	-0.48	0.43	0.67	1.00	0.59	0.19	-0.58	-0.55	-0.79	-0.75	-0.38	0.63	-0.12	-0.11	-0.35	-0.27	0.50	-0.45
h	0.81	-0.64	-0.61	-0.65	0.83	0.67	0.59	1.00	0.70	-0.33	-0.79	-0.21	-0.66	-0.16	0.70	-0.65	-0.51	0.05	-0.52	0.67	-0.40
i	0.73	-0.59	-0.50	-0.58	0.82	0.48	0.19	0.70	1.00	0.06	-0.81	0.11	-0.35	0.19	0.30	-0.90	-0.68	0.25	-0.70	0.29	-0.45
j	-0.26	-0.04	-0.05	0.06	-0.21	-0.38	-0.58	-0.33	0.06	1.00	0.12	0.60	0.47	0.41	-0.49	-0.14	-0.07	0.45	0.05	-0.38	0.21
k	-0.63	0.56	0.53	0.66	-0.87	-0.68	-0.55	-0.79	-0.81	0.12	1.00	0.12	0.61	0.05	-0.58	0.68	0.50	-0.11	0.65	-0.50	0.45
l	-0.22	-0.30	-0.28	0.27	-0.04	-0.30	-0.79	-0.21	0.11	0.60	0.12	1.00	0.37	0.31	-0.29	-0.11	-0.12	0.53	0.01	-0.22	0.51
m	-0.60	0.56	0.55	0.56	-0.69	-0.84	-0.75	-0.66	-0.35	0.47	0.61	0.37	1.00	0.07	-0.79	0.42	0.49	0.15	0.23	-0.78	0.10
n	-0.06	-0.20	-0.23	-0.12	0.06	-0.25	-0.38	-0.16	0.19	0.41	0.05	0.31	0.07	1.00	-0.24	-0.29	-0.31	0.35	0.00	-0.07	0.02
o	0.56	-0.69	-0.68	-0.50	0.68	0.68	0.63	0.70	0.30	-0.49	-0.58	-0.29	-0.79	-0.24	1.00	-0.34	-0.30	-0.09	-0.35	0.90	0.07
p	-0.79	0.71	0.58	0.52	-0.81	-0.50	-0.12	-0.65	-0.90	-0.14	0.68	-0.11	0.42	-0.29	-0.34	1.00	0.79	-0.20	0.51	-0.40	0.31
q	-0.68	0.71	0.68	0.59	-0.72	-0.52	-0.11	-0.51	-0.68	-0.07	0.50	-0.12	0.49	-0.31	-0.30	0.79	1.00	-0.49	0.17	-0.46	0.07
r	-0.01	-0.42	-0.53	-0.38	0.16	-0.07	-0.35	0.05	0.25	0.45	-0.11	0.53	0.15	0.35	-0.09	-0.20	-0.49	1.00	0.07	-0.03	0.32
s	-0.39	0.34	0.25	0.34	-0.59	-0.44	-0.27	-0.52	-0.70	0.05	0.65	0.01	0.23	0.00	-0.35	0.51	0.17	0.07	1.00	-0.18	0.34
t	0.62	-0.74	-0.77	-0.51	0.67	0.61	0.50	0.67	0.29	-0.38	-0.50	-0.22	-0.78	-0.07	0.90	-0.40	-0.46	-0.03	-0.18	1.00	0.09
u	-0.36	-0.29	-0.33	0.26	-0.24	-0.08	-0.45	-0.40	-0.45	0.21	0.45	0.51	0.10	0.02	0.07	0.31	0.07	0.32	0.34	0.09	1.00

Table 4.3 Results of the regionwise fission rates in %

Case #	Region #	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	Ave
1	1	0.2	0.1	0.2	0.2	0.2	0.2	0.1	0.2	0.2	0.2	0.2	0.2	0.2	0.1	0.2	0.2	0.2	0.2	0.2	0.1	0.2	0.2
	2	8.5	5.7	5.8	6.4	7.5	6.0	4.8	6.4	8.4	6.4	6.3	6.0	6.4	6.5	5.4	5.8	6.1	6.4	6.0	5.2	6.1	6.3
	3	53.2	44.1	44.1	42.0	50.4	44.3	40.2	44.9	42.6	47.3	47.4	44.0	45.4	56.7	44.5	44.1	46.4	43.2	43.1	48.4	42.4	45.7
	4	28.9	36.2	36.2	36.1	30.7	34.8	39.5	35.2	31.1	34.9	35.0	35.2	34.4	31.3	36.2	36.2	34.8	35.5	35.8	34.6	36.8	34.7
	5	5.3	7.9	7.9	8.6	6.1	8.6	8.9	7.7	9.2	6.6	6.6	8.1	7.6	4.1	8.0	7.9	7.2	8.5	8.3	7.0	8.0	7.5
	6	1.8	2.8	2.8	3.1	2.2	2.9	3.1	2.7	3.7	2.1	2.2	3.0	2.8	0.9	2.7	2.8	2.5	2.9	3.1	2.2	3.0	2.6
	7	1.2	1.9	1.9	2.2	1.8	2.0	2.1	1.8	2.7	1.5	1.5	2.1	2.0	0.3	1.9	1.9	1.7	2.0	2.2	1.5	2.1	1.8
	8	0.7	1.0	1.1	1.2	1.0	1.1	1.1	1.0	1.6	0.9	0.8	1.2	1.2	0.1	1.1	1.1	1.0	1.1	1.3	0.9	1.2	1.0
	9	0.1	0.1	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.1	0.1	0.2	0.2	0.0	0.2	0.2	0.1	0.2	0.2	0.1	0.2	0.2
5	1	0.1	0.0	0.0	0.0	0.1	0.1	0.0	0.1	0.2	0.1	0.0	0.1	0.1	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.1	0.1
	2	1.4	0.1	0.2	0.7	1.9	2.1	0.6	1.6	5.2	2.5	0.2	1.8	1.7	0.3	0.9	0.2	0.3	3.4	0.3	0.2	1.3	1.3
	3	8.3	3.7	4.5	6.8	11.0	7.8	4.9	7.4	20.1	12.5	4.2	9.8	10.5	4.5	6.2	3.8	5.0	13.1	4.1	3.2	6.4	7.5
	4	29.8	26.5	28.1	28.3	26.1	25.6	27.2	25.7	26.2	29.8	27.4	27.3	28.1	35.8	25.7	26.5	28.7	26.1	26.7	23.5	26.1	27.4
	5	21.9	24.5	24.2	23.3	22.0	22.2	23.7	22.9	17.7	20.5	23.6	21.6	21.4	23.9	23.4	24.4	23.7	19.9	24.6	25.4	22.3	22.7
	6	13.5	15.7	15.0	14.3	13.4	14.7	15.1	14.6	10.8	12.3	15.3	13.5	13.1	13.1	14.9	15.6	14.7	12.8	15.8	16.4	14.5	14.2
	7	13.2	15.6	14.7	14.0	13.2	14.5	15.0	14.5	10.4	11.7	15.2	13.5	13.1	12.1	15.1	15.6	14.5	12.9	15.2	16.5	15.2	14.1
	8	9.8	11.5	10.9	10.3	10.3	10.9	11.1	10.9	7.8	8.7	11.5	10.2	9.8	8.5	11.4	11.6	10.8	9.7	11.1	12.3	11.8	10.5
	9	2.1	2.4	2.3	2.3	2.1	2.2	2.4	2.2	1.7	1.9	2.5	2.2	2.1	1.7	2.3	2.4	2.3	2.0	2.2	2.5	2.4	2.2
6	1	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.0	0.2	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.0	0.0
	2	0.5	0.0	0.0	0.0	1.6	0.4	0.1	0.6	3.1	1.5	0.0	0.7	0.7	0.0	0.4	0.0	0.0	1.0	0.0	0.0	0.0	0.5
	3	2.3	0.5	0.6	0.9	4.4	1.1	1.0	2.1	7.8	3.9	0.4	3.0	2.4	0.3	1.2	0.5	0.5	2.5	0.3	0.6	0.5	1.8
	4	13.1	13.2	14.1	14.8	16.8	14.3	14.3	14.3	16.8	17.6	12.6	15.9	14.9	13.3	13.4	13.1	13.5	15.1	12.7	14.4	12.3	14.3
	5	21.5	22.6	22.9	22.9	20.4	22.5	22.4	21.7	19.9	22.1	22.6	21.6	21.6	23.0	21.7	22.4	22.7	21.6	23.0	22.4	22.1	22.1
	6	18.0	18.8	18.6	18.3	16.6	18.1	18.3	17.9	15.8	16.7	19.0	17.3	17.8	18.9	18.4	18.8	18.8	17.8	18.3	18.4	19.1	18.1
	7	21.6	22.1	21.5	21.0	19.8	21.3	21.3	21.2	17.9	18.8	22.2	20.2	20.8	21.8	21.9	22.1	21.9	20.8	22.2	21.5	22.6	21.2
	8	18.4	18.5	18.0	17.6	16.5	18.2	18.1	18.0	14.9	15.3	18.7	17.1	17.5	18.5	18.7	18.7	18.2	17.2	19.1	18.5	19.1	17.8
	9	4.6	4.3	4.3	4.3	3.8	4.1	4.4	4.1	3.6	4.1	4.5	4.1	4.2	4.2	4.3	4.3	4.4	4.0	4.5	4.2	4.3	4.2

Table 4.3 Results of the regionwise fission rates in % (cont'd)

Case #	Region #	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	Ave
7	1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.2	0.1	0.1
	2	0.5	0.0	0.0	0.0	0.4	0.2	0.0	0.2	0.2	0.7	0.0	0.5	0.5	0.0	0.2	0.0	0.0	0.5	0.0	17.9	1.3	1.1
	3	1.5	0.1	0.1	0.2	1.5	0.4	0.3	0.7	2.4	1.5	0.2	1.5	1.4	0.1	0.6	0.1	0.1	1.1	0.2	5.0	0.4	0.9
	4	8.9	7.4	7.6	8.3	8.2	7.6	7.3	8.0	10.7	9.1	7.5	8.3	8.6	7.7	7.4	7.2	7.3	8.4	8.2	6.3	8.0	8.0
	5	18.1	18.2	18.4	18.8	16.5	18.1	17.8	17.7	16.6	18.4	18.3	18.1	18.1	18.5	18.0	17.9	18.0	18.6	18.5	13.7	17.6	17.8
	6	18.0	18.8	18.8	18.6	17.7	18.8	18.5	18.3	17.2	17.8	18.3	18.1	18.2	18.7	18.5	18.7	18.7	18.5	18.6	14.1	18.2	18.1
	7	23.7	25.1	24.9	24.3	24.7	24.6	24.9	24.6	23.5	23.5	24.8	24.0	24.1	25.0	24.7	25.2	25.1	24.0	24.7	18.9	24.5	24.2
	8	22.9	24.0	23.8	23.3	24.6	23.9	24.3	24.1	23.0	22.6	24.1	23.0	22.7	25.3	24.3	24.4	24.0	22.8	23.5	18.2	23.5	23.4
	9	6.4	6.4	6.5	6.5	6.4	6.3	6.8	6.4	6.5	6.3	6.8	6.4	6.3	4.7	6.4	6.5	6.8	6.0	6.3	4.7	6.4	6.3
14	1	0.3	0.2	0.3	0.3	0.4	0.4	0.3	0.3	0.2	0.3	0.2	0.3	0.3	0.1	0.3	0.2	0.3	0.3	0.2	0.3	0.4	0.3
	2	5.6	5.9	5.9	6.6	8.5	10.3	6.7	6.3	5.6	7.2	4.1	6.1	7.2	3.5	8.0	5.9	6.2	6.6	4.4	7.9	8.3	6.5
	3	41.7	43.9	43.9	41.8	51.5	50.1	46.6	44.6	41.4	42.2	40.8	40.0	41.8	46.4	49.8	43.9	45.3	46.7	43.6	53.5	50.8	45.3
	4	37.4	36.0	36.0	35.7	29.4	26.7	33.1	33.8	35.6	35.0	38.8	37.6	34.5	42.4	31.9	35.9	34.1	33.5	38.7	30.2	31.4	34.7
	5	7.7	7.9	7.9	8.6	5.9	6.7	7.6	8.2	9.3	8.3	9.1	8.9	9.0	5.7	5.8	7.9	7.9	6.9	7.5	5.0	5.9	7.5
	6	3.0	2.8	2.8	3.2	1.9	2.6	2.8	3.0	3.5	3.1	3.2	3.2	3.3	1.2	1.9	2.8	2.9	2.6	2.6	1.5	1.7	2.6
	7	2.4	2.0	2.0	2.3	1.4	1.9	1.8	2.2	2.5	2.2	2.3	2.3	2.3	0.5	1.3	2.0	2.0	1.9	1.9	1.0	1.0	1.9
	8	1.5	1.1	1.1	1.3	0.8	1.1	1.0	1.3	1.5	1.3	1.3	1.3	1.3	0.2	0.8	1.1	1.1	1.3	1.0	0.5	0.6	1.1
	9	0.4	0.2	0.3	0.3	0.2	0.2	0.2	0.3	0.4	0.3	0.3	0.3	0.3	0.0	0.2	0.2	0.2	0.3	0.2	0.1	0.1	0.2

Table 4.4 Differences from the average of the regionwise fission rates in %

Case #	Region #	a	b	c	d	e	f	g	h	i	j	k	l	m	n	o	p	q	r	s	t	u	max	min
1	1	0.0	-0.1	0.0	0.0	0.0	0.0	-0.1	0.0	0.0	0.0	0.0	0.0	0.0	-0.1	0.0	0.0	0.0	0.0	0.0	-0.1	0.0	0.0	-0.1
	2	2.2	-0.6	-0.5	0.1	1.2	-0.3	-1.5	0.1	2.1	0.1	0.0	-0.3	0.1	0.2	-0.9	-0.5	-0.2	0.1	-0.3	-1.1	-0.2	2.2	-1.5
	3	7.5	-1.6	-1.6	-3.7	4.7	-1.4	-5.5	-0.8	-3.1	1.6	1.7	-1.7	-0.3	11.0	-1.2	-1.6	0.7	-2.5	-2.6	2.7	-3.3	11.0	-5.5
	4	-5.8	1.5	1.5	1.4	-4.0	0.1	4.8	0.5	-3.6	0.2	0.3	0.5	-0.3	-3.4	1.5	1.5	0.1	0.8	1.1	-0.1	2.1	4.8	-5.8
	5	-2.2	0.4	0.4	1.1	-1.4	1.1	1.4	0.2	1.7	-0.9	-0.9	0.6	0.1	-3.4	0.5	0.4	-0.3	1.0	0.8	-0.5	0.5	1.7	-3.4
	6	-0.8	0.2	0.2	0.5	-0.4	0.3	0.5	0.1	1.1	-0.5	-0.4	0.4	0.2	-1.7	0.1	0.2	-0.1	0.3	0.5	-0.4	0.4	1.1	-1.7
	7	-0.6	0.1	0.1	0.4	0.0	0.2	0.3	0.0	0.9	-0.3	-0.3	0.3	0.2	-1.5	0.1	0.1	-0.1	0.2	0.4	-0.3	0.3	0.9	-1.5
	8	-0.3	0.0	0.1	0.2	0.0	0.1	0.1	0.0	0.6	-0.1	-0.2	0.2	0.2	-0.9	0.1	0.1	0.0	0.1	0.3	-0.1	0.2	0.6	-0.9
	9	-0.1	-0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-0.1	-0.1	0.0	0.0	-0.2	0.0	0.0	-0.1	0.0	0.0	-0.1	0.0	0.0	-0.2
5	1	0.0	-0.1	-0.1	-0.1	0.0	0.0	-0.1	0.0	0.1	0.0	-0.1	0.0	0.0	-0.1	-0.1	-0.1	-0.1	0.0	-0.1	-0.1	0.0	0.1	-0.1
	2	0.1	-1.2	-1.1	-0.6	0.6	0.8	-0.7	0.3	3.9	1.2	-1.1	0.5	0.4	-1.0	-0.4	-1.1	-1.0	2.1	-1.0	-1.1	0.0	3.9	-1.2
	3	0.8	-3.8	-3.0	-0.7	3.5	0.3	-2.6	-0.1	12.6	5.0	-3.3	2.3	3.0	-3.0	-1.3	-3.7	-2.5	5.6	-3.4	-4.3	-1.1	12.6	-4.3
	4	2.4	-0.9	0.7	0.9	-1.3	-1.8	-0.2	-1.7	-1.2	2.4	0.0	-0.1	0.7	8.4	-1.7	-0.9	1.3	-1.3	-0.7	-3.9	-1.3	8.4	-3.9
	5	-0.8	1.8	1.5	0.6	-0.7	-0.5	1.0	0.2	-5.0	-2.2	0.9	-1.1	-1.3	1.2	0.7	1.7	1.0	-2.8	1.9	2.7	-0.4	2.7	-5.0
	6	-0.7	1.5	0.8	0.1	-0.8	0.5	0.9	0.4	-3.4	-1.9	1.1	-0.7	-1.1	-1.1	0.7	1.4	0.5	-1.4	1.6	2.2	0.3	2.2	-3.4
	7	-0.9	1.5	0.6	-0.1	-0.9	0.4	0.9	0.4	-3.7	-2.4	1.1	-0.6	-1.0	-2.0	1.0	1.5	0.4	-1.2	1.1	2.4	1.1	2.4	-3.7
	8	-0.7	1.0	0.4	-0.2	-0.2	0.4	0.6	0.4	-2.7	-1.8	1.0	-0.3	-0.7	-2.0	0.9	1.1	0.3	-0.8	0.6	1.8	1.3	1.8	-2.7
	9	-0.1	0.2	0.1	0.1	-0.1	0.0	0.2	0.0	-0.5	-0.3	0.3	0.0	-0.1	-0.5	0.1	0.2	0.1	-0.2	0.0	0.3	0.2	0.3	-0.5
6	1	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.0	0.2	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.0	0.2	0.0
	2	0.0	-0.5	-0.5	-0.5	1.1	-0.1	-0.4	0.1	2.6	1.0	-0.5	0.2	0.2	-0.5	-0.1	-0.5	-0.5	0.5	-0.5	-0.5	-0.5	2.6	-0.5
	3	0.5	-1.3	-1.2	-0.9	2.6	-0.7	-0.8	0.3	6.0	2.1	-1.4	1.2	0.6	-1.5	-0.6	-1.3	-1.3	0.7	-1.5	-1.2	-1.3	6.0	-1.5
	4	-1.2	-1.1	-0.2	0.5	2.5	0.0	0.0	0.0	2.5	3.3	-1.7	1.6	0.6	-1.0	-0.9	-1.2	-0.8	0.8	-1.6	0.1	-2.0	3.3	-2.0
	5	-0.6	0.5	0.8	0.8	-1.7	0.4	0.3	-0.4	-2.2	0.0	0.5	-0.5	-0.5	0.9	-0.4	0.3	0.6	-0.5	0.9	0.3	0.0	0.9	-2.2
	6	-0.1	0.7	0.5	0.2	-1.5	0.0	0.2	-0.2	-2.3	-1.4	0.9	-0.8	-0.3	0.8	0.3	0.7	0.7	-0.3	0.2	0.3	1.0	1.0	-2.3
	7	0.4	0.9	0.3	-0.2	-1.4	0.1	0.1	0.0	-3.3	-2.4	1.0	-1.0	-0.4	0.6	0.7	0.9	0.7	-0.4	1.0	0.3	1.4	1.4	-3.3
	8	0.6	0.7	0.2	-0.2	-1.3	0.4	0.3	0.2	-2.9	-2.5	0.9	-0.7	-0.3	0.7	0.9	0.9	0.4	-0.6	1.3	0.7	1.3	1.3	-2.9
	9	0.4	0.1	0.1	0.1	-0.4	-0.1	0.2	-0.1	-0.6	-0.1	0.3	-0.1	0.0	0.0	0.1	0.1	0.2	-0.2	0.3	0.0	0.1	0.4	-0.6

Table 4.4 Differences from the average of the regionwise fission rates in % (cont'd)

Case #	Region #	a	b	c	d	e	f	g	h	i	j	k	l	m	n	o	p	q	r	s	t	u	max	min
7	1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	0.0	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	1.1	0.0	1.1	-0.1
	2	-0.6	-1.1	-1.1	-1.1	-0.7	-0.9	-1.1	-0.9	-0.9	-0.4	-1.1	-0.6	-0.6	-1.1	-0.9	-1.1	-1.1	-0.6	-1.1	16.8	0.2	16.8	-1.1
	3	0.6	-0.8	-0.8	-0.7	0.6	-0.5	-0.6	-0.2	1.5	0.6	-0.7	0.6	0.5	-0.8	-0.3	-0.8	-0.8	0.2	-0.7	4.1	-0.5	4.1	-0.8
	4	0.9	-0.6	-0.4	0.3	0.2	-0.4	-0.7	0.0	2.7	1.1	-0.5	0.3	0.6	-0.3	-0.6	-0.8	-0.7	0.4	0.2	-1.7	0.0	2.7	-1.7
	5	0.3	0.4	0.6	1.0	-1.3	0.3	0.0	-0.1	-1.2	0.6	0.5	0.3	0.3	0.7	0.2	0.1	0.2	0.8	0.7	-4.1	-0.2	1.0	-4.1
	6	-0.1	0.7	0.7	0.5	-0.4	0.7	0.4	0.2	-0.9	-0.3	0.2	0.0	0.1	0.6	0.4	0.6	0.6	0.4	0.5	-4.0	0.1	0.7	-4.0
	7	-0.5	0.9	0.7	0.1	0.5	0.4	0.7	0.4	-0.7	-0.7	0.6	-0.2	-0.1	0.8	0.5	1.0	0.9	-0.2	0.5	-5.3	0.3	1.0	-5.3
	8	-0.5	0.6	0.4	-0.1	1.2	0.5	0.9	0.7	-0.4	-0.8	0.7	-0.4	-0.7	1.9	0.9	1.0	0.6	-0.6	0.1	-5.2	0.1	1.9	-5.2
	9	0.1	0.1	0.2	0.2	0.1	0.0	0.5	0.1	0.2	0.0	0.5	0.1	0.0	-1.6	0.1	0.2	0.5	-0.3	0.0	-1.6	0.1	0.5	-1.6
14	1	0.0	-0.1	0.0	0.0	0.1	0.1	0.0	0.0	-0.1	0.0	-0.1	0.0	0.0	-0.2	0.0	-0.1	0.0	0.0	-0.1	0.0	0.1	0.1	-0.2
	2	-0.9	-0.6	-0.6	0.1	2.0	3.8	0.2	-0.2	-0.9	0.7	-2.4	-0.4	0.7	-3.0	1.5	-0.6	-0.3	0.1	-2.1	1.4	1.8	3.8	-3.0
	3	-3.6	-1.4	-1.4	-3.5	6.2	4.8	1.3	-0.7	-3.9	-3.1	-4.5	-5.3	-3.5	1.1	4.5	-1.4	0.0	1.4	-1.7	8.2	5.5	8.2	-5.3
	4	2.7	1.3	1.3	1.0	-5.3	-8.0	-1.6	-0.9	0.9	0.3	4.1	2.9	-0.2	7.7	-2.8	1.2	-0.6	-1.2	4.0	-4.5	-3.3	7.7	-8.0
	5	0.2	0.4	0.4	1.1	-1.6	-0.8	0.1	0.7	1.8	0.8	1.6	1.4	1.5	-1.8	-1.7	0.4	0.4	-0.6	0.0	-2.5	-1.6	1.8	-2.5
	6	0.4	0.2	0.2	0.6	-0.7	0.0	0.2	0.4	0.9	0.5	0.6	0.6	0.7	-1.4	-0.7	0.2	0.3	0.0	0.0	-1.1	-0.9	0.9	-1.4
	7	0.5	0.1	0.1	0.4	-0.5	0.0	-0.1	0.3	0.6	0.3	0.4	0.4	0.4	-1.4	-0.6	0.1	0.1	0.0	0.0	-0.9	-0.9	0.6	-1.4
	8	0.4	0.0	0.0	0.2	-0.3	0.0	-0.1	0.2	0.4	0.2	0.2	0.2	0.2	-0.9	-0.3	0.0	0.0	0.2	-0.1	-0.6	-0.5	0.4	-0.9
	9	0.2	0.0	0.1	0.1	0.0	0.0	0.0	0.1	0.2	0.1	0.1	0.1	0.1	-0.2	0.0	0.0	0.0	0.1	0.0	-0.1	-0.1	0.2	-0.2

Table 4.5 The end, void profile and cooling time effects

ID	End Effect [%]*1		Void Profile Effect [%]*2				Effect of Cooling Time [%]*3		
			FPs=Yes		FPs=No		Burnup [GWd/tU]		
	FPs=Yes (#7-#8)	FPs=No (#15-#16)	40% Unif (#8-#11)	70% Unif (#8-#14)	40% Unif. (#16-#19)	70% Unif. (#16-#22)	20 (#2-#5)	30 (#3-#6)	40 (#4-#7)
A	0.75	-0.50	3.08	-0.88	3.20	-0.97	0.94	2.04	2.97
B	0.97	-0.25	3.10	-0.84	3.26	-0.85	1.21	1.98	2.73
C	0.95	-0.33	3.12	-0.87	3.30	-0.90	1.25	2.06	2.87
D	0.90	-0.37	3.06	-0.93	3.25	-0.96	1.26	2.05	2.87
E	1.13	-0.40	2.85	-1.23	3.02	-1.19	1.16	2.43	2.91
F	0.89	-0.38	3.25	-0.73	3.31	-0.91	1.32	2.18	3.07
G	1.04	-0.40	3.16	-0.91	3.35	-0.88	1.26	2.05	2.86
H	1.00	-0.45	3.06	-0.94	3.26	-0.92	1.04	2.07	2.75
I	0.77	-0.54	2.96	-0.98	2.92	-1.37	1.19	2.27	2.80
J	0.86	-0.40	2.91	-1.05	3.24	-0.99	1.60	1.68	2.91
K	0.71	-0.33	3.28	-0.87	3.19	-0.95	1.11	1.90	3.05
L	0.93	-0.44	2.98	-0.96	3.30	-0.97	1.24	1.99	2.86
M	0.93	-0.44	2.99	-0.89	3.00	-1.11	1.34	1.94	2.57
N	0.71	-0.13	3.06	-1.50	3.19	-1.52	1.40	1.90	3.11
O	1.26	-0.36	3.07	-0.96	3.37	-0.85	1.26	2.14	2.72
P	1.06	-0.27	3.14	-0.85	3.34	-0.89	1.24	2.04	2.80
Q	1.16	-0.24	3.01	-0.90	3.35	-0.99	1.30	2.00	2.60
R	0.95	-0.78	3.16	-0.98	3.23	-0.92	1.41	1.99	2.84
S	0.82	-0.58	3.07	-1.00	3.33	-0.74	1.15	1.74	3.16
T	1.00	-0.19	2.98	-1.06	3.27	-0.76	1.01	1.99	2.96
U	0.94	-0.25	3.21	-0.70	3.26	-1.02	1.50	1.98	3.11
Ave.	0.94	-0.38	3.07	-0.95	3.24	-0.98	1.25	2.02	2.88
Max.	1.26	-0.13	3.28	-0.70	3.37	-0.74	1.60	2.43	3.16
Min.	0.71	-0.78	2.85	-1.50	2.92	-1.52	0.94	1.68	2.57
2σ	0.29	0.29	0.22	0.34	0.24	0.37	0.31	0.32	0.32

*1 The differences of k_{eff} results with and without burnup profile for the cases of 40 GWd/tU, 5 year cooling and considering the void profile.

*2 The differences of k_{eff} results with and without void profile for the cases of 40 GWd/tU, 5 year cooling and disregarding the burnup profiles.

*3 The differences of k_{eff} results with cooling time of 1 and 5 years for the cases considering fission products, burnup and void profiles.

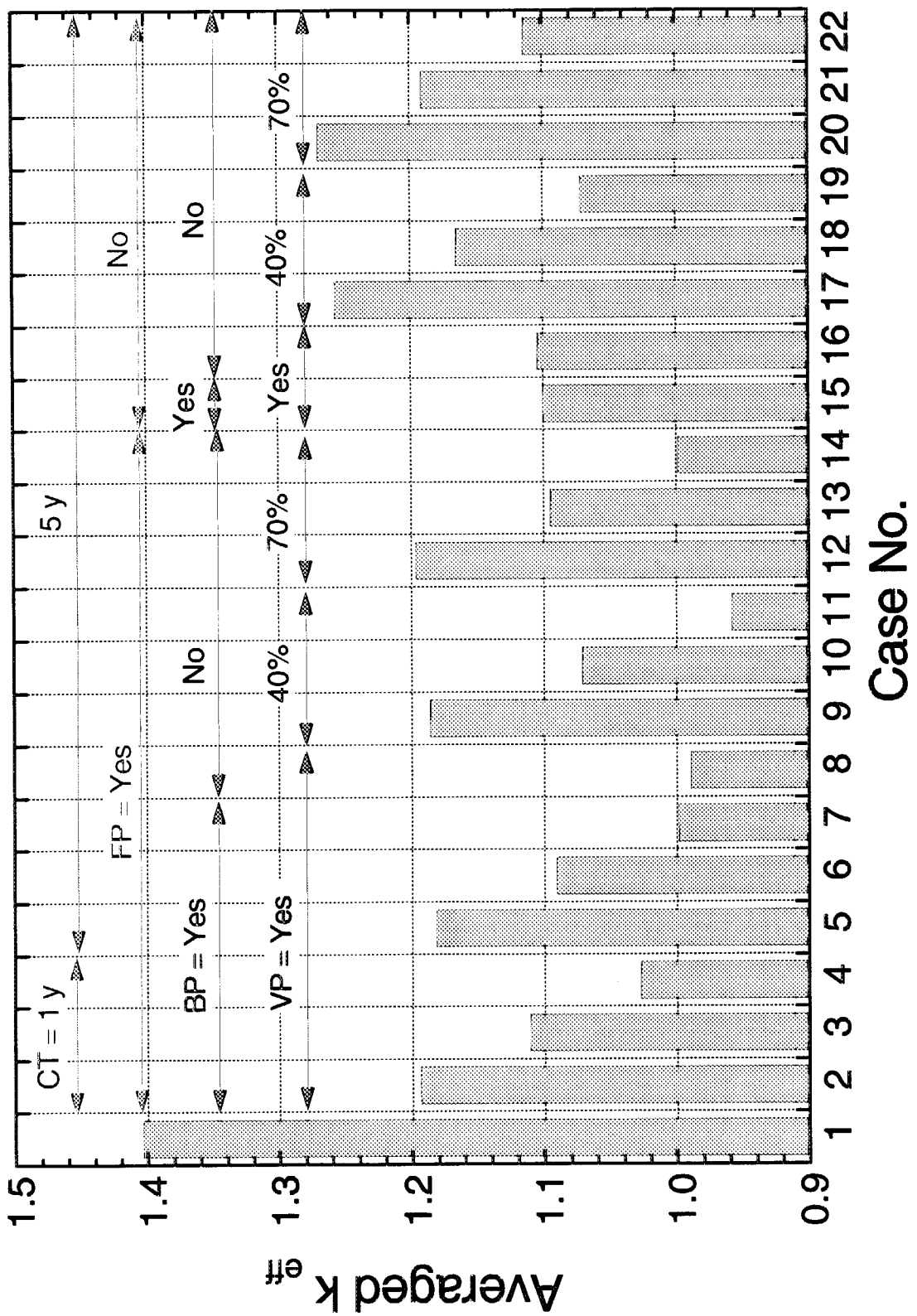

Fig. 4.1 The average of the neutron multiplication factor from participants (indicated in the text as $\bar{k}_{eff,i}$)

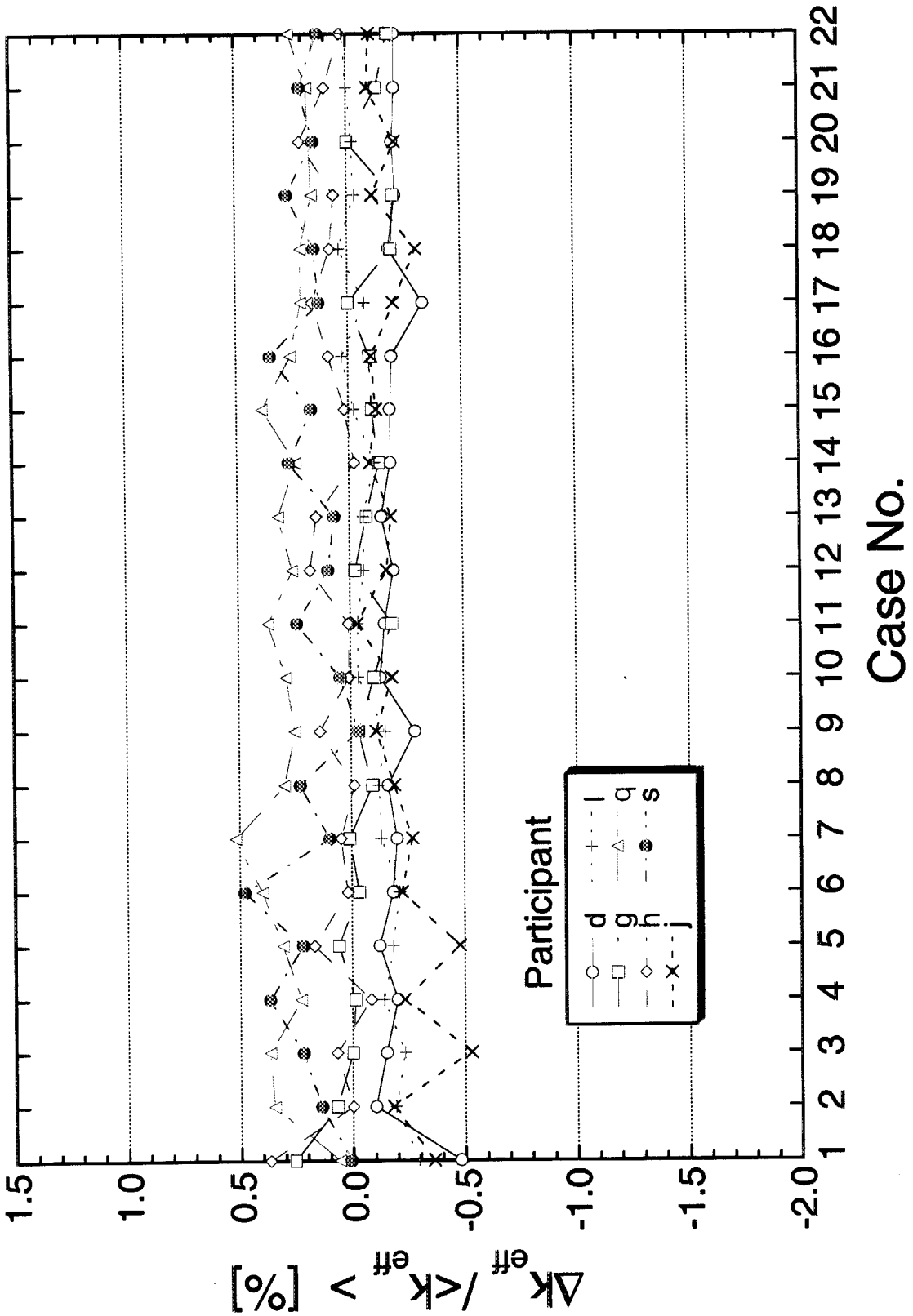

Fig. 4.2a Relative deviation of k_{eff} from the average (1)

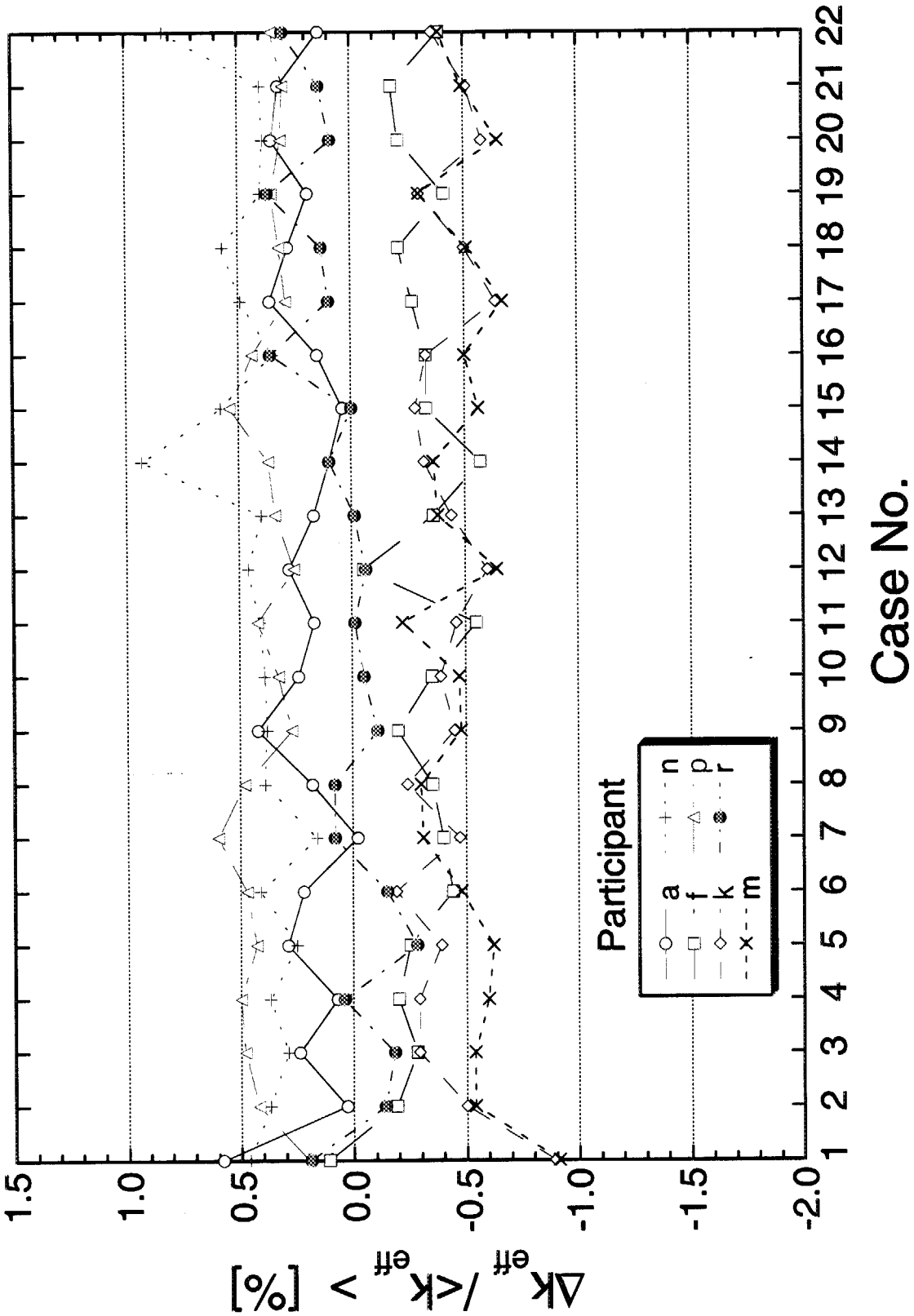

Fig. 4.2b Relative deviation of k_{eff} from the average (2)

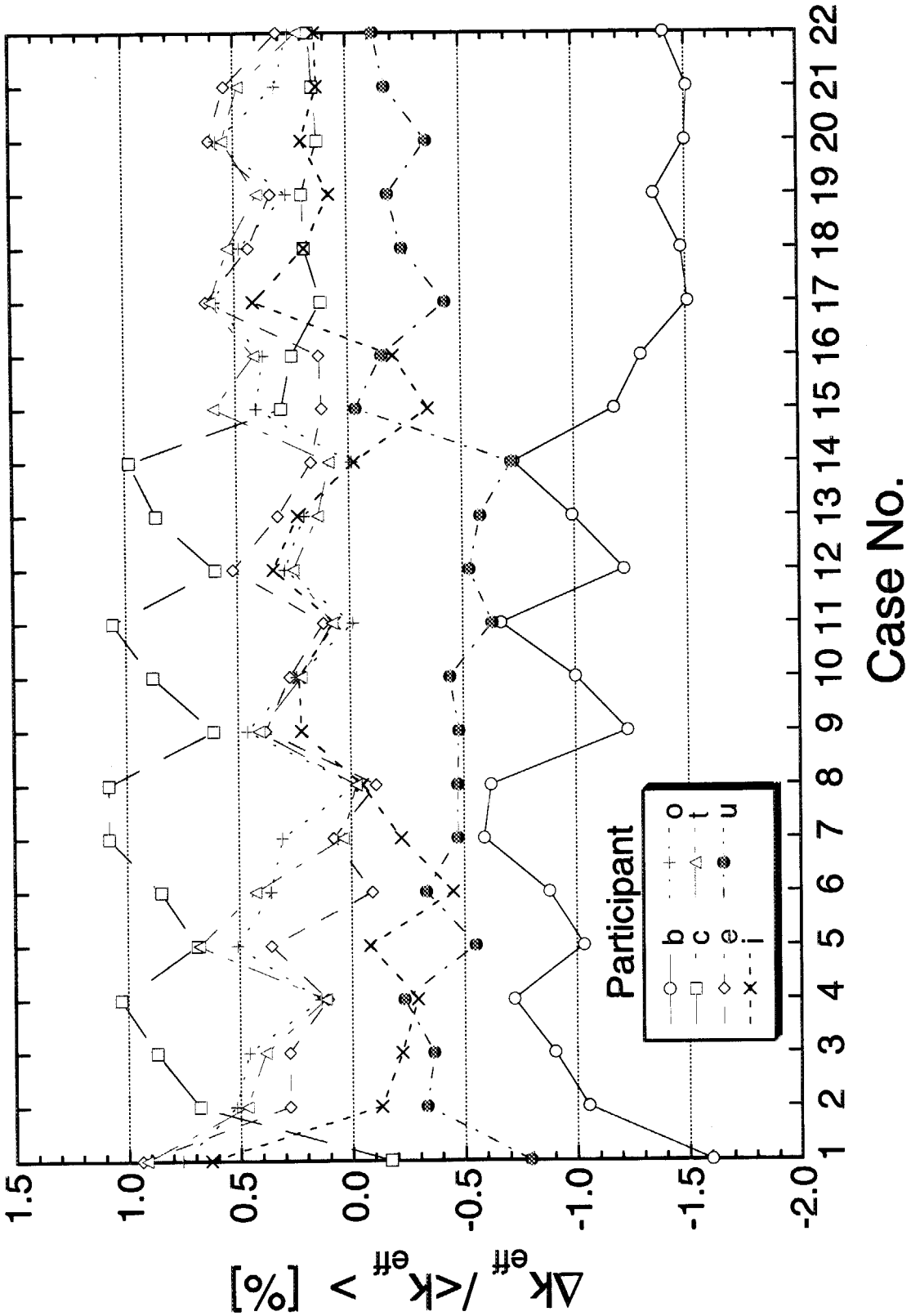

Fig. 4.2c Relative deviation of k_{eff} from the average (3)

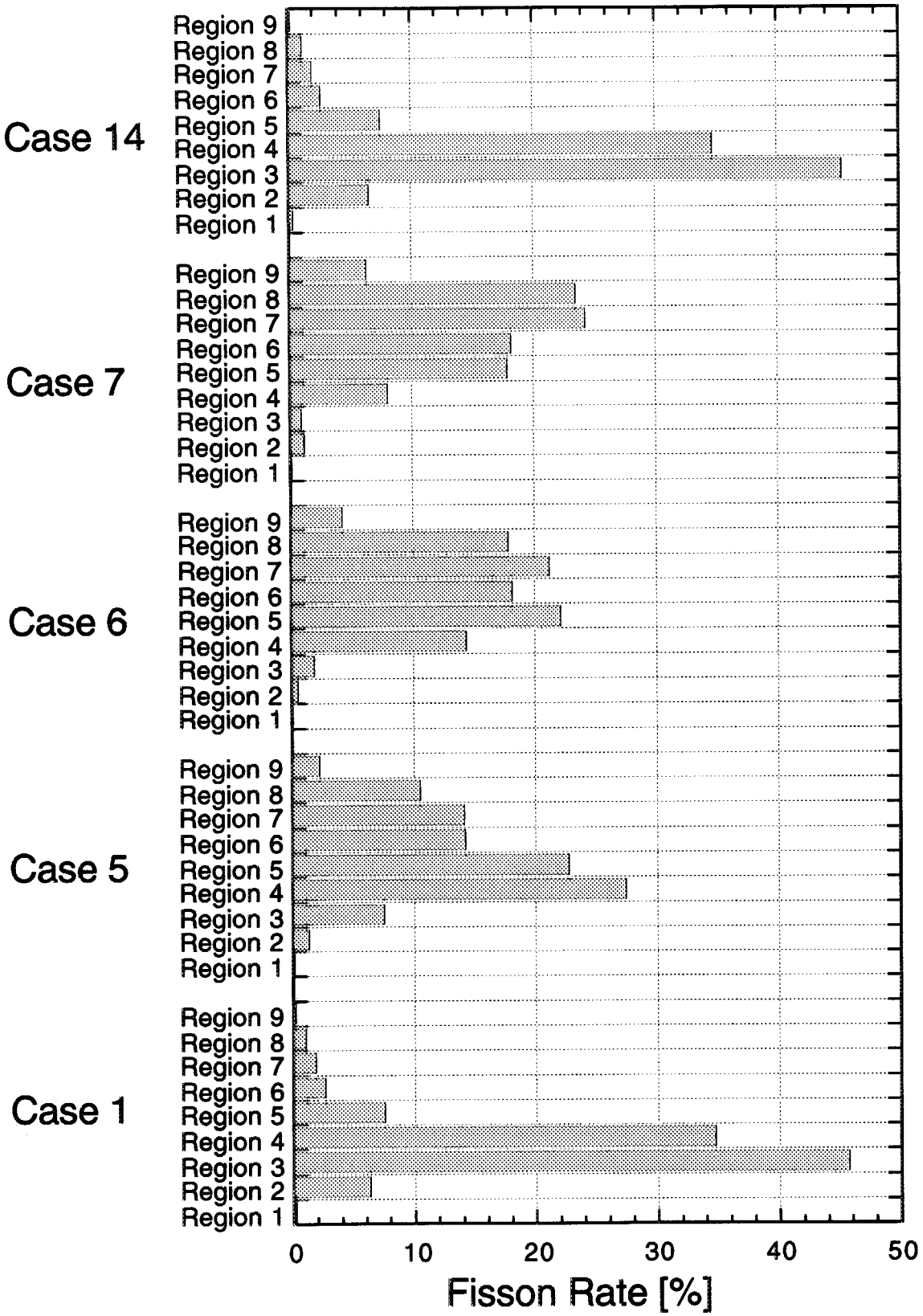

Fig 4.3a Regionwise fission rates (averaged results)

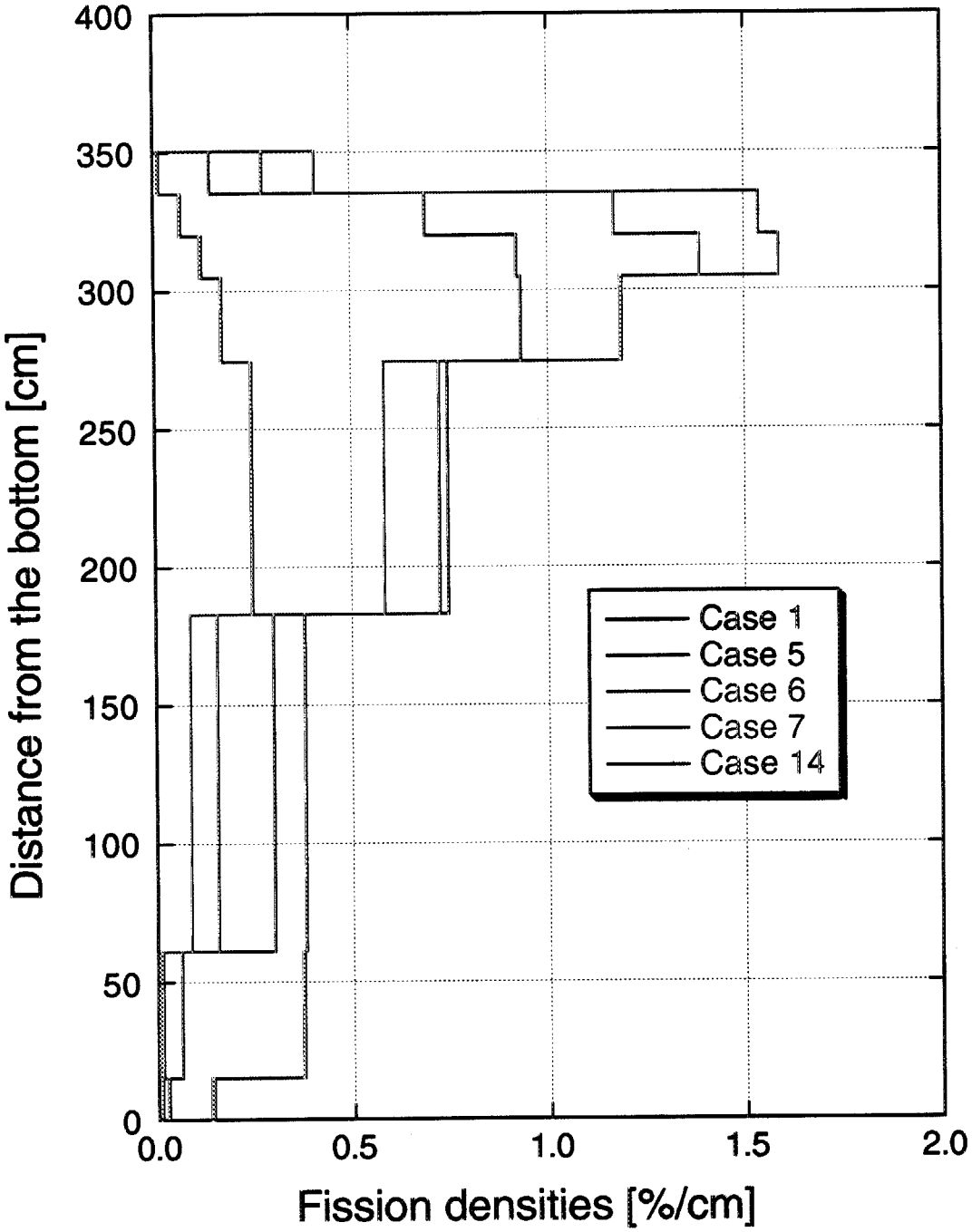

Fig 4.3b Normalized fission densities (averaged results)

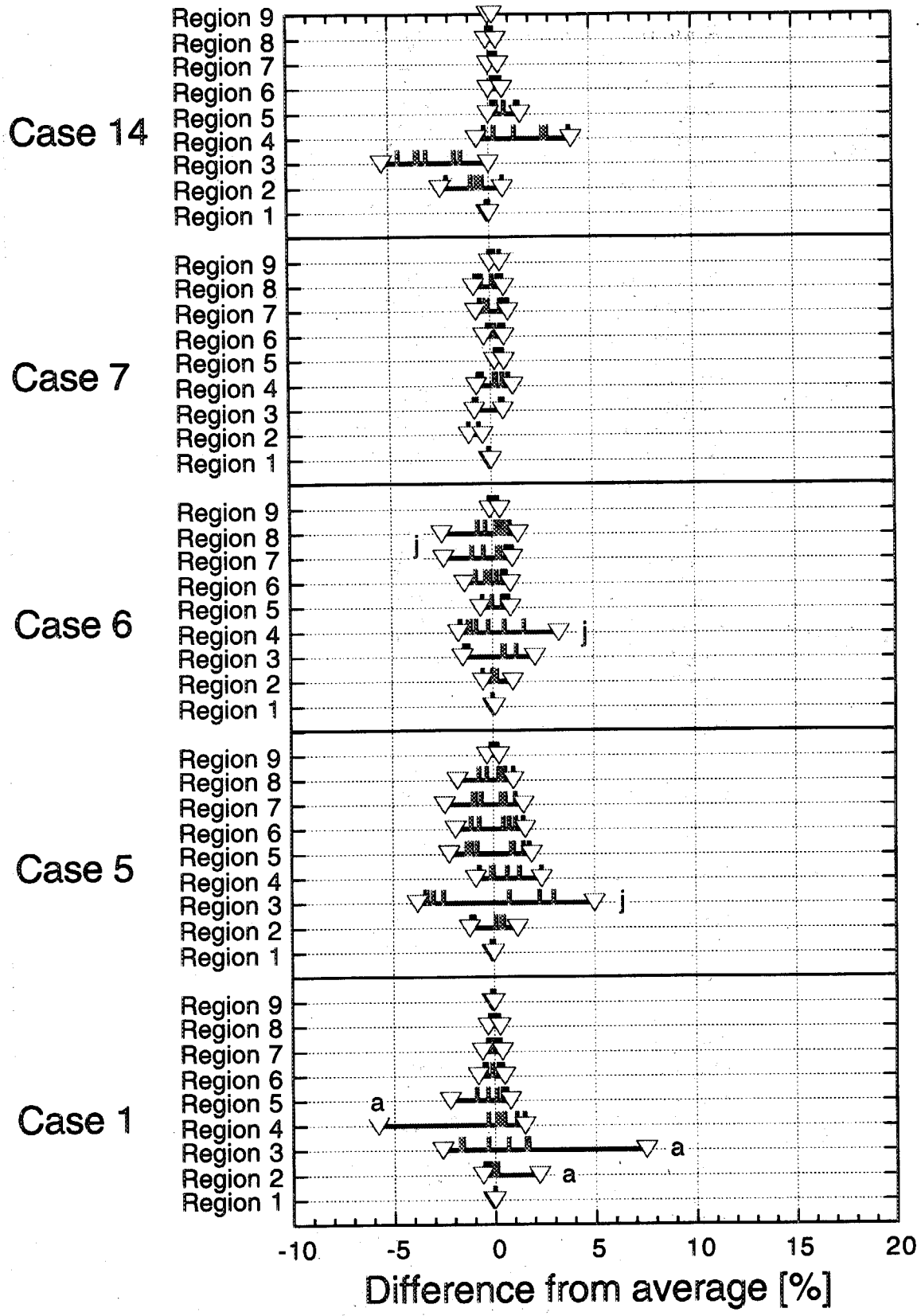

Fig.4.4a Differences from the average of regionwise fission rates (Plotted for calculations based on ENDF)

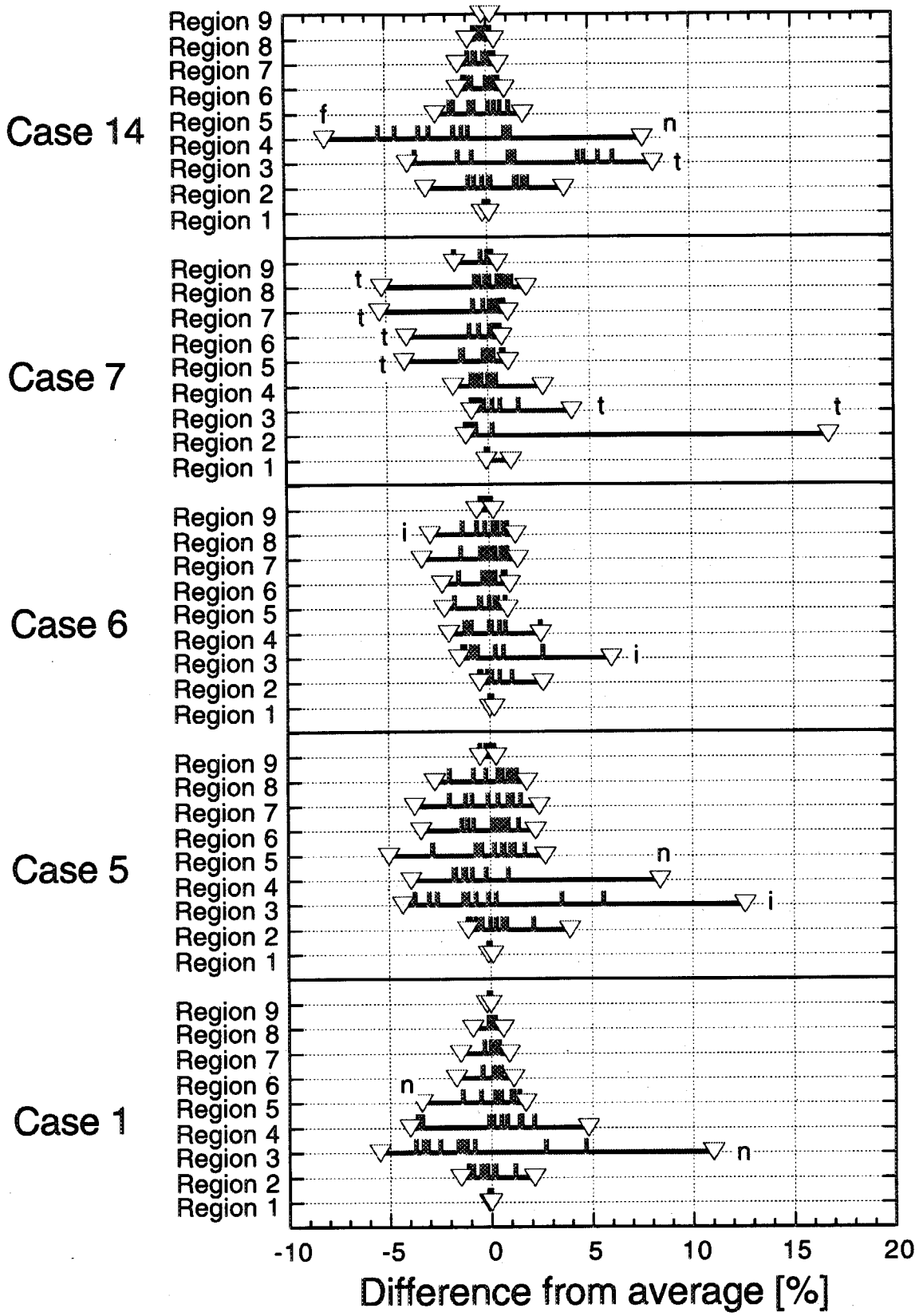

Fig.4.4b Differences from the average of regionwise fission rates
(Plotted for calculations based on JEF, JENDL and UKNDL)

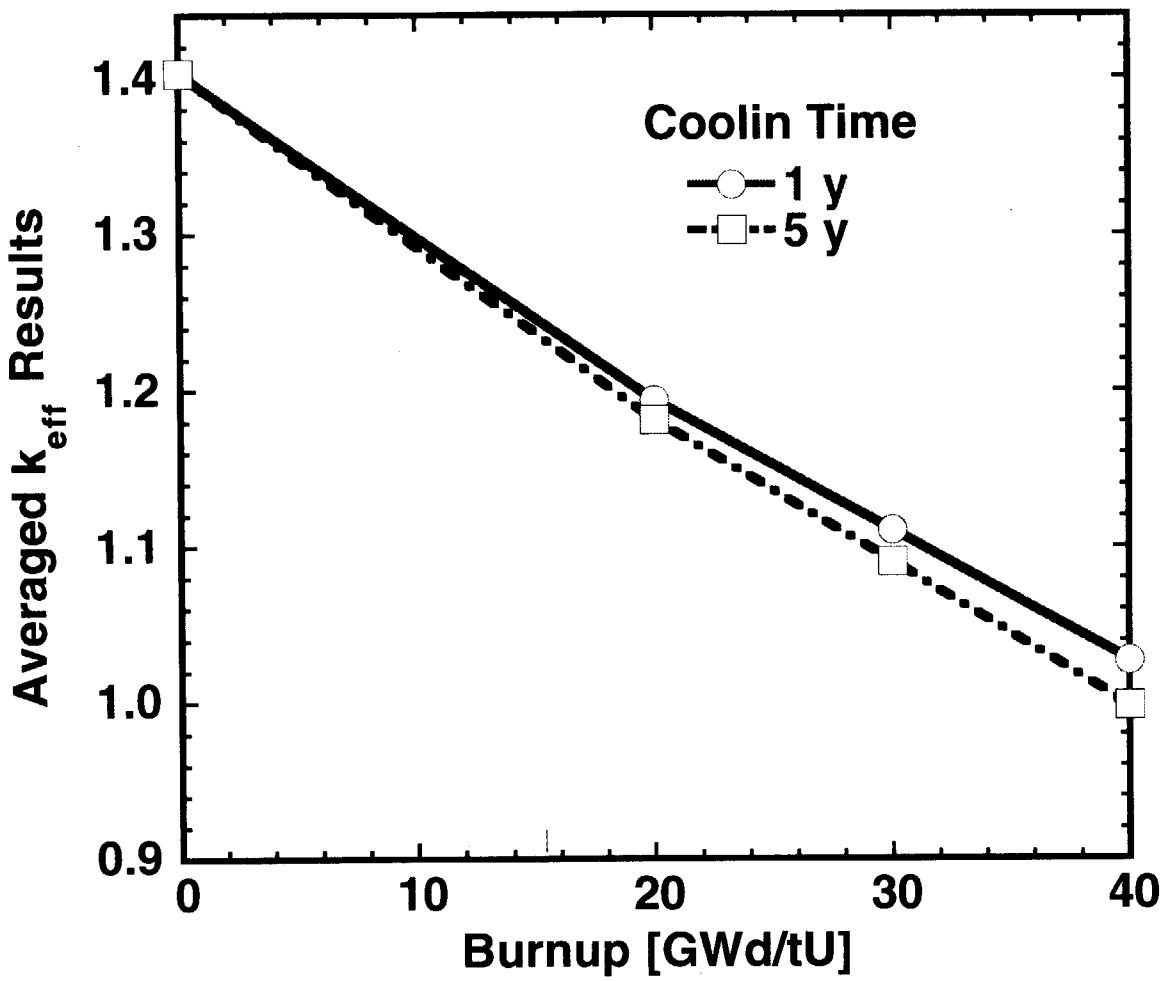

Fig. 4.5 Averaged k_{eff} results from participants (considering FPs, burnup and void profiles)

5. Concluding Remarks

The present benchmarks are intended to confirm the prediction capability of the current generation computer code and data library combinations for the neutron multiplication factor (k_{eff}) of an irradiated BWR fuel assembly model. Twenty-one sets of results from 17 institutes of 9 countries have been submitted for the 22 problems. The relative difference from the average k_{eff} calculated by the participants almost all lay within a band of ±1 %Δk/k.

Axial profiles of fractional fission rates were further requested for five cases out of the 22 problems. The deviations from the average of the calculated fission rate profiles were found to be within ±5 % for most cases. In some cases, the deviations exceeded the ±5 % band, however, the relative disperse in k_{eff} from the averaged k_{eff} remained within the ±1%Δk/k band, as stated above.

The end effect, defined as the difference of k_{eff} with and without burnup profiles, has a similar tendency as the PWR cases, however, it appeared less pronounced, i.e. up to 1 %Δk.

In addition to consideration of void profiles, constant uniform void fractions of 40 % and 70 % have also been studied. The 70% case overestimates k_{eff} relative to the cases considering void profiles when the burnup profile is disregarded. However, this conclusion is not likely valid when the burnup profile is considered.

In the framework of the present benchmarks, the k_{eff} decreases as the cooling time increases from 1 y to 5y. This tendency is consistent with the results of PWR benchmarks.

Acknowledgments

We would like to thank the participants in the international benchmark studies for their contributions to the work and for their helpful comments in the preparation of this report. Appendix VI lists the participants of the working group meetings from 1996 to 1999 with their current email addresses. We wish to thank especially M.C. Brady-Raap, the chair of the group, for conducting the meeting to make it fruitful; E. Sartori and his staff at the NEA for their contribution in organizing the exercise and arranging the meetings; and M.D. DeHart, N.T. Gulliford and T. Yamamoto for proof-reading the manuscript.

References

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5) M. Takano, "OECD/NEA Burnup Credit Criticality Benchmark - Result of Phase-1A -," *NEA/NSC/DOC*(93)22, *JAERI-M* 94-003 (1994).

6) M.D. DeHart, M.C. Brady and C.V. Parks, "OECD/NEA Burnup Credit Criticality Benchmark Phase I-B," *NEA/NSC/DOC*(96)06, *ORNL*-6901 (1996).

7) M. Takano and H. Okuno, "OECD/NEA Burnup Credit Criticality Benchmark - Result of Phase IIA -," *NEA/NSC/DOC*(96)01, *JAERI-Research* 96-003 (1996).

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Appendix I

Problem Specification for Burnup Credit Benchmark Phase IIIA: Criticality Calculations of BWR Spent Fuel in Storage and Transportation

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Problem Specification for Burnup Credit Benchmark Phase IIIA : Criticality Calculations of BWR Spent Fuel in Storage and Transportation

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December 1995

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1. Introduction

Taking reactivity credit for burned fuel in LWR is a world-wide trend in the field of nuclear criticality safety, which comes from cost-reduction needs. Burnup credit (BUC) criticality benchmark studies for PWR spent fuel have been carried out in the OECD/NEA/BUC working group as Phases I and II. BUC criticality benchmark studies for BWR spent fuel will be made in this Phase III.

The main features of BWR different from PWR in respect to criticality safety calculations are (1) moderator void distribution in a core, and (2) a complicated composition of a fuel assembly.

In BWRs, moderator voids are distributed in a core. Their volume fraction amounts to about 70% near the top region of the core, which makes the neutron energy spectrum hard and Pu production rate high compared to the bottom region where the void volume fraction is almost zero. The average void fraction over the entire core is about 40%.

A fuel assembly of BWR consists of many kinds of fuel rods whose enrichments are different from each other. Some fuel rods contain Gd, which is a strong neutron absorber. And a large water rod is located in the center.

In order to avoid unnecessary complication of the problem, simplified modeling is proposed for criticality calculations. The composition of all the fuel rods in an assembly is assumed to be the same, keeping the water rod in the center. Gadolinium is neglected in this modeling, since its reactivity worth is known to be negligibly small after the irradiation of 20 GWd/tU.

It is noted, however, that the simplified modeling is made only for the purpose of criticality calculation. As is shown in Appendix A, the number densities have been calculated in consideration of moderator void distribution, the complicated configuration of fuel in assembly and the existence of poison.

There are at least two evaluation methods for criticality safety of spent BWR fuels in storage facilities or transportation casks. One is to find the enrichment of fresh fuel to which the

reactivity worth of the burned fuel is equivalent. This method is commonly applied to PWR spent fuels. For this method, the neutron multiplication factors of fresh fuels are to be calculated, and will be surveyed in the benchmark of Phase IIIB.

In this Phase IIIA benchmark, we will seek after the 2nd evaluation method: the fuel assembly is evaluated as if it was irradiated until the average burnup under a high moderation void fraction, e.g. 70%, without consideration of burnup nor void distribution in a core. The number densities of nuclides in a fuel rod are prepared by Toshiba of Japan[1], and given in Appendix A. The participants are expected to calculate the neutron multiplication factor of an infinite array of assembly for 22 cases.

2. Parameters and Case Numbers

The multiplication factor is requested to calculate for 22 cases in total. The parameters and case numbers are shown in Table 1. For the cases from 15 to 22 FP nuclides (Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Nd-143, Nd-145, Eu-153 and Gd-155) should be disregarded from the nuclide list shown in Appendix A.

3. Geometry Data

An infinite array of fuel assemblies can be represented as a mirror-reflected assembly on its four faces. The dimensions of the assembly are specified by the following data.

A. Radial Dimensions (see Fig. 1)

Assembly Pitch	: 15.24 cm
Cell Pitch	: 1.63 cm
Fuel Rod Outer Radius	: 0.615 cm
Inner Radius	: 0.529 cm
Fuel Pellet Radius	: 0.529 cm
Fuel Rod Cladding Thickness	: 0.086 cm
Water Rod Outer Radius	: 1.6 cm
Inner Radius	: 1.5 cm

Water Rod Cladding Thickness : 0.1 cm
Channel Box Thickness : 0.254 cm
1/2 Thickness of Water Gap : 0.846 cm
Boundary Condition : Reflective

B. Axial Dimensions (see Fig. 2 and Fig. 3)

Fuel Length : 365.76 cm
Gas Plenum (upper side) : 30.0 cm
End Plug (each side) : 2.0 cm
Water Thickness (each side) : 30.0 cm
Boundary Condition : Vacuum

C. Axial Fuel Modeling (see Fig. 2)

The fuel rod is divided into 9 regions, which are numbered from bottom to top:

Region 1 (Bottom Blanket) : 15.24 cm
Region 2 : 45.72 cm
Region 3 : 121.92 cm
Region 4 : 91.44 cm
Region 5 : 30.48 cm
Region 6 : 15.24 cm
Region 7 : 15.24 cm
Region 8 : 15.24 cm
Region 9 (Top Blanket) : 15.24 cm

4. Material Data(at 300K)

A. Fuel

Fresh Fuel (3.80 wt %) (for Case 1 in Table 1)

Nuclide	Number Density [atom/(barn·cm)]
U-234	7.8161E-06
U-235	8.5393E-04
U-236	5.3079E-06
U-238	2.1365E-02
O	4.4760E-02

Fresh Blanket Fuel (0.71 wt %) (for Case 1 in Table 1)

Nuclide	Number Density [atom/(barn·cm)]
U-234	1.1732E-06
U-235	1.6128E-04
U-236	1.2360E-06
U-238	2.2235E-02
O	4.4797E-02

The number densities of spent fuels are given in Appendix A.

B. Cladding, Channel Box and End Plug

Zircalloy-4

Nuclide	Number Density [atom/(barn·cm)]
Cr	7.5891E-05
Fe	1.4838E-04
Zr	4.2982E-02

C. Gas Plenum

For some members who wish to smear the gas plenum region of a rod, the following data may be used.

Zircalloy-4 (for Fuel Rod)

Nuclide	Number Density [atom/(barn·cm)]
Cr	1.9741E-05
Fe	3.8597E-05
Zr	1.1181E-02

D. Moderator

Water

Nuclide	Number Density [atom/(barn·cm)]
H	6.6621E-02
O	3.3310E-02

5. Requested Information and Results

Please forward the results by E-mail to Dr. H. Okuno of JAERI.

Line No.	Contents
1	Date
2	Institute
3	Contact person
4	E-mail address or telefax number of the contact person
5	Computer code
6-27	Multiplication factors for the studied cases
28-36	Fractional fission densities from Regions 1 to 9 for Case 1
37-45	Fractional fission densities from Regions 1 to 9 for Case 5
46-54	Fractional fission densities from Regions 1 to 9 for Case 6
55-63	Fractional fission densities from Regions 1 to 9 for Case 7
64-72	Fractional fission densities from Regions 1 to 9 for Case 14
73-	Please describe your analysis environment here. It will be included in Phase IIIA report. The description should include: Institute and country, Participants, Neutron data library, Neutron data processing code or method, Neutron energy groups, Description of your code system, Geometry modeling, if any, Omitted or replaced nuclides, if any, Employed convergence limit or statistical errors for the eigen value calculations, Other information, if any.

Note : Definition of Fractional fission density for Region n

$[F.F.D.]_n$ is as follows.

$$[F.F.D.]_n = \frac{\iint dE d\bar{r}_n \Sigma_f(\bar{r}_n, E)\Phi_n(\bar{r}_n, E)}{\sum_{i=1}^{9} \iint dE d\bar{r}_i \Sigma_f(\bar{r}_i, E)\Phi_i(\bar{r}_i, E)}$$

6. Schedule

End March 1996	Results should be at the coordinator
Early June 1996	A first draft report will be prepared and distributed to participants

Table 1 A list of parameters and case numbers

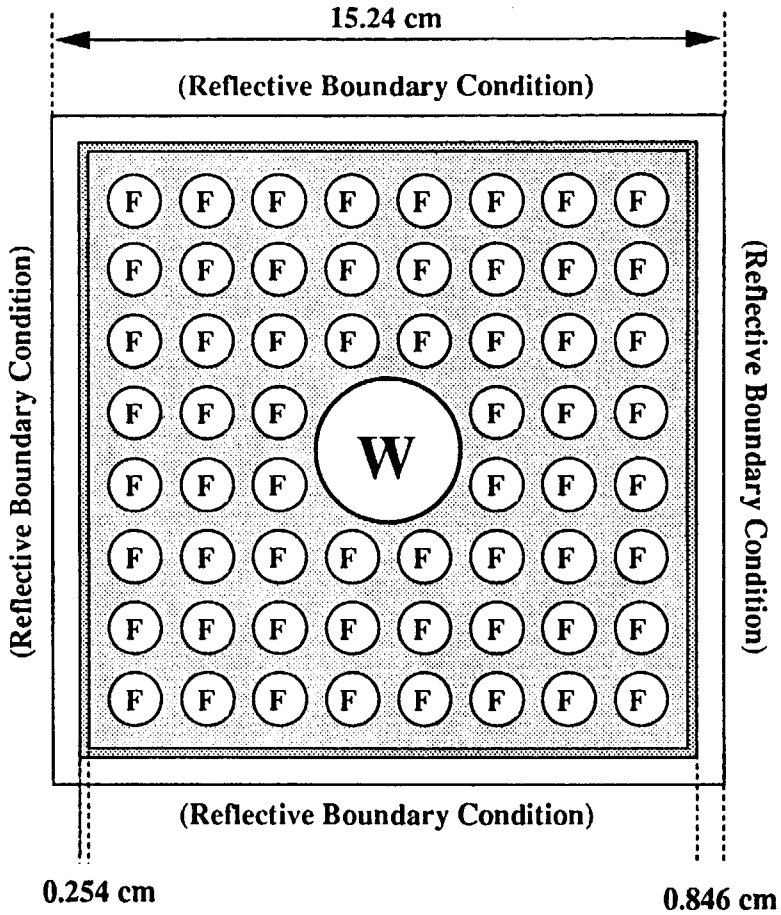
Cooling Time [y]	FPs	Burnup Profile	Void Profile	Case No.			
				Burnup [GWd/tU]			
				0	20	30	40
1	Yes	Yes	Yes	1	2	3	4
			40% Uniform		N/A	N/A	N/A
			70% Uniform		N/A	N/A	N/A
		No	Yes		N/A	N/A	N/A
			40% Uniform		N/A	N/A	N/A
			70% Uniform		N/A	N/A	N/A
	No	Yes	Yes		N/A	N/A	N/A
			40% Uniform		N/A	N/A	N/A
			70% Uniform		N/A	N/A	N/A
		No	Yes		N/A	N/A	N/A
			40% Uniform		N/A	N/A	N/A
			70% Uniform		N/A	N/A	N/A
5	Yes	Yes	Yes		5	6	7
			40% Uniform		N/A	N/A	N/A
			70% Uniform		N/A	N/A	N/A
		No	Yes		N/A	N/A	8
			40% Uniform		9	10	11
			70% Uniform		12	13	14
	No	Yes	Yes		N/A	N/A	15
			40% Uniform		N/A	N/A	N/A
			70% Uniform		N/A	N/A	N/A
		No	Yes		N/A	N/A	16
			40% Uniform		17	18	19
			70% Uniform		20	21	22

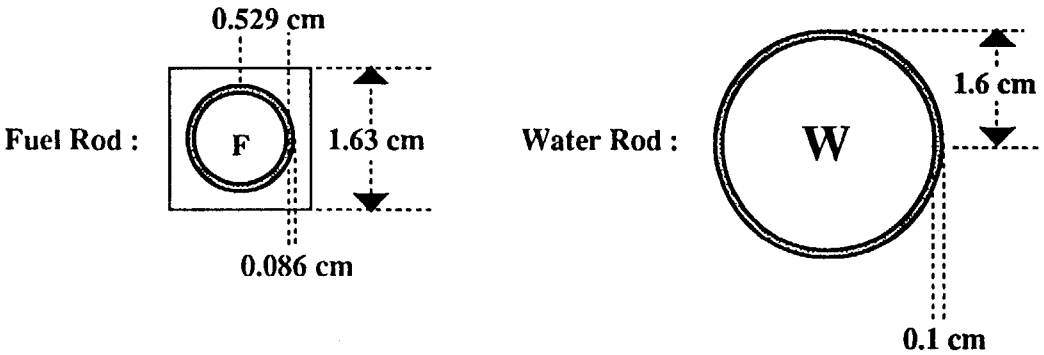
Note N/A: Not Adopted
40% and 70% uniform void cases are considered as "Void Profile = No".

Table 2 The data set numbers of the atomic number densities shown in Appendix A applicable for each case to criticality calculation

Case No.	Region No.								
	1	2	3	4	5	6	7	8	9
1	See "Material Data"								
2	1	2	3	4	5	6	7	8	9
3	19	20	21	22	23	24	25	26	27
4	37	38	39	40	41	42	43	44	45
5	10	11	12	13	14	15	16	17	18
6	28	29	30	31	32	33	34	35	36
7	46	47	48	49	50	51	52	53	54
8	55	56	57	58	59	60	61	62	63
9	64	65							64
10	66	67							66
11	68	69							68
12	70	71							70
13	72	73							72
14	74	75							74
15	46	47	48	49	50	51	52	53	54
16	55	56	57	58	59	60	61	62	63
17	64	65							64
18	66	67							66
19	68	69							68
20	70	71							70
21	72	73							72
22	74	75							74

Note: For the cases from 15 to 22, FP nuclides (Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Nd-143, Nd-145, Eu-153 and Gd-155) should be disregarded from the nuclide list shown in Appendix A.





Geometry Data

Assembly Pitch	= 15.24 cm
Cell Pitch	= 1.63 cm
OuterRadius of Fuel Rod	= 0.615 cm
Inner Radius of Fuel Rod	= 0.529 cm
Cladding Thickness of Fuel Rod	= 0.086 cm
Outer Radius of Water Rod	= 1.6 cm
Inner Radius of Water Rod	= 1.5 cm
Cladding Thickness of Water Rod	= 0.1 cm
Channel BoxThickness	= 0.254 cm
1/2-Thickness of Water Gap	= 0.846 cm

Fig. 1 Radial Dimensions

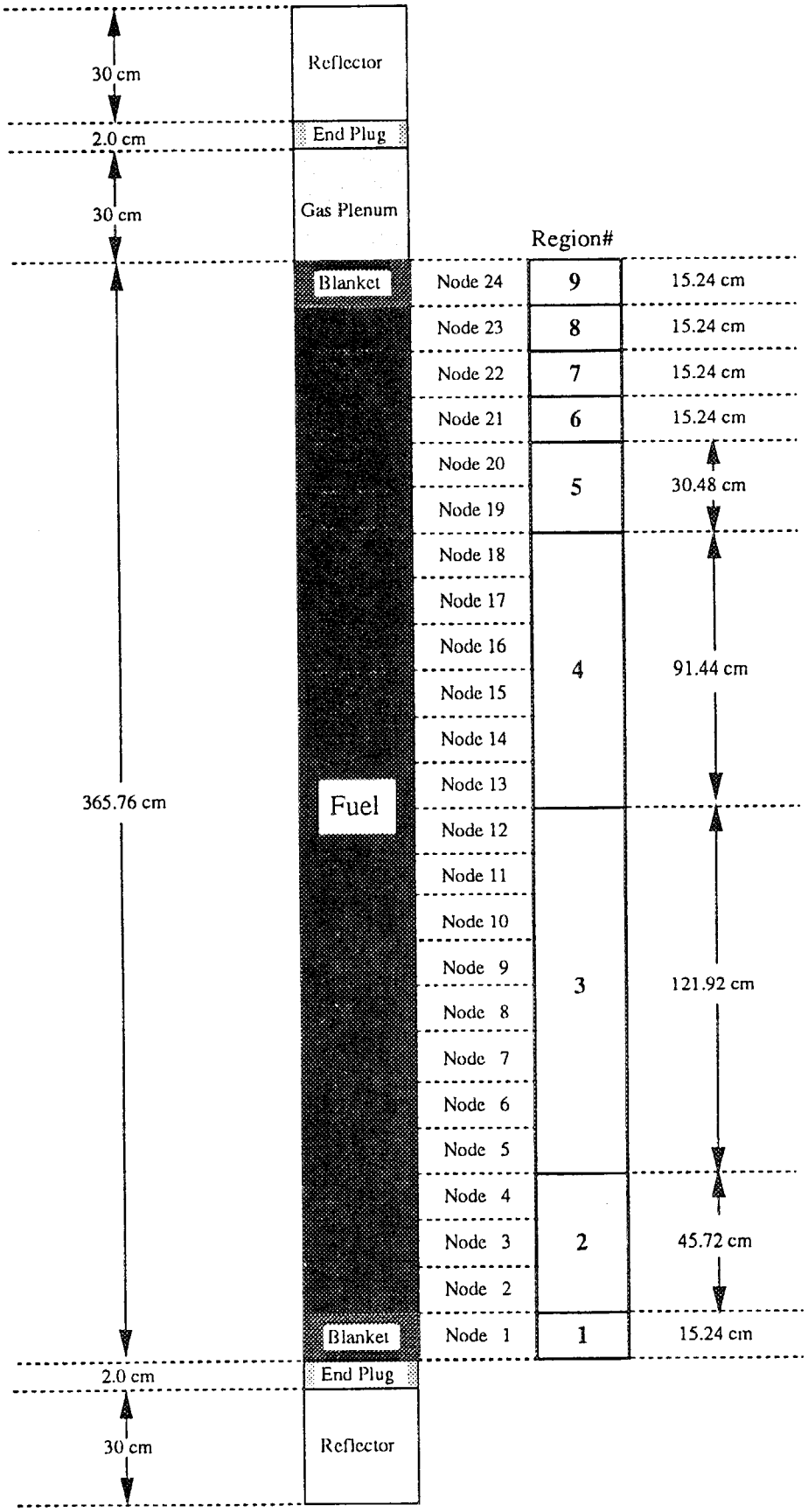


Fig. 2 Axial Dimensions

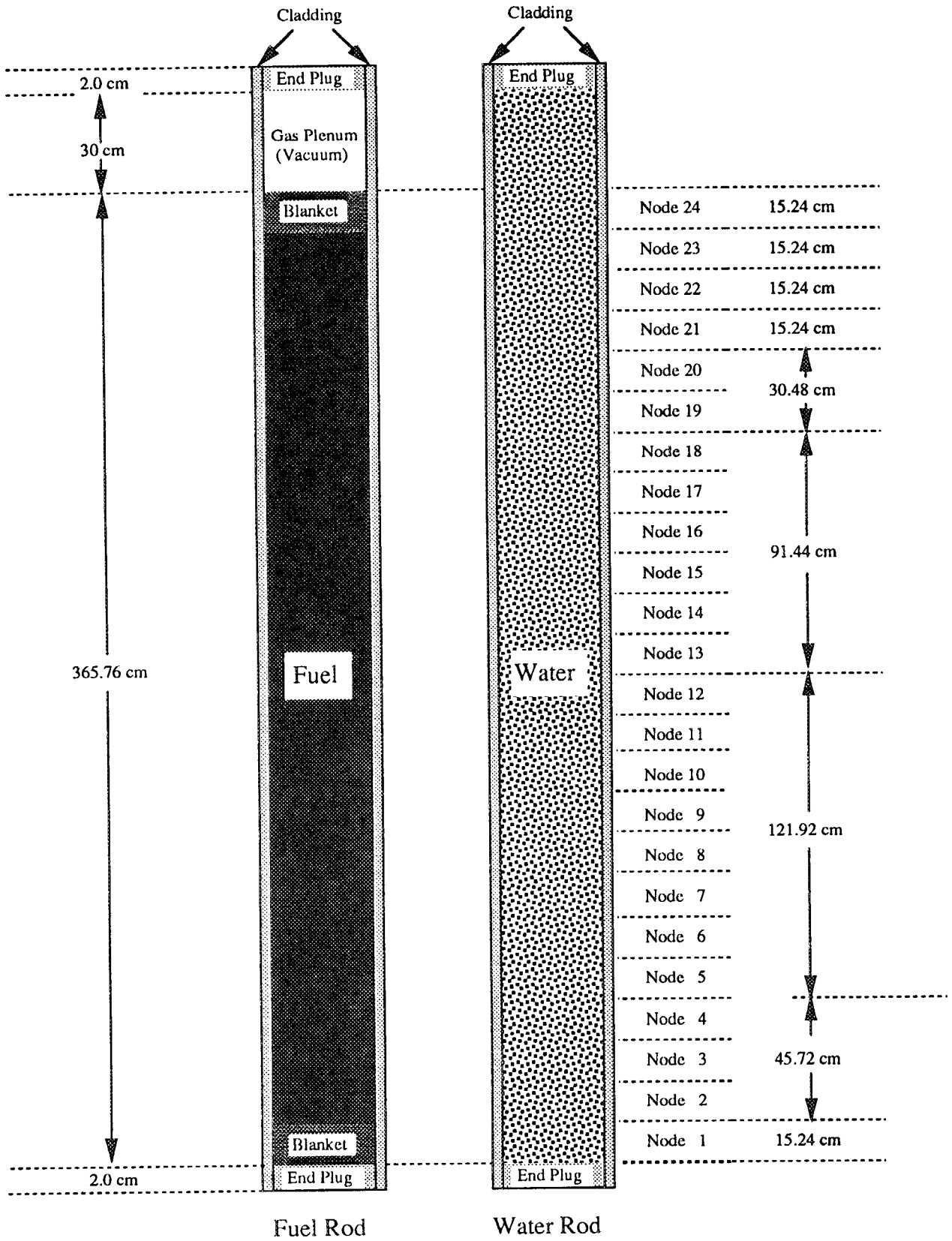


Fig. 3 Axial Structure in Fuel / Water Rod

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Appendix A

Atomic Number Densities of Spent Fuels for Analysis of BWR Burnup Credit Criticality Benchmarks

Prepared by Yoshihira Ando

1. Operating History in BWRs

To calculate the fuel composition of spent fuel, void and burnup profiles in BWRs are needed. So, these operating histories were calculated on the assumption of Haling distribution for 4 batch core in which average enrichment is 3.80 w/o in fuel region and a node fuel with natural uranium is set up as the blanket in each side of the fuel. Burnup profile is shown in Fig.A1, and void profile is shown in Fig.A2.

2. Selection of BWR Fuel Assembly

I selected a typical BWR fuel assembly shown in Fig.A3. This type of fuel assembly has been used widely in the USA and Japanese commercial BWR plants. The fuel enrichment distribution shown in Fig.A3 is a typical example in which fuel rod average enrichment is 3.80 w/o. This enrichment is set up based on the fuel design to achieve a batch average exposure of about 40 GWd/tM. A fuel assembly in blanket region is shown in Fig.A4. All fuel rods in this fuel assembly contain natural uranium and the assembly has a large center water rod same as the typical fuel assembly shown in Fig.A3.

For spent fuels of theses assembly, we calculated atomic number densities used in benchmark problems.

3. Method in Calculating Atomic Number Densities of Spent Fuel

I used SPINOZA (Spectrum Induced Nuclear OrganiZation Analysis) system in calculating atomic number densities of spent fuels.

SPINOZA system has been developed in Toshiba. In this system, a user can execute ORIGEN2 calculation with cross section corresponding to neutron spectrum in BWR lattice. The schematic flow diagram of SPINOZA system is shown in Fig.A5.

In calculating the atomic number densities of spent fuels, I assumed burnup and void profiles, which are shown in Table A1 and Table A2, respectively.

4. Results of Calculation

The atomic number densities of the BWR spent fuels calculated according to the above method are shown in Table A3.

Acknowledgments

I would like to express my sincere thanks to Messrs. M. Yamamoto, K. Sakurada and I. Mitsuhashi for their valuable comments and my special thanks to Mr. Y. Hirano for the estimation of void and burnup profiles by 3D BWR core simulation.

Table A1 Burnup Profile in each Case#

Case#	Burnup	BP	VP	Region#	Discharged Burnup	Burnup at EOC (GWd/tM) Cycle-1	Cycle-2	Cycle-3	Cycle-4	Specific Power (Watt/gram) Cycle-1	Cycle-2	Cycle-3	Cycle-4	Data Set#
2&5	20 GWd/tM	Yes	Yes	1	4.646 GWd/tM	2.375	4.646			6.090	5.823			1&10
	"	"	"	2	19.610 "	10.520	19.610			26.975	23.308			2&11
	"	"	"	3	22.767 "	12.213	22.767			31.316	27.062			3&12
	"	"	"	4	22.948 "	12.233	22.948			31.367	27.474			4&13
	"	"	"	5	21.331 "	11.272	21.331			28.903	25.793			5&14
	"	"	"	6	19.116 "	10.085	19.116			25.859	23.157			6&15
	"	"	"	7	16.813 "	8.801	16.813			22.567	20.544			7&16
	"	"	"	8	13.214 "	6.788	13.214			17.405	16.477			8&17
	"	"	"	9	4.893 "	2.405	4.893			6.167	6.379			9&18
3&6	30 GWd/tM	Yes	Yes	1	7.203 GWd/tM	2.558	5.002	7.203		6.559	6.267	5.644		19&28
	"	"	"	2	29.063 "	11.326	21.114	29.063		29.041	25.098	20.383		20&29
	"	"	"	3	33.953 "	13.149	24.512	33.953		33.716	29.136	24.208		21&30
	"	"	"	4	34.461 "	13.171	24.708	34.461		33.772	29.582	25.008		22&31
	"	"	"	5	32.195 "	12.137	22.966	32.195		31.121	27.767	23.664		23&32
	"	"	"	6	28.992 "	10.858	20.582	28.992		27.841	24.934	21.565		24&33
	"	"	"	7	25.627 "	9.476	18.102	25.627		24.298	22.118	19.295		25&34
	"	"	"	8	20.340 "	7.309	14.227	20.340		18.741	17.739	15.675		26&35
	"	"	"	9	7.870 "	2.589	5.267	7.870		6.639	6.867	6.674		27&36
4&7	40 GWd/tM	Yes	Yes	1	9.402 GWd/tM	2.922	5.713	8.227	9.402	7.493	7.156	6.446	3.013	37&46
	"	"	"	2	37.343 "	12.935	24.114	33.194	37.343	33.167	28.664	23.282	10.638	38&47
	"	"	"	3	44.799 "	15.018	27.996	38.674	44.799	38.508	33.277	27.380	15.705	39&48
	"	"	"	4	46.484 "	15.043	28.219	39.359	46.484	38.572	33.785	28.564	18.270	40&49
	"	"	"	5	43.760 "	13.862	26.230	36.771	43.760	35.544	31.713	27.028	17.921	41&50
	"	"	"	6	39.610 "	12.402	23.507	33.112	39.610	31.800	28.474	24.629	16.662	42&51
	"	"	"	7	35.172 "	10.823	20.675	29.269	35.172	27.752	25.262	22.036	15.136	43&52
	"	"	"	8	28.034 "	8.347	16.249	23.231	28.034	21.403	20.262	17.903	12.316	44&53
	"	"	"	9	10.935 "	2.957	6.017	8.988	10.935	7.582	7.846	7.618	4.992	45&54

Table A1 (Cont'd)

Case#	Burnup	BP	VP	Region#	Discharged Burnup	Burnup at EOC (GWd/tM) Cycle-1	Cycle-2	Cycle-3	Cycle-4	Specific Power (Watt/gram) Cycle-1	Cycle-2	Cycle-3	Cycle-4	Data Set#
8	40 GWd/tM	No	Yes	1	40.0 GWd/tM	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	55
	"	"	"	2	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	56
	"	"	"	3	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	57
	"	"	"	4	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	58
	"	"	"	5	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	59
	"	"	"	6	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	60
	"	"	"	7	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	61
	"	"	"	8	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	62
	"	"	"	9	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	63
9&17	20 GWd/tM	No	40VH	1&9	20.0 GWd/tM	10.653	20.000			27.315	23.967			64
	"	"	"	2 ~ 8	"	10.653	20.000			27.315	23.967			65
10&18	30 GWd/tM	"	"	1&9	30.0 GWd/tM	11.465	21.533	30.000		29.398	25.816	21.711		66
	"	"	"	2 ~ 8	"	11.465	21.533	30.000		29.398	25.816	21.711		67
11&19	40 GWd/tM	"	"	1&9	40.0 GWd/tM	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	68
	"	"	"	2 ~ 8	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	69
12&20	20 GWd/tM	"	70VH	1&9	20.0 GWd/tM	10.653	20.000			27.315	23.967			70
	"	"	"	2 ~ 8	"	10.653	20.000			27.315	23.967			71
13&21	30 GWd/tM	"	"	1&9	30.0 GWd/tM	11.465	21.533	30.000		29.398	25.816	21.711		72
	"	"	"	2 ~ 8	"	11.465	21.533	30.000		29.398	25.816	21.711		73
14&22	40 GWd/tM	"	"	1&9	40.0 GWd/tM	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	74
	"	"	"	2 ~ 8	"	13.621	24.594	34.264	40.000	34.926	28.136	24.795	14.708	75

Note: The folloing operation history is assumed in the burnup calculations.
Operation period per cycle is 390 days (13 months).
Outage period per cycle is 90 days (3monts).

Table A2 Axial Void Profile

Region#	*Void Fraction (%) Cycle-1	Cycle-2	Cycle-3	Cycle-4
1	0.000	0.000	0.000	0.000
2	3.230	2.695	2.351	1.986
3	41.051	37.909	34.049	28.243
4	68.228	65.566	62.020	55.886
5	75.710	73.337	70.260	64.827
6	77.661	75.369	72.407	67.171
7	78.687	76.445	73.515	68.377
8	79.489	77.263	74.366	69.260
9	79.936	77.661	74.634	69.511

Table A3 OECD/NEA Burnup Credit Benchmark Phase IIIA : BWR Spent Fuel Benchmarks

Data Set No. 1		Data Set No. 2		Data Set No. 3		Data Set No. 4		Data Set No. 5	
Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year	
Region# : 1		Region# : 2		Region# : 3		Region# : 4		Region# : 5	
Burnup = 4.646GWd/tM		Burnup = 19.610GWd/tM		Burnup = 22.767GWd/tM		Burnup = 22.948GWd/tM		Burnup = 21.331Gwd/tM	
u-234	1.0199E-06	u-234	6.1832E-06	u-234	5.8184E-06	u-234	5.7004E-06	u-234	5.8062E-06
u-235	8.5538E-05	u-235	4.3234E-04	u-235	3.9335E-04	u-235	4.0334E-04	u-235	4.3132E-04
u-236	1.3079E-05	u-236	7.6962E-05	u-236	8.5178E-05	u-236	8.6091E-05	u-236	8.2021E-05
u-238	2.2128E-02	u-238	2.1143E-02	u-238	2.1073E-02	u-238	2.1042E-02	u-238	2.1058E-02
pu-238	6.3818E-08	pu-238	6.4614E-07	pu-238	1.1306E-06	pu-238	1.3821E-06	pu-238	1.2292E-06
pu-239	4.9183E-05	pu-239	7.9385E-05	pu-239	9.2730E-05	pu-239	1.0417E-04	pu-239	1.0561E-04
pu-240	1.4026E-05	pu-240	2.5223E-05	pu-240	3.1838E-05	pu-240	3.4031E-05	pu-240	3.1975E-05
pu-241	3.0675E-06	pu-241	9.6492E-06	pu-241	1.3855E-05	pu-241	1.5745E-05	pu-241	1.4689E-05
pu-242	4.5546E-07	pu-242	1.9575E-06	pu-242	3.2558E-06	pu-242	3.4944E-06	pu-242	2.9300E-06
am-241	2.4901E-07	am-241	7.8005E-07	am-241	1.1123E-06	am-241	1.2647E-06	am-241	1.1868E-06
am-243	1.2931E-08	am-243	1.4543E-07	am-243	3.3594E-07	am-243	4.2653E-07	am-243	3.4864E-07
np-237	5.0422E-07	np-237	3.7757E-06	np-237	5.2564E-06	np-237	5.9017E-06	np-237	5.4950E-06
mo-95	6.2080E-06	mo-95	2.7067E-05	mo-95	3.0784E-05	mo-95	3.0698E-05	mo-95	2.8623E-05
tc-99	6.4386E-06	tc-99	2.6668E-05	tc-99	3.0641E-05	tc-99	3.0744E-05	tc-99	2.8643E-05
ru-101	5.8948E-06	ru-101	2.3871E-05	ru-101	2.7741E-05	ru-101	2.7962E-05	ru-101	2.6000E-05
rh-103	4.5721E-06	rh-103	1.5408E-05	rh-103	1.7867E-05	rh-103	1.8058E-05	rh-103	1.6878E-05
ag-109	6.0159E-07	ag-109	1.4559E-06	ag-109	1.9589E-06	ag-109	2.1027E-06	ag-109	1.9236E-06
cs-133	7.1873E-06	cs-133	2.9344E-05	cs-133	3.3456E-05	cs-133	3.3443E-05	cs-133	3.1225E-05
sm-147	8.9678E-07	sm-147	3.3537E-06	sm-147	3.5878E-06	sm-147	3.4873E-06	sm-147	3.2989E-06
sm-149	2.0604E-08	sm-149	8.2298E-08	sm-149	9.4319E-08	sm-149	1.0687E-07	sm-149	1.0490E-07
sm-150	1.2228E-06	sm-150	5.1237E-06	sm-150	6.0584E-06	sm-150	6.1436E-06	sm-150	5.7084E-06
sm-151	8.3034E-08	sm-151	3.2064E-07	sm-151	3.7207E-07	sm-151	4.1935E-07	sm-151	4.2587E-07
sm-152	8.9675E-07	sm-152	2.9332E-06	sm-152	3.3250E-06	sm-152	3.2795E-06	sm-152	3.0454E-06
nd-143	5.0075E-06	nd-143	2.1548E-05	nd-143	2.4146E-05	nd-143	2.4369E-05	nd-143	2.3080E-05
nd-145	3.7918E-06	nd-145	1.6180E-05	nd-145	1.8291E-05	nd-145	1.8207E-05	nd-145	1.7014E-05
eu-153	3.6438E-07	eu-153	1.6230E-06	eu-153	2.0982E-06	eu-153	2.1981E-06	eu-153	2.0042E-06
gd-155	5.0457E-09	gd-155	2.0994E-08	gd-155	2.9494E-08	gd-155	3.4196E-08	gd-155	3.2181E-08
o	4.4783E-02	o	4.4745E-02	o	4.4745E-02	o	4.4745E-02	o	4.4745E-02

Table A3 (Cont'd)

Data Set No. 6		Data Set No. 7		Data Set No. 8		Data Set No. 9	
Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year	
Region# : 6		Region# : 7		Region# : 8		Region# : 9	
Burnup = 19.116GWd/tM		Burnup = 16.813GWd/tM		Burnup = 13.214GWd/tM		Burnup = 4.893GWd/tM	
u-234	5.9871E-06	u-234	6.1818E-06	u-234	6.4977E-06	u-234	1.0029E-06
u-235	4.6641E-04	u-235	5.0451E-04	u-235	5.6749E-04	u-235	8.8656E-05
u-236	7.6468E-05	u-236	6.9897E-05	u-236	5.8905E-05	u-236	1.3024E-05
u-238	2.1090E-02	u-238	2.1124E-02	u-238	2.1176E-02	u-238	2.2107E-02
pu-238	9.7533E-07	pu-238	7.3234E-07	pu-238	4.2645E-07	pu-238	1.0028E-07
pu-239	1.0288E-04	pu-239	9.8901E-05	pu-239	8.9804E-05	pu-239	5.7922E-05
pu-240	2.8289E-05	pu-240	2.4318E-05	pu-240	1.7912E-05	pu-240	1.5247E-05
pu-241	1.2736E-05	pu-241	1.0593E-05	pu-241	7.2878E-06	pu-241	4.2876E-06
pu-242	2.2195E-06	pu-242	1.5873E-06	pu-242	8.2810E-07	pu-242	6.0995E-07
am-241	1.0373E-06	am-241	8.6714E-07	am-241	5.9662E-07	am-241	3.4838E-07
am-243	2.3912E-07	am-243	1.5109E-07	am-243	6.1773E-08	am-243	2.7267E-08
np-237	4.8027E-06	np-237	4.0643E-06	np-237	2.9722E-06	np-237	6.9275E-07
mo-95	2.5849E-05	mo-95	2.2928E-05	mo-95	1.8269E-05	mo-95	6.4231E-06
tc-99	2.5778E-05	tc-99	2.2786E-05	tc-99	1.8043E-05	tc-99	6.7505E-06
ru-101	2.3312E-05	ru-101	2.0514E-05	ru-101	1.6127E-05	ru-101	6.2439E-06
rh-103	1.5225E-05	rh-103	1.3486E-05	rh-103	1.0676E-05	rh-103	4.9126E-06
ag-109	1.6514E-06	ag-109	1.3782E-06	ag-109	9.7843E-07	ag-109	6.9462E-07
cs-133	2.8184E-05	cs-133	2.5013E-05	cs-133	1.9835E-05	cs-133	7.5107E-06
sm-147	3.0688E-06	sm-147	2.8052E-06	sm-147	2.3389E-06	sm-147	9.0876E-07
sm-149	1.0628E-07	sm-149	1.0496E-07	sm-149	1.0451E-07	sm-149	2.5061E-08
sm-150	5.0703E-06	sm-150	4.4383E-06	sm-150	3.4246E-06	sm-150	1.3126E-06
sm-151	4.1946E-07	sm-151	4.0436E-07	sm-151	3.7736E-07	sm-151	1.0283E-07
sm-152	2.7425E-06	sm-152	2.4234E-06	sm-152	1.9001E-06	sm-152	9.4018E-07
nd-143	2.1190E-05	nd-143	1.9081E-05	nd-143	1.5551E-05	nd-143	5.2539E-06
nd-145	1.5422E-05	nd-145	1.3733E-05	nd-145	1.1009E-05	nd-145	3.9308E-06
eu-153	1.7239E-06	eu-153	1.4439E-06	eu-153	1.0335E-06	eu-153	4.2074E-07
gd-155	2.8465E-08	gd-155	2.4307E-08	gd-155	1.9574E-08	gd-155	6.2083E-09
o	4.4745E-02	o	4.4745E-02	o	4.4745E-02	o	4.4783E-02

Table A3 (Cont'd)

Data Set No. 10 Cooling : 5 years Region# : 1 Burnup = 4.646GWd/tM		Data Set No. 11 Cooling : 5 years Region# : 2 Burnup = 19.610GWd/tM		Data Set No. 12 Cooling : 5 years Region# : 3 Burnup = 22.767GWd/tM		Data Set No. 13 Cooling : 5 years Region# : 4 Burnup = 22.948GWd/tM		Data Set No. 14 Cooling : 5 years Region# : 5 Burnup = 21.331GWd/tM	
u-234	1.0220E-06	u-234	6.2036E-06	u-234	5.8540E-06	u-234	5.7440E-06	u-234	5.8449E-06
u-235	8.5544E-05	u-235	4.3235E-04	u-235	3.9336E-04	u-235	4.0335E-04	u-235	4.3133E-04
u-236	1.3085E-05	u-236	7.6973E-05	u-236	8.5191E-05	u-236	8.6105E-05	u-236	8.2035E-05
u-238	2.2128E-02	u-238	2.1143E-02	u-238	2.1073E-02	u-238	2.1042E-02	u-238	2.1058E-02
pu-238	6.3963E-08	pu-238	6.3714E-07	pu-238	1.1148E-06	pu-238	1.3617E-06	pu-238	1.2107E-06
pu-239	4.9177E-05	pu-239	7.9376E-05	pu-239	9.2719E-05	pu-239	1.0415E-04	pu-239	1.0559E-04
pu-240	1.4020E-05	pu-240	2.5214E-05	pu-240	3.1832E-05	pu-240	3.4027E-05	pu-240	3.1970E-05
pu-241	2.5303E-06	pu-241	7.9592E-06	pu-241	1.1428E-05	pu-241	1.2987E-05	pu-241	1.2117E-05
pu-242	4.5546E-07	pu-242	1.9575E-06	pu-242	3.2558E-06	pu-242	3.4944E-06	pu-242	2.9300E-06
am-241	7.8289E-07	am-241	2.4595E-06	am-241	3.5238E-06	am-241	4.0051E-06	am-241	3.7434E-06
am-243	1.2926E-08	am-243	1.4538E-07	am-243	3.3581E-07	am-243	4.2637E-07	am-243	3.4851E-07
np-237	5.0760E-07	np-237	3.7863E-06	np-237	5.2716E-06	np-237	5.9190E-06	np-237	5.5111E-06
mo-95	6.2373E-06	mo-95	2.7190E-05	mo-95	3.0924E-05	mo-95	3.0840E-05	mo-95	2.8756E-05
tc-99	6.4386E-06	tc-99	2.6667E-05	tc-99	3.0641E-05	tc-99	3.0744E-05	tc-99	2.8642E-05
ru-101	5.8948E-06	ru-101	2.3871E-05	ru-101	2.7741E-05	ru-101	2.7962E-05	ru-101	2.6000E-05
rh-103	4.5727E-06	rh-103	1.5410E-05	rh-103	1.7870E-05	rh-103	1.8060E-05	rh-103	1.6881E-05
ag-109	6.0159E-07	ag-109	1.4559E-06	ag-109	1.9589E-06	ag-109	2.1027E-06	ag-109	1.9236E-06
cs-133	7.1873E-06	cs-133	2.9344E-05	cs-133	3.3456E-05	cs-133	3.3443E-05	cs-133	3.1225E-05
sm-147	1.6763E-06	sm-147	6.2545E-06	sm-147	6.7057E-06	sm-147	6.5417E-06	sm-147	6.2003E-06
sm-149	2.0604E-08	sm-149	8.2298E-08	sm-149	9.4319E-08	sm-149	1.0687E-07	sm-149	1.0490E-07
sm-150	1.2228E-06	sm-150	5.1237E-06	sm-150	6.0584E-06	sm-150	6.1436E-06	sm-150	5.7084E-06
sm-151	8.0514E-08	sm-151	3.1092E-07	sm-151	3.6078E-07	sm-151	4.0663E-07	sm-151	4.1295E-07
sm-152	8.9677E-07	sm-152	2.9333E-06	sm-152	3.3251E-06	sm-152	3.2796E-06	sm-152	3.0455E-06
nd-143	5.0075E-06	nd-143	2.1548E-05	nd-143	2.4146E-05	nd-143	2.4369E-05	nd-143	2.3080E-05
nd-145	3.7918E-06	nd-145	1.6180E-05	nd-145	1.8291E-05	nd-145	1.8207E-05	nd-145	1.7014E-05
eu-153	3.6439E-07	eu-153	1.6231E-06	eu-153	2.0983E-06	eu-153	2.1982E-06	eu-153	2.0043E-06
gd-155	1.8525E-08	gd-155	7.4457E-08	gd-155	1.0544E-07	gd-155	1.2089E-07	gd-155	1.1201E-07
o	4.4783E-02	o	4.4745E-02	o	4.4745E-02	o	4.4745E-02	o	4.4745E-02

Table A3 (Cont'd)

Data Set No. 15		Data Set No. 16		Data Set No. 17		Data Set No. 18	
Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years	
Region# : 6		Region# : 7		Region# : 8		Region# : 9	
Burnup = 19.116GWd/tM		Burnup = 16.813GWd/tM		Burnup = 13.214GWd/tM		Burnup = 4.893GWd/tM	
u-234	6.0178E-06	u-234	6.2048E-06	u-234	6.5111E-06	u-234	1.0061E-06
u-235	4.6643E-04	u-235	5.0452E-04	u-235	5.6750E-04	u-235	8.8662E-05
u-236	7.6480E-05	u-236	6.9908E-05	u-236	5.8912E-05	u-236	1.3031E-05
u-238	2.1090E-02	u-238	2.1124E-02	u-238	2.1176E-02	u-238	2.2107E-02
pu-238	9.6032E-07	pu-238	7.2071E-07	pu-238	4.1910E-07	pu-238	1.0041E-07
pu-239	1.0287E-04	pu-239	9.8890E-05	pu-239	8.9793E-05	pu-239	5.7915E-05
pu-240	2.8282E-05	pu-240	2.4310E-05	pu-240	1.7906E-05	pu-240	1.5241E-05
pu-241	1.0506E-05	pu-241	8.7375E-06	pu-241	6.0113E-06	pu-241	3.5367E-06
pu-242	2.2195E-06	pu-242	1.5873E-06	pu-242	8.2810E-07	pu-242	6.0995E-07
am-241	3.2540E-06	am-241	2.7107E-06	am-241	1.8650E-06	am-241	1.0946E-06
am-243	2.3903E-07	am-243	1.5103E-07	am-243	6.1750E-08	am-243	2.7257E-08
np-237	4.8167E-06	np-237	4.0761E-06	np-237	2.9803E-06	np-237	6.9747E-07
mo-95	2.5970E-05	mo-95	2.3037E-05	mo-95	1.8357E-05	mo-95	6.4547E-06
tc-99	2.5778E-05	tc-99	2.2786E-05	tc-99	1.8042E-05	tc-99	6.7504E-06
ru-101	2.3312E-05	ru-101	2.0514E-05	ru-101	1.6127E-05	ru-101	6.2439E-06
rh-103	1.5227E-05	rh-103	1.3488E-05	rh-103	1.0677E-05	rh-103	4.9134E-06
ag-109	1.6514E-06	ag-109	1.3782E-06	ag-109	9.7843E-07	ag-109	6.9462E-07
cs-133	2.8184E-05	cs-133	2.5013E-05	cs-133	1.9835E-05	cs-133	7.5107E-06
sm-147	5.7657E-06	sm-147	5.2751E-06	sm-147	4.4088E-06	sm-147	1.7129E-06
sm-149	1.0628E-07	sm-149	1.0496E-07	sm-149	1.0451E-07	sm-149	2.5061E-08
sm-150	5.0703E-06	sm-150	4.4383E-06	sm-150	3.4246E-06	sm-150	1.3126E-06
sm-151	4.0673E-07	sm-151	3.9210E-07	sm-151	3.6591E-07	sm-151	9.9711E-08
sm-152	2.7426E-06	sm-152	2.4235E-06	sm-152	1.9002E-06	sm-152	9.4020E-07
nd-143	2.1190E-05	nd-143	1.9081E-05	nd-143	1.5551E-05	nd-143	5.2539E-06
nd-145	1.5422E-05	nd-145	1.3733E-05	nd-145	1.1009E-05	nd-145	3.9308E-06
eu-153	1.7240E-06	eu-153	1.4440E-06	eu-153	1.0335E-06	eu-153	4.2076E-07
gd-155	9.6328E-08	gd-155	8.0580E-08	gd-155	6.0390E-08	gd-155	2.2655E-08
o	4.4745E-02	o	4.4745E-02	o	4.4745E-02	o	4.4783E-02

Table A3 (Cont'd)

Data Set No. 19		Data Set No. 20		Data Set No. 21		Data Set No. 22		Data Set No. 23	
Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year	
Region# : 1		Region# : 2		Region# : 3		Region# : 4		Region# : 5	
Burnup = 7.203GWd/tM		Burnup = 29.063GWd/tM		Burnup = 33.953GWd/tM		Burnup = 34.461GWd/tM		Burnup = 32.195GWd/tM	
u-234	9.4835E-07	u-234	5.4282E-06	u-234	4.9344E-06	u-234	4.7992E-06	u-234	4.9436E-06
u-235	6.1963E-05	u-235	2.8462E-04	u-235	2.4229E-04	u-235	2.5589E-04	u-235	2.8669E-04
u-236	1.6630E-05	u-236	9.9294E-05	u-236	1.0738E-04	u-236	1.0785E-04	u-236	1.0441E-04
u-238	2.2067E-02	u-238	2.1018E-02	u-238	2.0907E-02	u-238	2.0859E-02	u-238	2.0886E-02
pu-238	1.9531E-07	pu-238	1.7913E-06	pu-238	3.0264E-06	pu-238	3.6381E-06	pu-238	3.2790E-06
pu-239	5.6589E-05	pu-239	8.3830E-05	pu-239	9.5456E-05	pu-239	1.0879E-04	pu-239	1.1248E-04
pu-240	2.3126E-05	pu-240	3.8016E-05	pu-240	4.6146E-05	pu-240	4.9630E-05	pu-240	4.7690E-05
pu-241	6.0263E-06	pu-241	1.6406E-05	pu-241	2.1849E-05	pu-241	2.4833E-05	pu-241	2.4066E-05
pu-242	1.4972E-06	pu-242	5.7668E-06	pu-242	8.9401E-06	pu-242	9.2561E-06	pu-242	7.9120E-06
am-241	6.0738E-07	am-241	1.5699E-06	am-241	2.0320E-06	am-241	2.3064E-06	am-241	2.2569E-06
am-243	6.9490E-08	am-243	6.6740E-07	am-243	1.4037E-06	am-243	1.6833E-06	am-243	1.4041E-06
np-237	8.4998E-07	np-237	6.5448E-06	np-237	8.8905E-06	np-237	9.8565E-06	np-237	9.2705E-06
mo-95	9.2947E-06	mo-95	3.8787E-05	mo-95	4.3993E-05	mo-95	4.4057E-05	mo-95	4.1380E-05
tc-99	9.8662E-06	tc-99	3.8700E-05	tc-99	4.4504E-05	tc-99	4.4887E-05	tc-99	4.2093E-05
ru-101	9.2047E-06	ru-101	3.5337E-05	ru-101	4.1213E-05	ru-101	4.1756E-05	ru-101	3.9033E-05
rh-103	7.1732E-06	rh-103	2.1771E-05	rh-103	2.4958E-05	rh-103	2.5304E-05	rh-103	2.3916E-05
ag-109	1.1114E-06	ag-109	2.6245E-06	ag-109	3.5000E-06	ag-109	3.7339E-06	ag-109	3.4337E-06
cs-133	1.0953E-05	cs-133	4.1936E-05	cs-133	4.7596E-05	cs-133	4.7775E-05	cs-133	4.4939E-05
sm-147	1.5509E-06	sm-147	5.2571E-06	sm-147	5.4377E-06	sm-147	5.2502E-06	sm-147	5.0246E-06
sm-149	1.9827E-08	sm-149	7.0036E-08	sm-149	8.2552E-08	sm-149	9.6179E-08	sm-149	1.0014E-07
sm-150	1.8854E-06	sm-150	7.4417E-06	sm-150	8.7544E-06	sm-150	9.0072E-06	sm-150	8.4080E-06
sm-151	9.6552E-08	sm-151	3.2960E-07	sm-151	3.9208E-07	sm-151	4.5670E-07	sm-151	4.6719E-07
sm-152	1.4252E-06	sm-152	4.1867E-06	sm-152	4.6415E-06	sm-152	4.5703E-06	sm-152	4.2946E-06
nd-143	6.9918E-06	nd-143	2.7884E-05	nd-143	3.0892E-05	nd-143	3.1769E-05	nd-143	3.0633E-05
nd-145	5.6491E-06	nd-145	2.2831E-05	nd-145	2.5651E-05	nd-145	2.5630E-05	nd-145	2.4156E-05
eu-153	6.6378E-07	eu-153	2.8192E-06	eu-153	3.5945E-06	eu-153	3.7524E-06	eu-153	3.4599E-06
gd-155	8.0102E-09	gd-155	3.7672E-08	gd-155	5.5695E-08	gd-155	6.5283E-08	gd-155	6.1132E-08
o	4.4783E-02	o	4.4745E-02	o	4.4744E-02	o	4.4744E-02	o	4.4744E-02

Table A3 (Cont'd)

Data Set No. 24		Data Set No. 25		Data Set No. 26		Data Set No. 27	
Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year	
Region# : 6		Region# : 7		Region# : 8		Region# : 9	
Burnup = 28.992GWd/tM		Burnup = 25.627GWd/tM		Burnup = 20.340GWd/tM		Burnup = 7.870GWd/tM	
u-234	5.1871E-06	u-234	5.4486E-06	u-234	5.8893E-06	u-234	9.2071E-07
u-235	3.2638E-04	u-235	3.7206E-04	u-235	4.4758E-04	u-235	6.5022E-05
u-236	9.8914E-05	u-236	9.1877E-05	u-236	7.9482E-05	u-236	1.6661E-05
u-238	2.0936E-02	u-238	2.0987E-02	u-238	2.1072E-02	u-238	2.2031E-02
pu-238	2.6437E-06	pu-238	2.0405E-06	pu-238	1.1965E-06	pu-238	3.1143E-07
pu-239	1.1205E-04	pu-239	1.1124E-04	pu-239	1.0439E-04	pu-239	6.8824E-05
pu-240	4.3366E-05	pu-240	3.8271E-05	pu-240	3.0404E-05	pu-240	2.5567E-05
pu-241	2.1764E-05	pu-241	1.9188E-05	pu-241	1.3604E-05	pu-241	8.2737E-06
pu-242	6.2085E-06	pu-242	4.6023E-06	pu-242	2.5839E-06	pu-242	1.9503E-06
am-241	2.0779E-06	am-241	1.8542E-06	am-241	1.3655E-06	am-241	8.2258E-07
am-243	1.0044E-06	am-243	6.7503E-07	am-243	2.9473E-07	am-243	1.4408E-07
np-237	8.2058E-06	np-237	7.1016E-06	np-237	5.1808E-06	np-237	1.1942E-06
mo-95	3.7693E-05	mo-95	3.3681E-05	mo-95	2.7352E-05	mo-95	9.9409E-06
tc-99	3.8186E-05	tc-99	3.3956E-05	tc-99	2.7343E-05	tc-99	1.0709E-05
ru-101	3.5199E-05	ru-101	3.1133E-05	ru-101	2.4786E-05	ru-101	1.0101E-05
rh-103	2.1905E-05	rh-103	1.9409E-05	rh-103	1.5998E-05	rh-103	7.9127E-06
ag-109	2.9731E-06	ag-109	2.4851E-06	ag-109	1.8018E-06	ag-109	1.3082E-06
cs-133	4.1013E-05	cs-133	3.6596E-05	cs-133	2.9714E-05	cs-133	1.1821E-05
sm-147	4.7844E-06	sm-147	4.4129E-06	sm-147	3.8859E-06	sm-147	1.5890E-06
sm-149	9.9486E-08	sm-149	9.5801E-08	sm-149	9.9486E-08	sm-149	2.6970E-08
sm-150	7.5661E-06	sm-150	6.6742E-06	sm-150	5.2499E-06	sm-150	2.0954E-06
sm-151	4.5761E-07	sm-151	4.4127E-07	sm-151	4.2192E-07	sm-151	1.2467E-07
sm-152	3.9451E-06	sm-152	3.5108E-06	sm-152	2.8985E-06	sm-152	1.5229E-06
nd-143	2.8675E-05	nd-143	2.6272E-05	nd-143	2.2188E-05	nd-143	7.6312E-06
nd-145	2.2127E-05	nd-145	1.9862E-05	nd-145	1.6296E-05	nd-145	6.0492E-06
eu-153	3.0239E-06	eu-153	2.5971E-06	eu-153	1.8716E-06	eu-153	8.0333E-07
gd-155	5.2041E-08	gd-155	4.6128E-08	gd-155	3.1544E-08	gd-155	1.0590E-08
o	4.4744E-02	o	4.4744E-02	o	4.4745E-02	o	4.4782E-02

Table A3 (Cont'd)

Data Set No. 28		Data Set No. 29		Data Set No. 30		Data Set No. 31		Data Set No. 32	
Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years	
Region# : 1		Region# : 2		Region# : 3		Region# : 4		Region# : 5	
Burnup = 7.203GWd/tM		Burnup = 29.063GWd/tM		Burnup = 33.953GWd/tM		Burnup = 34.461GWd/tM		Burnup = 32.195GWd/tM	
u-234	9.5465E-07	u-234	5.4848E-06	u-234	5.0300E-06	u-234	4.9140E-06	u-234	5.0471E-06
u-235	6.1969E-05	u-235	2.8463E-04	u-235	2.4230E-04	u-235	2.5590E-04	u-235	2.8671E-04
u-236	1.6640E-05	u-236	9.9310E-05	u-236	1.0740E-04	u-236	1.0787E-04	u-236	1.0443E-04
u-238	2.2067E-02	u-238	2.1018E-02	u-238	2.0907E-02	u-238	2.0859E-02	u-238	2.0886E-02
pu-238	1.9746E-07	pu-238	1.7710E-06	pu-238	2.9884E-06	pu-238	3.5886E-06	pu-238	3.2343E-06
pu-239	5.6583E-05	pu-239	8.3821E-05	pu-239	9.5446E-05	pu-239	1.0878E-04	pu-239	1.1247E-04
pu-240	2.3117E-05	pu-240	3.8017E-05	pu-240	4.6176E-05	pu-240	4.9677E-05	pu-240	4.7724E-05
pu-241	4.9709E-06	pu-241	1.3532E-05	pu-241	1.8022E-05	pu-241	2.0484E-05	pu-241	1.9851E-05
pu-242	1.4972E-06	pu-242	5.7667E-06	pu-242	8.9401E-06	pu-242	9.2560E-06	pu-242	7.9120E-06
am-241	1.6555E-06	am-241	4.4237E-06	am-241	5.8330E-06	am-241	6.6266E-06	am-241	6.4436E-06
am-243	6.9464E-08	am-243	6.6715E-07	am-243	1.4032E-06	am-243	1.6826E-06	am-243	1.4036E-06
np-237	8.5738E-07	np-237	6.5644E-06	np-237	8.9162E-06	np-237	9.8857E-06	np-237	9.2990E-06
mo-95	9.3218E-06	mo-95	3.8890E-05	mo-95	4.4112E-05	mo-95	4.4179E-05	mo-95	4.1496E-05
tc-99	9.8661E-06	tc-99	3.8699E-05	tc-99	4.4503E-05	tc-99	4.4886E-05	tc-99	4.2092E-05
ru-101	9.2047E-06	ru-101	3.5337E-05	ru-101	4.1213E-05	ru-101	4.1756E-05	ru-101	3.9033E-05
rh-103	7.1738E-06	rh-103	2.1773E-05	rh-103	2.4960E-05	rh-103	2.5306E-05	rh-103	2.3919E-05
ag-109	1.1114E-06	ag-109	2.6245E-06	ag-109	3.5000E-06	ag-109	3.7339E-06	ag-109	3.4337E-06
cs-133	1.0953E-05	cs-133	4.1936E-05	cs-133	4.7596E-05	cs-133	4.7775E-05	cs-133	4.4939E-05
sm-147	2.5055E-06	sm-147	8.4046E-06	sm-147	8.7516E-06	sm-147	8.5139E-06	sm-147	8.1786E-06
sm-149	1.9827E-08	sm-149	7.0036E-08	sm-149	8.2552E-08	sm-149	9.6179E-08	sm-149	1.0014E-07
sm-150	1.8854E-06	sm-150	7.4417E-06	sm-150	8.7544E-06	sm-150	9.0072E-06	sm-150	8.4080E-06
sm-151	9.3622E-08	sm-151	3.1961E-07	sm-151	3.8019E-07	sm-151	4.4285E-07	sm-151	4.5302E-07
sm-152	1.4252E-06	sm-152	4.1868E-06	sm-152	4.6416E-06	sm-152	4.5703E-06	sm-152	4.2947E-06
nd-143	6.9918E-06	nd-143	2.7884E-05	nd-143	3.0892E-05	nd-143	3.1769E-05	nd-143	3.0633E-05
nd-145	5.6491E-06	nd-145	2.2831E-05	nd-145	2.5651E-05	nd-145	2.5630E-05	nd-145	2.4156E-05
eu-153	6.6380E-07	eu-153	2.8193E-06	eu-153	3.5947E-06	eu-153	3.7526E-06	eu-153	3.4600E-06
gd-155	2.9300E-08	gd-155	1.3543E-07	gd-155	2.0132E-07	gd-155	2.3360E-07	gd-155	2.1603E-07
o	4.4783E-02	o	4.4745E-02	o	4.4744E-02	o	4.4744E-02	o	4.4744E-02

Table A3 (Cont'd)

Data Set No. 33		Data Set No. 34		Data Set No. 35		Data Set No. 36	
Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years	
Region# : 6		Region# : 7		Region# : 8		Region# : 9	
Burnup = 28.992GWd/tM		Burnup = 25.627GWd/tM		Burnup = 20.340GWd/tM		Burnup = 7.870GWd/tM	
u-234	5.2705E-06	u-234	5.5130E-06	u-234	5.9270E-06	u-234	9.3073E-07
u-235	3.2639E-04	u-235	3.7207E-04	u-235	4.4759E-04	u-235	6.5029E-05
u-236	9.8932E-05	u-236	9.1893E-05	u-236	7.9494E-05	u-236	1.6672E-05
u-238	2.0936E-02	u-238	2.0987E-02	u-238	2.1072E-02	u-238	2.2031E-02
pu-238	2.6083E-06	pu-238	2.0133E-06	pu-238	1.1805E-06	pu-238	3.1401E-07
pu-239	1.1204E-04	pu-239	1.1123E-04	pu-239	1.0438E-04	pu-239	6.8816E-05
pu-240	4.3382E-05	pu-240	3.8275E-05	pu-240	3.0398E-05	pu-240	2.5558E-05
pu-241	1.7952E-05	pu-241	1.5827E-05	pu-241	1.1221E-05	pu-241	6.8246E-06
pu-242	6.2085E-06	pu-242	4.6023E-06	pu-242	2.5839E-06	pu-242	1.9503E-06
am-241	5.8638E-06	am-241	5.1919E-06	am-241	3.7314E-06	am-241	2.2616E-06
am-243	1.0040E-06	am-243	6.7478E-07	am-243	2.9462E-07	am-243	1.4402E-07
np-237	8.2317E-06	np-237	7.1246E-06	np-237	5.1974E-06	np-237	1.2042E-06
mo-95	3.7800E-05	mo-95	3.3778E-05	mo-95	2.7433E-05	mo-95	9.9725E-06
tc-99	3.8186E-05	tc-99	3.3956E-05	tc-99	2.7342E-05	tc-99	1.0709E-05
ru-101	3.5199E-05	ru-101	3.1133E-05	ru-101	2.4786E-05	ru-101	1.0101E-05
rh-103	2.1907E-05	rh-103	1.9411E-05	rh-103	1.6000E-05	rh-103	7.9135E-06
ag-109	2.9731E-06	ag-109	2.4851E-06	ag-109	1.8018E-06	ag-109	1.3082E-06
cs-133	4.1013E-05	cs-133	3.6596E-05	cs-133	2.9714E-05	cs-133	1.1821E-05
sm-147	7.7782E-06	sm-147	7.1614E-06	sm-147	6.3296E-06	sm-147	2.6073E-06
sm-149	9.9486E-08	sm-149	9.5801E-08	sm-149	9.9486E-08	sm-149	2.6970E-08
sm-150	7.5661E-06	sm-150	6.6742E-06	sm-150	5.2499E-06	sm-150	2.0954E-06
sm-151	4.4373E-07	sm-151	4.2788E-07	sm-151	4.0912E-07	sm-151	1.2089E-07
sm-152	3.9452E-06	sm-152	3.5110E-06	sm-152	2.8986E-06	sm-152	1.5229E-06
nd-143	2.8675E-05	nd-143	2.6272E-05	nd-143	2.2188E-05	nd-143	7.6312E-06
nd-145	2.2127E-05	nd-145	1.9862E-05	nd-145	1.6296E-05	nd-145	6.0492E-06
eu-153	3.0240E-06	eu-153	2.5973E-06	eu-153	1.8716E-06	eu-153	8.0337E-07
gd-155	1.8158E-07	gd-155	1.5426E-07	gd-155	1.0290E-07	gd-155	3.8543E-08
o	4.4744E-02	o	4.4744E-02	o	4.4745E-02	o	4.4782E-02

Table A3 (Cont'd)

Data Set No. 37		Data Set No. 38		Data Set No. 39		Data Set No. 40		Data Set No. 41	
Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year	
Region# : 1		Region# : 2		Region# : 3		Region# : 4		Region# : 5	
Burnup = 9.402GWd/tM		Burnup = 37.343GWd/tM		Burnup = 44.799GWd/tM		Burnup = 46.484GWd/tM		Burnup = 43.760GWd/tM	
u-234	8.9316E-07	u-234	4.7794E-06	u-234	4.1324E-06	u-234	3.9567E-06	u-234	4.1262E-06
u-235	4.7044E-05	u-235	1.8309E-04	u-235	1.3806E-04	u-235	1.4794E-04	u-235	1.7496E-04
u-236	1.8779E-05	u-236	1.1291E-04	u-236	1.1947E-04	u-236	1.2067E-04	u-236	1.1852E-04
u-238	2.2014E-02	u-238	2.0893E-02	u-238	2.0721E-02	u-238	2.0643E-02	u-238	2.0681E-02
pu-238	3.9591E-07	pu-238	3.3592E-06	pu-238	5.7679E-06	pu-238	7.1191E-06	pu-238	6.5197E-06
pu-239	6.0080E-05	pu-239	8.2042E-05	pu-239	9.3844E-05	pu-239	1.0856E-04	pu-239	1.1237E-04
pu-240	3.0090E-05	pu-240	4.8865E-05	pu-240	5.5755E-05	pu-240	5.9750E-05	pu-240	6.1451E-05
pu-241	8.2373E-06	pu-241	1.9312E-05	pu-241	2.6717E-05	pu-241	3.1481E-05	pu-241	3.0057E-05
pu-242	2.8953E-06	pu-242	1.0878E-05	pu-242	1.6727E-05	pu-242	1.7670E-05	pu-242	1.5346E-05
am-241	1.1394E-06	am-241	2.4274E-06	am-241	2.9588E-06	am-241	3.3857E-06	am-241	3.3245E-06
am-243	1.8343E-07	am-243	1.6446E-06	am-243	3.5433E-06	am-243	4.3119E-06	am-243	3.5608E-06
np-237	1.1820E-06	np-237	8.9794E-06	np-237	1.2348E-05	np-237	1.3835E-05	np-237	1.3028E-05
mo-95	1.1846E-05	mo-95	4.8236E-05	mo-95	5.5368E-05	mo-95	5.7227E-05	mo-95	5.4391E-05
tc-99	1.2770E-05	tc-99	4.8713E-05	tc-99	5.7114E-05	tc-99	5.8626E-05	tc-99	5.5425E-05
ru-101	1.2071E-05	ru-101	4.5325E-05	ru-101	5.4050E-05	ru-101	5.5792E-05	ru-101	5.2582E-05
rh-103	9.3130E-06	rh-103	2.6530E-05	rh-103	3.0186E-05	rh-103	3.0919E-05	rh-103	2.9806E-05
ag-109	1.5925E-06	ag-109	3.8299E-06	ag-109	5.1908E-06	ag-109	5.6225E-06	ag-109	5.2277E-06
cs-133	1.4091E-05	cs-133	5.2010E-05	cs-133	5.9569E-05	cs-133	6.0628E-05	cs-133	5.7770E-05
sm-147	2.2814E-06	sm-147	7.0633E-06	sm-147	6.9677E-06	sm-147	6.6474E-06	sm-147	6.5196E-06
sm-149	1.8732E-08	sm-149	5.4246E-08	sm-149	6.4430E-08	sm-149	7.8322E-08	sm-149	8.4438E-08
sm-150	2.4384E-06	sm-150	9.3869E-06	sm-150	1.1313E-05	sm-150	1.1792E-05	sm-150	1.1151E-05
sm-151	1.0567E-07	sm-151	3.4006E-07	sm-151	4.0252E-07	sm-151	4.7410E-07	sm-151	4.9572E-07
sm-152	1.8562E-06	sm-152	5.1202E-06	sm-152	5.6516E-06	sm-152	5.5805E-06	sm-152	5.3196E-06
nd-143	8.3795E-06	nd-143	3.1297E-05	nd-143	3.4029E-05	nd-143	3.5809E-05	nd-143	3.5667E-05
nd-145	7.1624E-06	nd-145	2.8015E-05	nd-145	3.1676E-05	nd-145	3.2074E-05	nd-145	3.0621E-05
eu-153	9.5742E-07	eu-153	3.9352E-06	eu-153	5.0452E-06	eu-153	5.3119E-06	eu-153	4.9732E-06
gd-155	1.1242E-08	gd-155	5.8059E-08	gd-155	8.7854E-08	gd-155	1.0494E-07	gd-155	1.0012E-07
o	4.4782E-02	o	4.4744E-02	o	4.4743E-02	o	4.4742E-02	o	4.4743E-02

Table A3 (Cont'd)

Data Set No. 42		Data Set No. 43		Data Set No. 44		Data Set No. 45	
Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year		Cooling : 1 Year	
Region# : 6		Region# : 7		Region# : 8		Region# : 9	
Burnup = 39.610GWd/tM		Burnup = 35.172GWd/tM		Burnup = 28.034GWd/tM		Burnup = 10.935GWd/tM	
u-234	4.4053E-06	u-234	4.7290E-06	u-234	5.2743E-06	u-234	8.4967E-07
u-235	2.1240E-04	u-235	2.5636E-04	u-235	3.3906E-04	u-235	4.7942E-05
u-236	1.1461E-04	u-236	1.0892E-04	u-236	9.6955E-05	u-236	1.9093E-05
u-238	2.0752E-02	u-238	2.0831E-02	u-238	2.0952E-02	u-238	2.1953E-02
pu-238	5.4505E-06	pu-238	4.2608E-06	pu-238	2.6019E-06	pu-238	6.8245E-07
pu-239	1.1501E-04	pu-239	1.1389E-04	pu-239	1.1112E-04	pu-239	7.4602E-05
pu-240	5.6927E-05	pu-240	5.2463E-05	pu-240	4.2812E-05	pu-240	3.4753E-05
pu-241	2.8695E-05	pu-241	2.5554E-05	pu-241	2.0001E-05	pu-241	1.1791E-05
pu-242	1.2426E-05	pu-242	9.5884E-06	pu-242	5.7014E-06	pu-242	4.0054E-06
am-241	3.2401E-06	am-241	2.9909E-06	am-241	2.4621E-06	am-241	1.4979E-06
am-243	2.7026E-06	am-243	1.8540E-06	am-243	8.8087E-07	am-243	4.0952E-07
np-237	1.1862E-05	np-237	1.0296E-05	np-237	7.8097E-06	np-237	1.7401E-06
mo-95	5.0053E-05	mo-95	4.4523E-05	mo-95	3.6555E-05	mo-95	1.3392E-05
tc-99	5.0713E-05	tc-99	4.5562E-05	tc-99	3.6985E-05	tc-99	1.4691E-05
ru-101	4.7724E-05	ru-101	4.2528E-05	ru-101	3.4057E-05	ru-101	1.4068E-05
rh-103	2.7753E-05	rh-103	2.5393E-05	rh-103	2.1176E-05	rh-103	1.0790E-05
ag-109	4.5743E-06	ag-109	3.8874E-06	ag-109	2.8365E-06	ag-109	1.9838E-06
cs-133	5.3247E-05	cs-133	4.8312E-05	cs-133	3.9766E-05	cs-133	1.6075E-05
sm-147	6.3411E-06	sm-147	6.0944E-06	sm-147	5.5237E-06	sm-147	2.3798E-06
sm-149	8.6026E-08	sm-149	8.6962E-08	sm-149	9.0403E-08	sm-149	2.6642E-08
sm-150	1.0044E-05	sm-150	8.9726E-06	sm-150	7.1744E-06	sm-150	2.8798E-06
sm-151	4.8702E-07	sm-151	4.7418E-07	sm-151	4.5654E-07	sm-151	1.4456E-07
sm-152	4.9360E-06	sm-152	4.5492E-06	sm-152	3.8247E-06	sm-152	2.0662E-06
nd-143	3.4181E-05	nd-143	3.2130E-05	nd-143	2.7990E-05	nd-143	9.6495E-06
nd-145	2.8378E-05	nd-145	2.5879E-05	nd-145	2.1522E-05	nd-145	8.0952E-06
eu-153	4.4407E-06	eu-153	3.8541E-06	eu-153	2.8836E-06	eu-153	1.2432E-06
gd-155	8.7631E-08	gd-155	7.2071E-08	gd-155	5.1440E-08	gd-155	1.6411E-08
o	4.4743E-02	o	4.4743E-02	o	4.4744E-02	o	4.4782E-02

Table A3 (Cont'd)

Data Set No. 46		Data Set No. 47		Data Set No. 48		Data Set No. 49		Data Set No. 50	
Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years	
Region# : 1		Region# : 2		Region# : 3		Region# : 4		Region# : 5	
Burnup = 9.402GWd/tM		Burnup = 37.343GWd/tM		Burnup = 44.799GWd/tM		Burnup = 46.484GWd/tM		Burnup = 43.760GWd/tM	
u-234	9.0581E-07	u-234	4.8851E-06	u-234	4.3140E-06	u-234	4.1808E-06	u-234	4.3315E-06
u-235	4.7051E-05	u-235	1.8310E-04	u-235	1.3808E-04	u-235	1.4795E-04	u-235	1.7497E-04
u-236	1.8792E-05	u-236	1.1293E-04	u-236	1.1949E-04	u-236	1.2070E-04	u-236	1.1854E-04
u-238	2.2014E-02	u-238	2.0893E-02	u-238	2.0721E-02	u-238	2.0643E-02	u-238	2.0681E-02
pu-238	3.9592E-07	pu-238	3.3036E-06	pu-238	5.6718E-06	pu-238	6.9986E-06	pu-238	6.4108E-06
pu-239	6.0073E-05	pu-239	8.2034E-05	pu-239	9.3835E-05	pu-239	1.0855E-04	pu-239	1.1236E-04
pu-240	3.0080E-05	pu-240	4.8902E-05	pu-240	5.5911E-05	pu-240	5.9979E-05	pu-240	6.1625E-05
pu-241	6.7946E-06	pu-241	1.5930E-05	pu-241	2.2037E-05	pu-241	2.5967E-05	pu-241	2.4793E-05
pu-242	2.8953E-06	pu-242	1.0878E-05	pu-242	1.6727E-05	pu-242	1.7670E-05	pu-242	1.5346E-05
am-241	2.5700E-06	am-241	5.7831E-06	am-241	7.6037E-06	am-241	8.8595E-06	am-241	8.5501E-06
am-243	1.8336E-07	am-243	1.6439E-06	am-243	3.5420E-06	am-243	4.3103E-06	am-243	3.5594E-06
np-237	1.1941E-06	np-237	9.0061E-06	np-237	1.2382E-05	np-237	1.3875E-05	np-237	1.3067E-05
mo-95	1.1860E-05	mo-95	4.8287E-05	mo-95	5.5441E-05	mo-95	5.7311E-05	mo-95	5.4474E-05
tc-99	1.2770E-05	tc-99	4.8712E-05	tc-99	5.7113E-05	tc-99	5.8625E-05	tc-99	5.5424E-05
ru-101	1.2071E-05	ru-101	4.5325E-05	ru-101	5.4050E-05	ru-101	5.5792E-05	ru-101	5.2582E-05
rh-103	9.3134E-06	rh-103	2.6531E-05	rh-103	3.0188E-05	rh-103	3.0921E-05	rh-103	2.9808E-05
ag-109	1.5925E-06	ag-109	3.8299E-06	ag-109	5.1908E-06	ag-109	5.6225E-06	ag-109	5.2277E-06
cs-133	1.4091E-05	cs-133	5.2010E-05	cs-133	5.9569E-05	cs-133	6.0628E-05	cs-133	5.7770E-05
sm-147	3.2155E-06	sm-147	9.8947E-06	sm-147	9.9599E-06	sm-147	9.6611E-06	sm-147	9.5113E-06
sm-149	1.8732E-08	sm-149	5.4246E-08	sm-149	6.4430E-08	sm-149	7.8322E-08	sm-149	8.4438E-08
sm-150	2.4384E-06	sm-150	9.3869E-06	sm-150	1.1313E-05	sm-150	1.1792E-05	sm-150	1.1151E-05
sm-151	1.0247E-07	sm-151	3.2974E-07	sm-151	3.9031E-07	sm-151	4.5972E-07	sm-151	4.8068E-07
sm-152	1.8563E-06	sm-152	5.1203E-06	sm-152	5.6517E-06	sm-152	5.5806E-06	sm-152	5.3197E-06
nd-143	8.3795E-06	nd-143	3.1297E-05	nd-143	3.4029E-05	nd-143	3.5809E-05	nd-143	3.5667E-05
nd-145	7.1624E-06	nd-145	2.8015E-05	nd-145	3.1676E-05	nd-145	3.2074E-05	nd-145	3.0621E-05
eu-153	9.5745E-07	eu-153	3.9353E-06	eu-153	5.0453E-06	eu-153	5.3121E-06	eu-153	4.9734E-06
gd-155	3.9360E-08	gd-155	2.0147E-07	gd-155	3.1243E-07	gd-155	3.7193E-07	gd-155	3.4969E-07
o	4.4782E-02	o	4.4744E-02	o	4.4743E-02	o	4.4742E-02	o	4.4743E-02

Table A3 (Cont'd)

Data Set No. 51		Data Set No. 52		Data Set No. 53		Data Set No. 54	
Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years	
Region# : 6		Region# : 7		Region# : 8		Region# : 9	
Burnup = 39.610GWd/tM		Burnup = 35.172GWd/tM		Burnup = 28.034GWd/tM		Burnup = 10.935GWd/tM	
u-234	4.5770E-06	u-234	4.8633E-06	u-234	5.3563E-06	u-234	8.7151E-07
u-235	2.1241E-04	u-235	2.5637E-04	u-235	3.3908E-04	u-235	4.7951E-05
u-236	1.1464E-04	u-236	1.0894E-04	u-236	9.6973E-05	u-236	1.9107E-05
u-238	2.0752E-02	u-238	2.0831E-02	u-238	2.0952E-02	u-238	2.1953E-02
pu-238	5.3629E-06	pu-238	4.1951E-06	pu-238	2.5635E-06	pu-238	6.8371E-07
pu-239	1.1500E-04	pu-239	1.1388E-04	pu-239	1.1111E-04	pu-239	7.4594E-05
pu-240	5.7038E-05	pu-240	5.2521E-05	pu-240	4.2823E-05	pu-240	3.4747E-05
pu-241	2.3669E-05	pu-241	2.1078E-05	pu-241	1.6498E-05	pu-241	9.7258E-06
pu-242	1.2426E-05	pu-242	9.5884E-06	pu-242	5.7015E-06	pu-242	4.0054E-06
am-241	8.2285E-06	am-241	7.4325E-06	am-241	5.9379E-06	am-241	3.5466E-06
am-243	2.7016E-06	am-243	1.8533E-06	am-243	8.8054E-07	am-243	4.0937E-07
np-237	1.1899E-05	np-237	1.0330E-05	np-237	7.8371E-06	np-237	1.7565E-06
mo-95	5.0131E-05	mo-95	4.4595E-05	mo-95	3.6617E-05	mo-95	1.3415E-05
tc-99	5.0712E-05	tc-99	4.5561E-05	tc-99	3.6984E-05	tc-99	1.4691E-05
ru-101	4.7724E-05	ru-101	4.2528E-05	ru-101	3.4057E-05	ru-101	1.4068E-05
rh-103	2.7755E-05	rh-103	2.5394E-05	rh-103	2.1177E-05	rh-103	1.0791E-05
ag-109	4.5743E-06	ag-109	3.8874E-06	ag-109	2.8365E-06	ag-109	1.9838E-06
cs-133	5.3247E-05	cs-133	4.8312E-05	cs-133	3.9766E-05	cs-133	1.6075E-05
sm-147	9.2384E-06	sm-147	8.8762E-06	sm-147	8.0336E-06	sm-147	3.4643E-06
sm-149	8.6026E-08	sm-149	8.6962E-08	sm-149	9.0403E-08	sm-149	2.6642E-08
sm-150	1.0044E-05	sm-150	8.9726E-06	sm-150	7.1744E-06	sm-150	2.8798E-06
sm-151	4.7225E-07	sm-151	4.5980E-07	sm-151	4.4269E-07	sm-151	1.4017E-07
sm-152	4.9361E-06	sm-152	4.5493E-06	sm-152	3.8249E-06	sm-152	2.0663E-06
nd-143	3.4181E-05	nd-143	3.2130E-05	nd-143	2.7990E-05	nd-143	9.6495E-06
nd-145	2.8378E-05	nd-145	2.5879E-05	nd-145	2.1522E-05	nd-145	8.0952E-06
eu-153	4.4408E-06	eu-153	3.8543E-06	eu-153	2.8837E-06	eu-153	1.2432E-06
gd-155	3.0132E-07	gd-155	2.4553E-07	gd-155	1.6692E-07	gd-155	5.8320E-08
o	4.4743E-02	o	4.4743E-02	o	4.4744E-02	o	4.4782E-02

Table A3 (Cont'd)

Data Set No. 55 Cooling : 5 years Region# : 1 Burnup = 40.000GWd/tM		Data Set No. 56 Cooling : 5 years Region# : 2 Burnup = 40.000GWd/tM		Data Set No. 57 Cooling : 5 years Region# : 3 Burnup = 40.000GWd/tM		Data Set No. 58 Cooling : 5 years Region# : 4 Burnup = 40.000GWd/tM		Data Set No. 59 Cooling : 5 years Region# : 5 Burnup = 40.000GWd/tM	
u-234	5.0581E-07	u-234	4.6844E-06	u-234	4.6311E-06	u-234	4.5758E-06	u-234	4.5592E-06
u-235	1.1809E-06	u-235	1.5694E-04	u-235	1.7915E-04	u-235	2.0054E-04	u-235	2.0728E-04
u-236	2.1143E-05	u-236	1.1591E-04	u-236	1.1523E-04	u-236	1.1525E-04	u-236	1.1491E-04
u-238	2.1252E-02	u-238	2.0847E-02	u-238	2.0806E-02	u-238	2.0762E-02	u-238	2.0748E-02
pu-238	3.0132E-06	pu-238	3.8990E-06	pu-238	4.5127E-06	pu-238	5.2103E-06	pu-238	5.4295E-06
pu-239	6.6743E-05	pu-239	8.2723E-05	pu-239	9.4722E-05	pu-239	1.0919E-04	pu-239	1.1414E-04
pu-240	6.3645E-05	pu-240	5.0326E-05	pu-240	5.3213E-05	pu-240	5.6085E-05	pu-240	5.7124E-05
pu-241	1.7260E-05	pu-241	1.7602E-05	pu-241	1.9978E-05	pu-241	2.2792E-05	pu-241	2.3598E-05
pu-242	3.6914E-05	pu-242	1.2861E-05	pu-242	1.2953E-05	pu-242	1.2847E-05	pu-242	1.2720E-05
am-241	5.6121E-06	am-241	6.1142E-06	am-241	7.0024E-06	am-241	8.0292E-06	am-241	8.3283E-06
am-243	8.9638E-06	am-243	2.1496E-06	am-243	2.4217E-06	am-243	2.6900E-06	am-243	2.7729E-06
np-237	3.9404E-06	np-237	9.9345E-06	np-237	1.0864E-05	np-237	1.1774E-05	np-237	1.1989E-05
mo-95	4.1714E-05	mo-95	5.1084E-05	mo-95	5.0553E-05	mo-95	4.9953E-05	mo-95	4.9759E-05
tc-99	4.8941E-05	tc-99	5.1780E-05	tc-99	5.1573E-05	tc-99	5.1314E-05	tc-99	5.1222E-05
ru-101	5.1073E-05	ru-101	4.8476E-05	ru-101	4.8399E-05	ru-101	4.8269E-05	ru-101	4.8212E-05
rh-103	2.7745E-05	rh-103	2.7738E-05	rh-103	2.8027E-05	rh-103	2.8100E-05	rh-103	2.8057E-05
ag-109	7.9643E-06	ag-109	4.2392E-06	ag-109	4.4219E-06	ag-109	4.5909E-06	ag-109	4.6365E-06
cs-133	4.9820E-05	cs-133	5.4916E-05	cs-133	5.4490E-05	cs-133	5.3981E-05	cs-133	5.3804E-05
sm-147	6.5998E-06	sm-147	1.0032E-05	sm-147	9.7681E-06	sm-147	9.4532E-06	sm-147	9.3463E-06
sm-149	3.5650E-08	sm-149	5.6248E-08	sm-149	6.6485E-08	sm-149	7.9068E-08	sm-149	8.3535E-08
sm-150	9.0195E-06	sm-150	9.9689E-06	sm-150	1.0082E-05	sm-150	1.0184E-05	sm-150	1.0220E-05
sm-151	2.0148E-07	sm-151	3.2777E-07	sm-151	3.8502E-07	sm-151	4.5177E-07	sm-151	4.7531E-07
sm-152	5.5582E-06	sm-152	5.3606E-06	sm-152	5.2184E-06	sm-152	5.0584E-06	sm-152	5.0013E-06
nd-143	1.5914E-05	nd-143	3.1779E-05	nd-143	3.3111E-05	nd-143	3.4194E-05	nd-143	3.4500E-05
nd-145	2.3563E-05	nd-145	2.9474E-05	nd-145	2.9166E-05	nd-145	2.8786E-05	nd-145	2.8652E-05
eu-153	4.8174E-06	eu-153	4.2983E-06	eu-153	4.4066E-06	eu-153	4.4867E-06	eu-153	4.5009E-06
gd-155	2.8513E-07	gd-155	2.3201E-07	gd-155	2.5813E-07	gd-155	2.9022E-07	gd-155	3.0202E-07
o	4.4780E-02	o	4.4744E-02	o	4.4744E-02	o	4.4743E-02	o	4.4743E-02

Table A3 (Cont'd)

Data Set No. 60		Data Set No. 61		Data Set No. 62		Data Set No. 63	
Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years	
Region# : 6		Region# : 7		Region# : 8		Region# : 9	
Burnup = 40.000GWd/tM		Burnup = 40.000GWd/tM		Burnup = 40.000GWd/tM		Burnup = 40.000GWd/tM	
u-234	4.5546E-06	u-234	4.5516E-06	u-234	4.5499E-06	u-234	5.9620E-07
u-235	2.0901E-04	u-235	2.0986E-04	u-235	2.1059E-04	u-235	3.9757E-06
u-236	1.1496E-04	u-236	1.1505E-04	u-236	1.1501E-04	u-236	2.1024E-05
u-238	2.0744E-02	u-238	2.0742E-02	u-238	2.0741E-02	u-238	2.1207E-02
pu-238	5.4989E-06	pu-238	5.5269E-06	pu-238	5.5557E-06	pu-238	4.3983E-06
pu-239	1.1547E-04	pu-239	1.1616E-04	pu-239	1.1670E-04	pu-239	9.1742E-05
pu-240	5.7422E-05	pu-240	5.7604E-05	pu-240	5.7739E-05	pu-240	7.3579E-05
pu-241	2.3832E-05	pu-241	2.3950E-05	pu-241	2.4035E-05	pu-241	2.3768E-05
pu-242	1.2686E-05	pu-242	1.2674E-05	pu-242	1.2636E-05	pu-242	3.1171E-05
am-241	8.4132E-06	am-241	8.4552E-06	am-241	8.4869E-06	am-241	7.9505E-06
am-243	2.7967E-06	am-243	2.8117E-06	am-243	2.8188E-06	am-243	9.1144E-06
np-237	1.2065E-05	np-237	1.2091E-05	np-237	1.2124E-05	np-237	5.0117E-06
mo-95	4.9703E-05	mo-95	4.9673E-05	mo-95	4.9651E-05	mo-95	4.1560E-05
tc-99	5.1195E-05	tc-99	5.1181E-05	tc-99	5.1171E-05	tc-99	4.8654E-05
ru-101	4.8197E-05	ru-101	4.8187E-05	ru-101	4.8181E-05	ru-101	5.0196E-05
rh-103	2.8034E-05	rh-103	2.8022E-05	rh-103	2.8013E-05	rh-103	2.8498E-05
ag-109	4.6488E-06	ag-109	4.6549E-06	ag-109	4.6597E-06	ag-109	7.9248E-06
cs-133	5.3754E-05	cs-133	5.3724E-05	cs-133	5.3706E-05	cs-133	4.9106E-05
sm-147	9.3162E-06	sm-147	9.2976E-06	sm-147	9.2861E-06	sm-147	6.4543E-06
sm-149	8.4360E-08	sm-149	8.4823E-08	sm-149	8.5396E-08	sm-149	4.7045E-08
sm-150	1.0232E-05	sm-150	1.0239E-05	sm-150	1.0243E-05	sm-150	9.7870E-06
sm-151	4.8156E-07	sm-151	4.8492E-07	sm-151	4.8786E-07	sm-151	2.8408E-07
sm-152	4.9857E-06	sm-152	4.9767E-06	sm-152	4.9702E-06	sm-152	5.1201E-06
nd-143	3.4571E-05	nd-143	3.4602E-05	nd-143	3.4641E-05	nd-143	2.0316E-05
nd-145	2.8613E-05	nd-145	2.8591E-05	nd-145	2.8577E-05	nd-145	2.3517E-05
eu-153	4.5046E-06	eu-153	4.5060E-06	eu-153	4.5076E-06	eu-153	4.9911E-06
gd-155	3.0537E-07	gd-155	3.0717E-07	gd-155	3.0893E-07	gd-155	3.5349E-07
o	4.4743E-02	o	4.4743E-02	o	4.4743E-02	o	4.4779E-02

Table A3 (Cont'd)

Data Set No. 64 Cooling : 5 years Region# : 1&9 Burnup = 20.000GWd/tM		Data Set No. 65 Cooling : 5 years Region# : 2-8 Burnup = 20.000GWd/tM		Data Set No. 66 Cooling : 5 years Region# : 1&9 Burnup = 30.000GWd/tM		Data Set No. 67 Cooling : 5 years Region# : 2-8 Burnup = 30.000GWd/tM		Data Set No. 68 Cooling : 5 years Region# : 1&9 Burnup = 40.000GWd/tM		Data Set No. 69 Cooling : 5 years Region# : 2-8 Burnup = 40.000GWd/tM	
u-234	6.9964E-07	u-234	6.0728E-06	u-234	5.8859E-07	u-234	5.3171E-06	u-234	5.4701E-07	u-234	4.6286E-06
u-235	1.5846E-05	u-235	4.3763E-04	u-235	4.6918E-06	u-235	2.9119E-04	u-235	2.0925E-06	u-235	1.8067E-04
u-236	2.2410E-05	u-236	7.8080E-05	u-236	2.2077E-05	u-236	1.0067E-04	u-236	2.1005E-05	u-236	1.1498E-04
u-238	2.1736E-02	u-238	2.1111E-02	u-238	2.1489E-02	u-238	2.0967E-02	u-238	2.1235E-02	u-238	2.0802E-02
pu-238	1.3121E-06	pu-238	8.3312E-07	pu-238	2.4284E-06	pu-238	2.2952E-06	pu-238	3.6102E-06	pu-238	4.6024E-06
pu-239	7.4078E-05	pu-239	9.0283E-05	pu-239	7.4464E-05	pu-239	9.5907E-05	pu-239	7.6813E-05	pu-239	9.6665E-05
pu-240	5.2179E-05	pu-240	2.7445E-05	pu-240	6.1319E-05	pu-240	4.2279E-05	pu-240	6.7165E-05	pu-240	5.3243E-05
pu-241	1.5440E-05	pu-241	9.5844E-06	pu-241	1.8248E-05	pu-241	1.5680E-05	pu-241	1.9944E-05	pu-241	2.0356E-05
pu-242	1.3285E-05	pu-242	2.2899E-06	pu-242	2.4522E-05	pu-242	6.5424E-06	pu-242	3.4508E-05	pu-242	1.2880E-05
am-241	4.6525E-06	am-241	2.9619E-06	am-241	5.6625E-06	am-241	5.1281E-06	am-241	6.5702E-06	am-241	7.1288E-06
am-243	2.0970E-06	am-243	2.0715E-07	am-243	5.2931E-06	am-243	9.0772E-07	am-243	9.1126E-06	am-243	2.4533E-06
np-237	2.9190E-06	np-237	4.4761E-06	np-237	3.7974E-06	np-237	7.6704E-06	np-237	4.4910E-06	np-237	1.1031E-05
mo-95	2.3162E-05	mo-95	2.7406E-05	mo-95	3.2435E-05	mo-95	3.9473E-05	mo-95	4.1525E-05	mo-95	5.0358E-05
tc-99	2.6299E-05	tc-99	2.7078E-05	tc-99	3.7824E-05	tc-99	3.9671E-05	tc-99	4.8816E-05	tc-99	5.1523E-05
ru-101	2.5881E-05	ru-101	2.4373E-05	ru-101	3.8444E-05	ru-101	3.6453E-05	ru-101	5.0718E-05	ru-101	4.8379E-05
rh-103	1.8583E-05	rh-103	1.5821E-05	rh-103	2.3872E-05	rh-103	2.2522E-05	rh-103	2.8110E-05	rh-103	2.7923E-05
ag-109	4.0459E-06	ag-109	1.6181E-06	ag-109	6.0694E-06	ag-109	2.9236E-06	ag-109	7.9245E-06	ag-109	4.4209E-06
cs-133	2.8052E-05	cs-133	2.9687E-05	cs-133	3.9318E-05	cs-133	4.2734E-05	cs-133	4.9489E-05	cs-133	5.4381E-05
sm-147	4.4215E-06	sm-147	6.1599E-06	sm-147	5.5044E-06	sm-147	8.2518E-06	sm-147	6.5271E-06	sm-147	9.6906E-06
sm-149	4.8320E-08	sm-149	9.2472E-08	sm-149	4.5571E-08	sm-149	8.2115E-08	sm-149	3.9647E-08	sm-149	6.6765E-08
sm-150	5.3175E-06	sm-150	5.2739E-06	sm-150	7.3817E-06	sm-150	7.7272E-06	sm-150	9.3495E-06	sm-150	1.0077E-05
sm-151	1.6582E-07	sm-151	3.5027E-07	sm-151	1.9106E-07	sm-151	3.7421E-07	sm-151	2.3225E-07	sm-151	3.8681E-07
sm-152	3.5033E-06	sm-152	2.9404E-06	sm-152	4.5729E-06	sm-152	4.1910E-06	sm-152	5.3779E-06	sm-152	5.1842E-06
nd-143	1.3466E-05	nd-143	2.1912E-05	nd-143	1.5273E-05	nd-143	2.8860E-05	nd-143	1.7698E-05	nd-143	3.3070E-05
nd-145	1.3651E-05	nd-145	1.6299E-05	nd-145	1.8870E-05	nd-145	2.3161E-05	nd-145	2.3589E-05	nd-145	2.9119E-05
eu-153	2.5779E-06	eu-153	1.7490E-06	eu-153	3.8291E-06	eu-153	3.0551E-06	eu-153	4.9632E-06	eu-153	4.4164E-06
gd-155	1.4073E-07	gd-155	8.6753E-08	gd-155	2.2813E-07	gd-155	1.6330E-07	gd-155	3.1413E-07	gd-155	2.6184E-07
o	4.4781E-02	o	4.4745E-02	o	4.4780E-02	o	4.4744E-02	o	4.4779E-02	o	4.4743E-02

Table A3 (Cont'd)

Data Set No. 70		Data Set No. 71		Data Set No. 72		Data Set No. 73		Data Set No. 74		Data Set No. 75	
Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years		Cooling : 5 years	
Region# : 1&9		Region# : 2-8		Region# : 1&9		Region# : 2-8		Region# : 1&9		Region# : 2-8	
Burnup = 20.000GWd/tM		Burnup = 20.000GWd/tM		Burnup = 30.000GWd/tM		Burnup = 30.000GWd/tM		Burnup = 40.000GWd/tM		Burnup = 40.000GWd/tM	
u-234	7.0596E-07	u-234	5.9709E-06	u-234	6.1868E-07	u-234	5.2188E-06	u-234	5.8686E-07	u-234	4.5692E-06
u-235	1.9626E-05	u-235	4.4970E-04	u-235	8.0349E-06	u-235	3.1033E-04	u-235	3.4724E-06	u-235	2.0453E-04
u-236	2.2113E-05	u-236	7.8529E-05	u-236	2.2147E-05	u-236	1.0050E-04	u-236	2.0940E-05	u-236	1.1447E-04
u-238	2.1720E-02	u-238	2.1083E-02	u-238	2.1468E-02	u-238	2.0927E-02	u-238	2.1213E-02	u-238	2.0753E-02
pu-238	1.4733E-06	pu-238	1.0234E-06	pu-238	2.8789E-06	pu-238	2.7365E-06	pu-238	4.2483E-06	pu-238	5.3753E-06
pu-239	8.3583E-05	pu-239	1.0185E-04	pu-239	8.5779E-05	pu-239	1.1020E-04	pu-239	8.9018E-05	pu-239	1.1307E-04
pu-240	5.3578E-05	pu-240	2.9282E-05	pu-240	6.5287E-05	pu-240	4.4898E-05	pu-240	7.2143E-05	pu-240	5.6628E-05
pu-241	1.7057E-05	pu-241	1.0933E-05	pu-241	2.0821E-05	pu-241	1.7905E-05	pu-241	2.3034E-05	pu-241	2.3351E-05
pu-242	1.2505E-05	pu-242	2.4546E-06	pu-242	2.2676E-05	pu-242	6.6569E-06	pu-242	3.1814E-05	pu-242	1.2634E-05
am-241	5.1597E-06	am-241	3.3835E-06	am-241	6.5308E-06	am-241	5.8623E-06	am-241	7.6816E-06	am-241	8.2295E-06
am-243	2.2365E-06	am-243	2.6639E-07	am-243	5.3944E-06	am-243	1.0814E-06	am-243	9.1477E-06	am-243	2.7459E-06
np-237	3.1911E-06	np-237	5.0296E-06	np-237	4.2433E-06	np-237	8.4505E-06	np-237	4.9541E-06	np-237	1.1962E-05
mo-95	2.3084E-05	mo-95	2.7123E-05	mo-95	3.2510E-05	mo-95	3.8979E-05	mo-95	4.1430E-05	mo-95	4.9698E-05
tc-99	2.6210E-05	tc-99	2.6962E-05	tc-99	3.7761E-05	tc-99	3.9453E-05	tc-99	4.8646E-05	tc-99	5.1218E-05
ru-101	2.5796E-05	ru-101	2.4384E-05	ru-101	3.8198E-05	ru-101	3.6404E-05	ru-101	5.0303E-05	ru-101	4.8216E-05
rh-103	1.8530E-05	rh-103	1.5885E-05	rh-103	2.4257E-05	rh-103	2.2527E-05	rh-103	2.8288E-05	rh-103	2.7925E-05
ag-109	4.0237E-06	ag-109	1.7341E-06	ag-109	6.0653E-06	ag-109	3.0825E-06	ag-109	7.8920E-06	ag-109	4.5971E-06
cs-133	2.7871E-05	cs-133	2.9457E-05	cs-133	3.9138E-05	cs-133	4.2305E-05	cs-133	4.9101E-05	cs-133	5.3783E-05
sm-147	4.3599E-06	sm-147	5.9775E-06	sm-147	5.5267E-06	sm-147	7.9493E-06	sm-147	6.4261E-06	sm-147	9.3257E-06
sm-149	5.2450E-08	sm-149	1.0546E-07	sm-149	5.2195E-08	sm-149	9.5769E-08	sm-149	4.4836E-08	sm-149	8.0535E-08
sm-150	5.4157E-06	sm-150	5.3071E-06	sm-150	7.6600E-06	sm-150	7.8222E-06	sm-150	9.6705E-06	sm-150	1.0193E-05
sm-151	1.8728E-07	sm-151	3.9591E-07	sm-151	2.3516E-07	sm-151	4.3419E-07	sm-151	2.7174E-07	sm-151	4.6048E-07
sm-152	3.3967E-06	sm-152	2.8801E-06	sm-152	4.4185E-06	sm-152	4.0697E-06	sm-152	5.1643E-06	sm-152	4.9983E-06
nd-143	1.4135E-05	nd-143	2.1939E-05	nd-143	1.7335E-05	nd-143	2.9238E-05	nd-143	1.9563E-05	nd-143	3.4217E-05
nd-145	1.3597E-05	nd-145	1.6104E-05	nd-145	1.8844E-05	nd-145	2.2828E-05	nd-145	2.3512E-05	nd-145	2.8678E-05
eu-153	2.6289E-06	eu-153	1.8228E-06	eu-153	3.9546E-06	eu-153	3.1507E-06	eu-153	5.0253E-06	eu-153	4.4977E-06
gd-155	1.5301E-07	gd-155	9.9093E-08	gd-155	2.5692E-07	gd-155	1.8792E-07	gd-155	3.4529E-07	gd-155	2.9876E-07
o	4.4781E-02	o	4.4745E-02	o	4.4780E-02	o	4.4744E-02	o	4.4779E-02	o	4.4743E-02

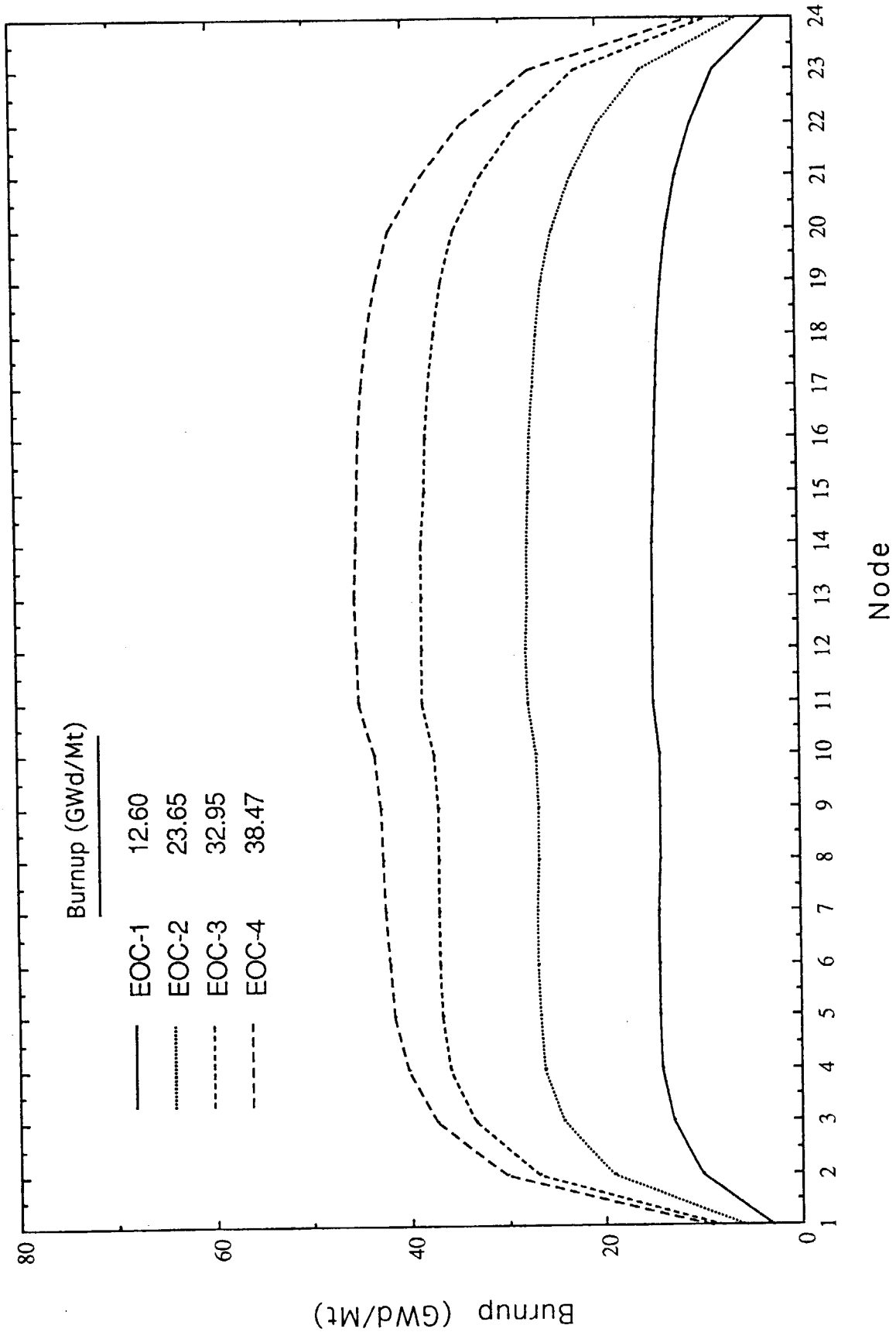


Fig. A1 Burnup Profile based on Haling Calculation

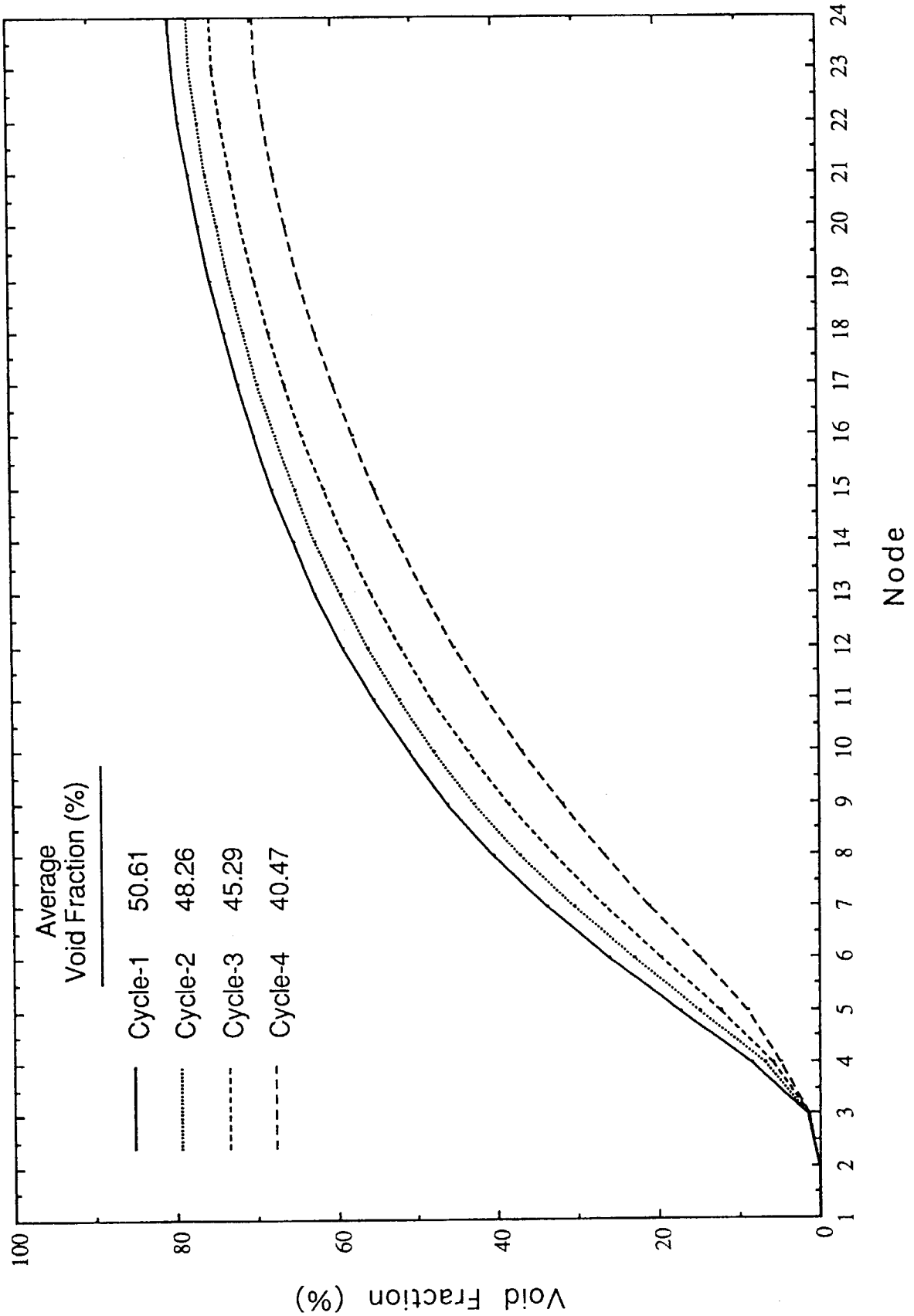
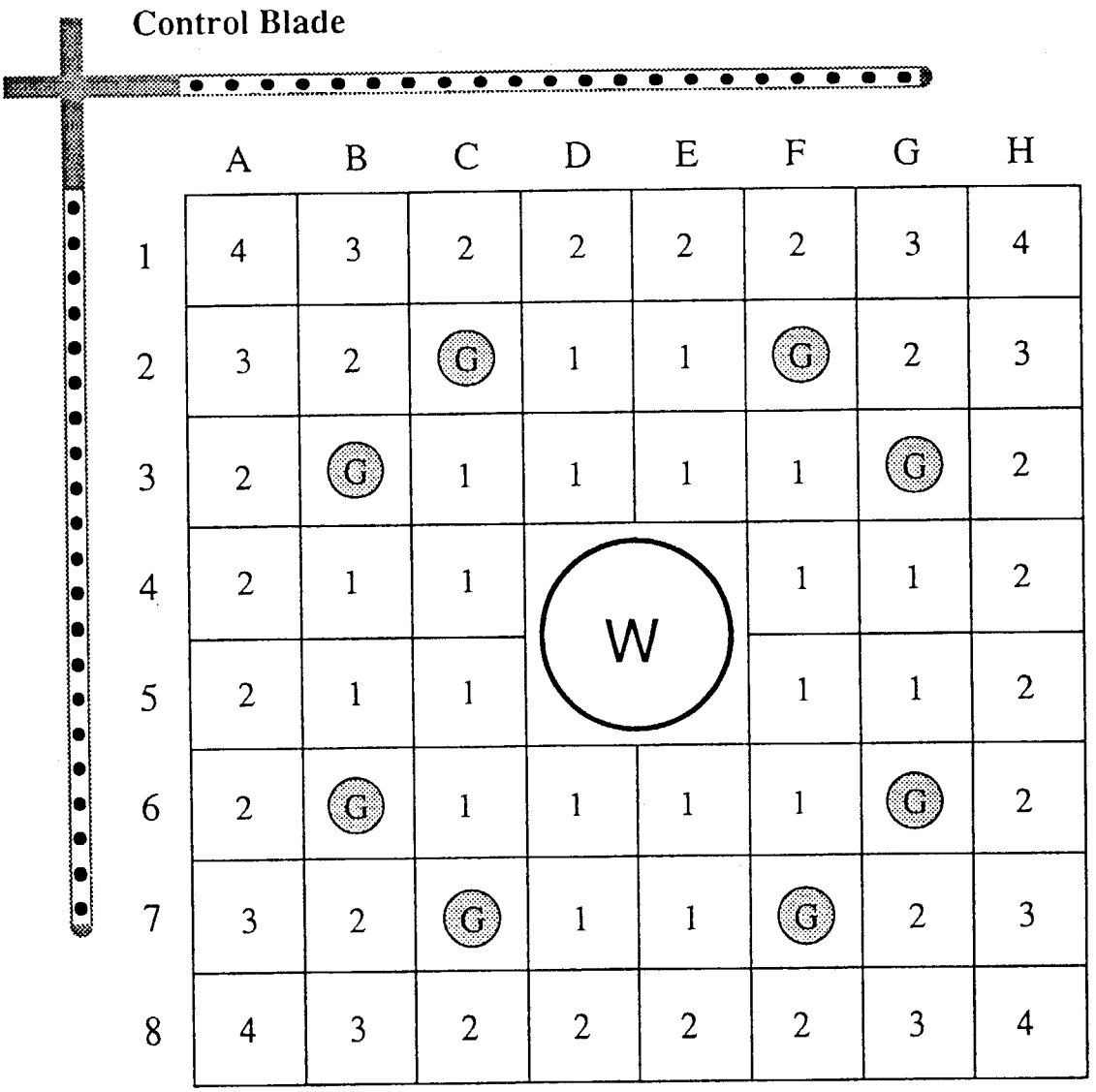


Fig. A2 Void Profile based on Haling Calculation



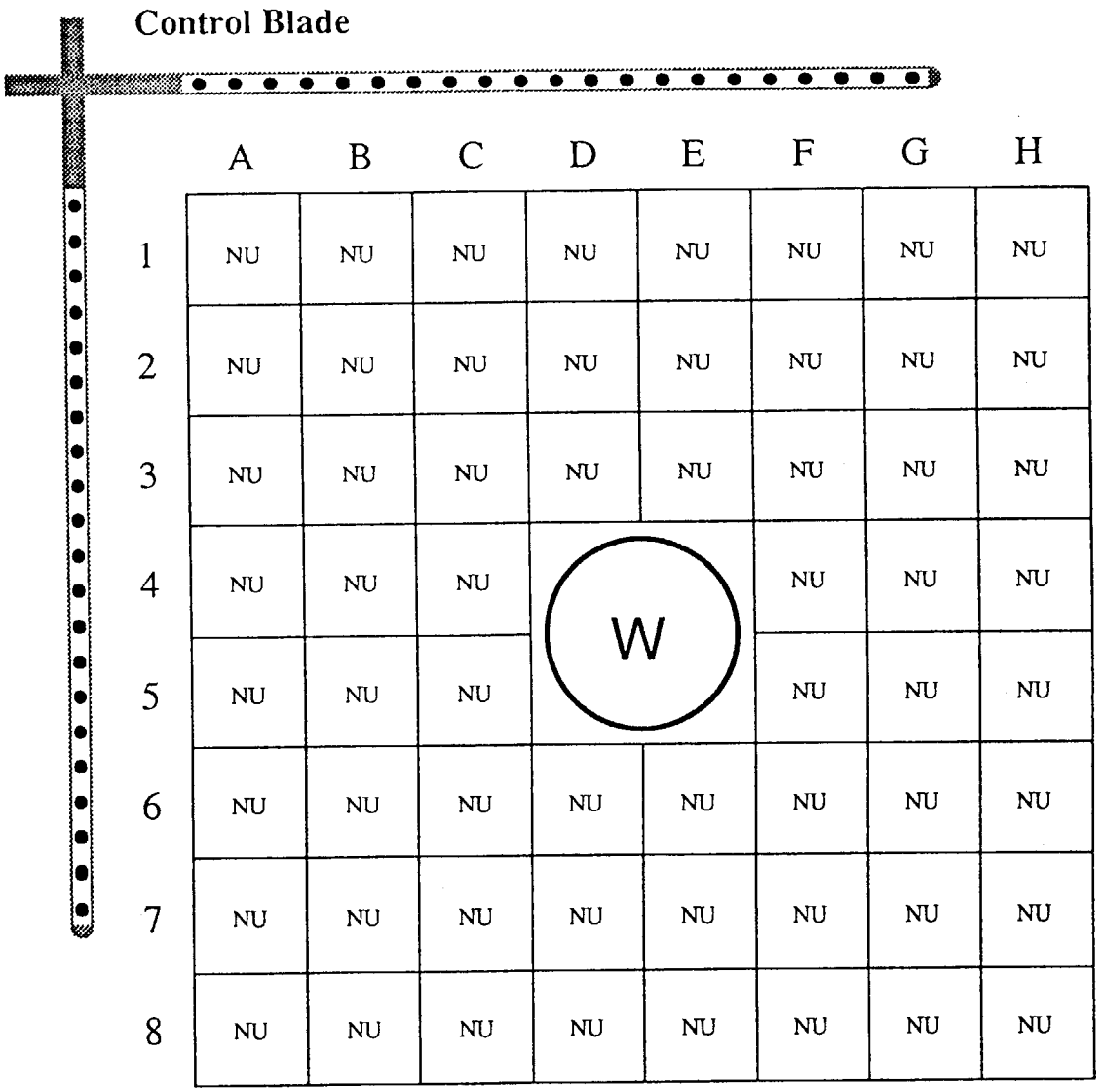
Gd Rod

(W) Large Water Rod

Rod Type#	U-235 Enrichment	Gadolinia Content	No. of Rods
1	4.9 w/o		20
2	3.6		20
3	3.0		8
4	2.3		4
G	3.0	4.5 w/o	8
W	Water Rod		1

Assembly Average U-235 Enrichment = 3.80 w/o

Fig. A3 Typical BWR Fuel Assembly



Rod Type#	U-235 Enrichment	No. of Rods
NU	0.71 w/o	60
W	Water Rod	1

Assembly Average U-235 Enrichment = 0.71 w/o

Fig. A4 Blanket Fuel Assembly

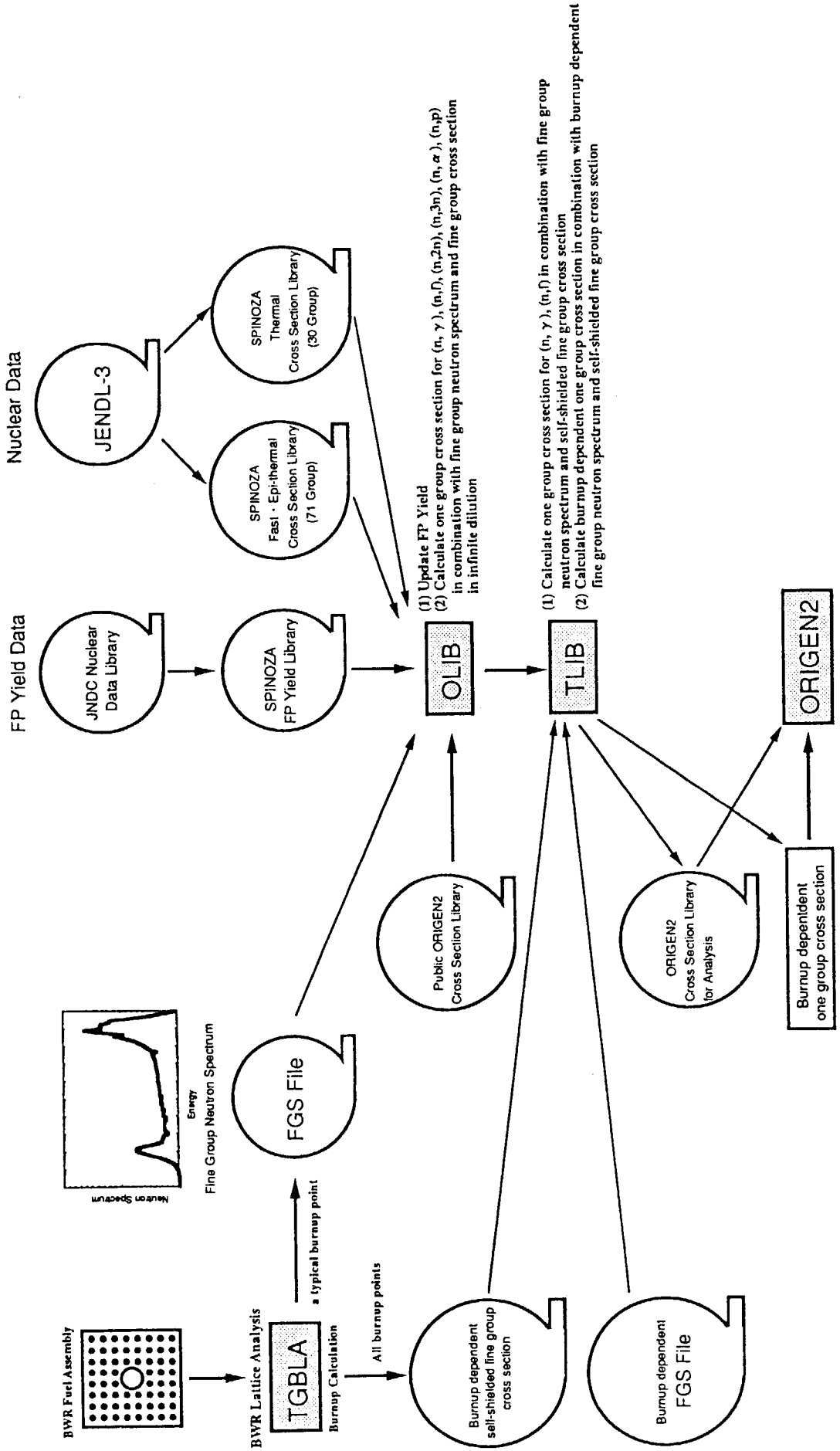


Fig. A5 Schematic Flow Diagram in SPINOZA System

Appendix II

Final Results from Participants

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<Participant A>

Date: 1996-03-06
ABB Atom, Reactor Core Engineering (BRM)
Waldemar Lipiec
E-mail: atowali@ato.abb.se, Telefax: +46 (0)21 147862
Computer codes: PHOENIX4/KENO-Va

1.4112	+/- 0.0007
1.1944	+/- 0.0007
1.1134	+/- 0.0007
1.0278	+/- 0.0007
1.1850	+/- 0.0007
1.0930	+/- 0.0006
0.9981	+/- 0.0007
0.9906	+/- 0.0005
1.1905	+/- 0.0006
1.0735	+/- 0.0006
0.9598	+/- 0.0005
1.2000	+/- 0.0007
1.0967	+/- 0.0006
0.9994	+/- 0.0005
1.1007	+/- 0.0007
1.1057	+/- 0.0006
1.2609	+/- 0.0006
1.1684	+/- 0.0006
1.0737	+/- 0.0006
1.2733	+/- 0.0007
1.1949	+/- 0.0007
1.1154	+/- 0.0006

2.34506e-03
8.51080e-02
5.32307e-01
2.88884e-01
5.26129e-02
1.84290e-02
1.22988e-02
6.91892e-03
1.09668e-03
6.47068e-04
1.39362e-02
8.33429e-02
2.97647e-01
2.19243e-01
1.34737e-01
1.31759e-01
9.75591e-02
2.11293e-02
3.29627e-04
5.48561e-03
2.32490e-02
1.30550e-01
2.14989e-01
1.79932e-01
2.15918e-01
1.83570e-01
4.59756e-02
2.95977e-04
5.13694e-03
1.53198e-02
8.88194e-02
1.80514e-01
1.80439e-01
2.36731e-01

2.28506e-01
6.42377e-02
2.60279e-03
5.58826e-02
4.16910e-01
3.73786e-01
7.73470e-02
3.02487e-02
2.44444e-02
1.51603e-02
3.61852e-03

ANALYSIS ENVIRONMENT

Institute: ABB Atom, Reactor Core Engineering (BRM), Sweden.
Participants: Waldemar Lipiec
Data Library: PHOENIX LIBRARY: 17 MAY 1994, 89 neutron and 18 gamma group cross section library based on ENDF/B-VI Release 2.
Computer Codes: PHOENIX4/KENO-Va
No. of Groups: 89 neutron energy groups in PHOENIX4 and 13 neutron groups in KENO-Va.

Comments:

The 3D geometry of the system has been modelled in KENO assuming four radial regions at the fuel level: water hole, fuel, box and water gap. The gas plenum of the fuel rods has been smeared and there is no box in the reflector regions.

The P0 transport corrected cross sections for KENO-Va has been generated by PHOENIX, a 2D lattice code using the collision probability and S4 methods.

The number of neutron histories was 500 000 for each case.

<Participant B>

1. Date : Mar. 27, 1996
2. Institute : TOSHIBA
3. Contact Person : Yoshihira ANDO
4. E-mail Adress : raa@rcg.nel.rdc.toshiba.co.jp
5. Computer code : TGBLA/ALEX
6. Multiplication Factors : 1.38064 (Case#1)
   1.18149 (Case#2)
   1.10075 (Case#3)
   1.01971 (Case#4)
   1.16941 (Case#5)
   1.08104 (Case#6)
   0.99238 (Case#7)
   0.98272 (Case#8)
   1.17097 (Case#9)
   1.06016 (Case#10)
   0.95171 (Case#11)
   1.18201 (Case#12)
   1.08395 (Case#13)
   0.99111 (Case#14)
   1.08718 (Case#15)
   1.08968 (Case#16)
   1.23744 (Case#17)
   1.14792 (Case#18)
   1.05713 (Case#19)
   1.24989 (Case#20)
   1.17305 (Case#21)
   1.09821 (Case#22)
7. Fission Density 0.001488 (Region#1 of Case#1)
   0.057309 (Region#2 of Case#1)

```
0.441192 (Region#3 of Case#1)
0.362337 (Region#4 of Case#1)
0.078869 (Region#5 of Case#1)
0.027707 (Region#6 of Case#1)
0.019164 (Region#7 of Case#1)
0.010445 (Region#8 of Case#1)
0.001490 (Region#9 of Case#1)
0.000050 (Region#1 of Case#5)
0.001450 (Region#2 of Case#5)
0.036761 (Region#3 of Case#5)
0.265279 (Region#4 of Case#5)
0.245140 (Region#5 of Case#5)
0.156689 (Region#6 of Case#5)
0.155759 (Region#7 of Case#5)
0.115256 (Region#8 of Case#5)
0.023614 (Region#9 of Case#5)
0.000001 (Region#1 of Case#6)
0.000041 (Region#2 of Case#6)
0.004987 (Region#3 of Case#6)
0.132412 (Region#4 of Case#6)
0.225669 (Region#5 of Case#6)
0.188382 (Region#6 of Case#6)
0.220595 (Region#7 of Case#6)
0.185112 (Region#8 of Case#6)
0.042800 (Region#9 of Case#6)
0.000000 (Region#1 of Case#7)
0.000003 (Region#2 of Case#7)
0.001225 (Region#3 of Case#7)
0.074178 (Region#4 of Case#7)
0.182382 (Region#5 of Case#7)
0.187832 (Region#6 of Case#7)
0.250642 (Region#7 of Case#7)
0.240043 (Region#8 of Case#7)
0.063695 (Region#9 of Case#7)
0.002400 (Region#1 of Case#14)
0.058869 (Region#2 of Case#14)
0.438707 (Region#3 of Case#14)
0.359689 (Region#4 of Case#14)
0.079046 (Region#5 of Case#14)
0.028072 (Region#6 of Case#14)
0.019719 (Region#7 of Case#14)
0.011093 (Region#8 of Case#14)
0.002405 (Region#9 of Case#14)
```

8. Analysis Environment

(1) Institute and country

TOSHIBA,Japan

(2) Participants

Yoshihira Ando

(3) Neutron data library

*Data source:

U-235,U-238,Pu-239,Pu-240	: ENDF/B-V
Pu-241,Pu-242	: ENDF/B-IV
All other actinides	: JENDL-3.1
FPs	: mostly ENDF/B-IV

- FP chain : 45 explicit FPs + 1 lumped pseudo FP model
  S.Iijima et al., J.Nucl.Sci.Technol., 19,96 (1982)
  * Mo-95 is not treated as an explicit FP in this FP model, while it has been taken into account as one of major contributors to the pseudo FP poisoning as described in the above reference.
  * Zr is treated as Zircalloy-2 (including Cr, Fe)
- FP yield data : based on Rider&Meek compilation
  B.F.Rider et al., NEDO-12154-2 (GE) (1977)

(4) Neutron data processing

CRECT-J

(5) Neutron energy groups

Fast&Epi-thermal (0.6825ev-10MeV) : 68 groups (GAM-type)

Thermal (0-0.6825ev) : 30 groups (THERMOS-type)

(6) Code system

TGBLA/ALEX

(7) Geometry modeling

Radial : Modified 2D Transport (1D Transport + 2D Diffusion)

Axial : 2D Diffusion (XY)

- Note for our calculations;

I calculate assembly averaged 3 group constants(fast,epi-thermal,thermal) for fuel, gas plenum, end plug and reflector region through TGBLA with reflective boundary conditions and calculate multiplication factors by 2D diffusion code ALEX using 3 group constants estimated by TGBLA calculation

(8) Omitted nuclides etc

Omitted nuclide : Mo-95

Lumped nuclides : Cr, Fe, Zr

(9) Employed convergence limit

1.0E-05 : Outer Iteration

1.0E-05 : Inner Iteration

(10) Other related information

(A) K-Infinity Values by TGBLA Unit Cell CalCulation

Geometory Data : Assembly Pitch = 15.24 cm

Fuel Rod Cell Pitch = 1.63 cm

Outer Radius of Fuel Rod = 0.615 cm

Cladding Thickness of Fuel Rod = 0.086 cm

Outer Radius of Water Rod = 1.6 cm

Cladding Thickness of Water Rod = 0.1 cm

Channel Box Thickness = 0.254 cm

No. of Fuel Rod Rows = 8

*Case-wise K-infinity

Case#	Region#	K-inf	Cooling	FP	BP	VP	Burnup
All	Gas Plenum	0.00000					
"	End Plug	0.00000					
"	Reflector	0.00000					
1	1&9	0.80179					0 GWd/Mt
"	2-8	1.38473					"
2	1	0.81306	1 Year	Yes	Yes	Yes	20 GWd/Mt
"	2	1.17739	"	"	"	"	"
"	3	1.15794	"	"	"	"	"
"	4	1.16998	"	"	"	"	"
"	5	1.18861	"	"	"	"	"
"	6	1.20829	"	"	"	"	"
"	7	1.22888	"	"	"	"	"
"	8	1.26024	"	"	"	"	"
"	9	0.85430	"	"	"	"	"
3	1	0.78822	"	"	"	"	30 GWd/Mt
"	2	1.07106	"	"	"	"	"
"	3	1.04424	"	"	"	"	"
"	4	1.06704	"	"	"	"	"
"	5	1.09420	"	"	"	"	"
"	6	1.12352	"	"	"	"	"
"	7	1.15572	"	"	"	"	"
"	8	1.19757	"	"	"	"	"
"	9	0.84320	"	"	"	"	"
4	1	0.76720	"	"	"	"	40 GWd/Mt
"	2	0.96556	"	"	"	"	"
"	3	0.93858	"	"	"	"	"
"	4	0.97124	"	"	"	"	"
"	5	0.99649	"	"	"	"	"
"	6	1.03547	"	"	"	"	"
"	7	1.06987	"	"	"	"	"

"	8	1.12932	"	"	"	"	"
"	9	0.82750	"	"	"	"	"
5	1	0.80705	5 Years	"	"	"	20 GWd/Mt
"	2	1.16687	"	"	"	"	"
"	3	1.14349	"	"	"	"	"
"	4	1.15442	"	"	"	"	"
"	5	1.17445	"	"	"	"	"
"	6	1.19623	"	"	"	"	"
"	7	1.21891	"	"	"	"	"
"	8	1.25325	"	"	"	"	"
"	9	0.84673	"	"	"	"	"
6	1	0.77748	"	"	"	"	30 GWd/Mt
"	2	1.05147	"	"	"	"	"
"	3	1.01794	"	"	"	"	"
"	4	1.03904	"	"	"	"	"
"	5	1.06846	"	"	"	"	"
"	6	1.10135	"	"	"	"	"
"	7	1.13695	"	"	"	"	"
"	8	1.18478	"	"	"	"	"
"	9	0.82986	"	"	"	"	"
7	1	0.75296	"	"	"	"	40 GWd/Mt
"	2	0.93861	"	"	"	"	"
"	3	0.90100	"	"	"	"	"
"	4	0.93033	"	"	"	"	"
"	5	0.95898	"	"	"	"	"
"	6	1.00216	"	"	"	"	"
"	7	1.04178	"	"	"	"	"
"	8	1.10933	"	"	"	"	"
"	9	0.80902	"	"	"	"	"
8	1	0.63481	"	"	No	"	"
"	2	0.91007	"	"	"	"	"
"	3	0.94955	"	"	"	"	"
"	4	0.98591	"	"	"	"	"
"	5	0.99631	"	"	"	"	"
"	6	0.99899	"	"	"	"	"
"	7	1.00031	"	"	"	"	"
"	8	1.00130	"	"	"	"	"
"	9	0.73451	"	"	"	"	"
9	1&9	0.72610	"	"	"	40VH	20 GWd/Mt
"	2-8	1.17443	"	"	"	"	"
10	1&9	0.69057	"	"	"	"	30 GWd/Mt
"	2-8	1.06326	"	"	"	"	"
11	1&9	0.67924	"	"	"	"	40 GWd/Mt
"	2-8	0.95455	"	"	"	"	"
12	1&9	0.76903	"	"	"	70VH	20 GWd/Mt
"	2-8	1.18553	"	"	"	"	"
13	1&9	0.73719	"	"	"	"	30 GWd/Mt
"	2-8	1.08711	"	"	"	"	"
14	1&9	0.72546	"	"	"	"	40 GWd/Mt
"	2-8	0.99403	"	"	"	"	"
15	1	0.78561	"	No	Yes	Yes	"
"	2	1.03747	"	"	"	"	"
"	3	1.01492	"	"	"	"	"
"	4	1.04927	"	"	"	"	"
"	5	1.07415	"	"	"	"	"
"	6	1.10958	"	"	"	"	"
"	7	1.14037	"	"	"	"	"
"	8	1.19323	"	"	"	"	"
"	9	0.84924	"	"	"	"	"
16	1	0.71913	"	"	No	"	"
"	2	1.01391	"	"	"	"	"
"	3	1.05549	"	"	"	"	"
"	4	1.09351	"	"	"	"	"

"	5	1.10442	"	"	"	"	"
"	6	1.10717	"	"	"	"	"
"	7	1.10853	"	"	"	"	"
"	8	1.10962	"	"	"	"	"
"	9	0.83109	"	"	"	"	"
17	1&9	0.78866	"	"	"	40VH	20 GWd/Mt
"	2-8	1.24108	"	"	"	"	"
18	1&9	0.76739	"	"	"	"	30 GWd/Mt
"	2-8	1.15138	"	"	"	"	"
19	1&9	0.76930	"	"	"	"	40 GWd/Mt
"	2-8	1.06031	"	"	"	"	"
20	1&9	0.83466	"	"	"	70VH	20 GWd/Mt
"	2-8	1.25358	"	"	"	"	"
21	1&9	0.82012	"	"	"	"	30 GWd/Mt
"	2-8	1.17652	"	"	"	"	"
22	1&9	0.82057	"	"	"	"	40 GWd/Mt
"	2-8	1.10147	"	"	"	"	"

Where, FP : Yes/no --- with/without FP
BP : " --- " Axial Burnup Profile
VP : " --- " Axial Void Profile
40VH/70VH --- 40%/70% Void Profile along Axial Axis
Burnup : Assembly Average Burnup

(B) Calculational conditons in 2D diffusion calculation

(i) Boundary Condition
X-axis : Vaccum (Albedo = 0.0)
Y-axis : Reflective (Albedo = 1.0)

(ii) No. of Meshes
X-axis : 308
Y-axis : 3

(iii) Mesh Sizes along X-axis
Water (Top) : 5.0cm x 4, 1.0cm x 10
(Bottom) : 1.0cm x 10, 5.0cm x 4
Gas Plenum : 1.0cm x 10
End Plug (Top&Bottom) : 0.4cm x 5
Fuel (Region1) : 1.524cm x 10
(Region2) : 1.524cm x 30
(Region3) : 1.524cm x 80
(Region4) : 1.524cm x 60
(Region5) : 1.524cm x 20
(Region6,7,8,9) : 1.524cm x 10

(iV) Mesh Sizes along Y-axis
All Regions : 1.524cm x 3

<Participant C>

1. March 28th, 1996 (Revised. June 6th, 1996)

2. Consejo de Seguridad Nuclear (CSN)
Madrid (Spain)

3. Jose M. Conde

4. E-mail: jmcl@csn.es
Fax: 34-1-3460588

5. CASMO-SIMULATE (Studsvik's CMS System)

6. Case 1: 1.40067
7. Case 2: 1.20224
8. Case 3: 1.12035
9. Case 4: 1.03770

10. Case 5: 1.18974
11. Case 6: 1.09980
12. Case 7: 1.00902
13. Case 8: 0.99953
14. Case 9: 1.19279
15. Case 10: 1.08027
16. Case 11: 0.96831
17. Case 12: 1.20378
18. Case 13: 1.10416
19. Case 14: 1.0082
20. Case 15: 1.10351
21. Case 16: 1.10681
22. Case 17: 1.25789
23. Case 18: 1.16729
24. Case 19: 1.07380
25. Case 20: 1.27051
26. Case 21: 1.19281
27. Case 22: 1.1158

Fractional Fission densities for Case 1
28. 1.6508E-3
29. 5.7850E-2
30. 4.4052E-1
31. 3.6159E-1
32. 7.8904E-2
33. 2.7807E-2
34. 1.9316E-2
35. 1.0708E-2
36. 1.6529E-3

Fractional Fission Densities for Case 5
37. 7.5833E-5
38. 2.0496E-3
39. 4.4937E-2
40. 2.8143E-1
41. 2.4215E-1
42. 1.5042E-1
43. 1.4713E-1
44. 1.0851E-1
45. 2.3303E-2

Fractional Fission Densities for Case 6
46. 2.0833E-6
47. 5.6666E-5
48. 5.8921E-3
49. 1.4095E-1
50. 2.2850E-1
51. 1.8645E-1
52. 2.1523E-1
53. 1.7989E-1
54. 4.3250E-2

Fractional Fission Densities for Case 7
55. 4.1667E-7
56. 4.5833E-6
57. 1.2908E-3
58. 7.5601E-2
59. 1.8361E-1
60. 1.8757E-1
61. 2.4873E-1
62. 2.3796E-1
63. 6.5239E-2

Fractional Fission Densities for Case 14
64. 0.003
65. 0.059
66. 0.439
67. 0.360
68. 0.079
69. 0.028
70. 0.020
71. 0.011
72. 0.003

73. The calculations have been performed by the Nuclear Engineering Division of the Consejo de Seguridad Nuclear (CSN) of Spain. Authors are Jose M. Conde and M. Recio.

The cross sections used for the analysis have been obtained with CASMO-3 version 4.7 for each of the compositions included in the benchmark specification. The usual BWR modelling guidelines of CASMO have been followed. The E4LTJB7 nuclear data library has been used. This 70 energy group library takes data mainly from ENDF-B/4, although some data comes from ENDF-B/5 and JEF-2.

Cross sections have been calculated for all the fuel segments and for the top and bottom axial reflectors for each of the 22 cases, using the specific CASMO model for the reflector.

The nuclides Mo-95, Tc-99 and Ru-101 are not included in the nuclear data library.

The cross sections obtained with CASMO are functionalized by means of the TABLES code, and used in the SIMULATE-3 nodal code. The SIMULATE model includes a fuel bundle with mirror reflection in all four radial directions. The active length is divided into 24 nodes 15.24 cm high, and one reflector node of the appropiate composition is added on top and bottom. A convergence criteria of 1E-5 on the eigenvalue has been used for all cases.

<Participant D>

March 29, 1996
Japan Atomic Energy Research Institute
Kenya SUYAMA
kenya@cyclone.tokai.jaeri.go.jp :(phone 81-22-282-5943 fax 81-22-282-6479)
SRAC93 / CITATION
1.3963
1.1928
1.1091
1.0250
1.1802
1.0886
0.9963
0.9873
1.1822
1.0695
0.9567
1.1944
1.0933
0.9966
1.0983
1.1020
1.2523
1.1630
1.0695
1.2663
1.1885
1.1116
1.83379E-03 CASE-1 FISSION DENSITY

```
6.38185E-02
4.20187E-01
3.60837E-01
8.60397E-02
3.11631E-02
2.19519E-02
1.22561E-02
1.91262E-03
2.62274E-04 CASE-5 FISSION DENSITY
6.61655E-03
6.81551E-02
2.83478E-01
2.32816E-01
1.43113E-01
1.39719E-01
1.03300E-01
2.25408E-02
9.82612E-06 CASE-6 FISSION DENSITY
2.62263E-04
9.26805E-03
1.48488E-01
2.28692E-01
1.83272E-01
2.10362E-01
1.76344E-01
4.33016E-02
1.05460E-06 CASE-7 FISSION DENSITY
2.78383E-05
2.28868E-03
8.35107E-02
1.87613E-01
1.85625E-01
2.43008E-01
2.32513E-01
6.54120E-02
2.83121E-03 CASE-14 FISSION DENSITY
6.55624E-02
4.17971E-01
3.57380E-01
8.61089E-02
3.15601E-02
2.25922E-02
1.30149E-02
2.97925E-03
```

** Institute **
Japan Atomic Energy Research Institute, JAPAN

** Participants **
Kenya SUYAMA

** Neutron Data Libraries **
JENDL-3.2

** Neutron data processing code or method **

With SRAC code, I made the 20-group constant library for CITATION using the collision probability method(PIJ routine). In the calculation, the resonance absorption is treated by PEACO routine(ultra fine resonance absorption calculation).

After making the library, I analyzed the eigenvalue problem with 3-dimensional(full modeled) calculation of CITATION. In that modeling, there is no omitted region.

<Participant E>

29 February 1996 (Revised 18 November 1996)
British Nuclear Fuels plc, Risley, Warrington, England
Peter R Thorne, Russell L Bowden
Fax: +44 (0) 1925 832161 E-mail: russb@amgmps.com
MONK7A
1.4162
1.1974
1.1139
1.0282
1.1858
1.0896
0.9991
0.9878
1.1900
1.0738
0.9593
1.2029
1.0983
1.0001
1.1015
1.1055
1.2643
1.1703
1.0753
1.2766
1.1975
1.1174
0.0020
0.0754
0.5036
0.3066
0.0606
0.0221
0.0175
0.0105
0.0016
0.0007
0.0190
0.1096
0.2610
0.2202
0.1338
0.1317
0.1032
0.0209
0.0010
0.0163
0.0443
0.1680
0.2037
0.1657
0.1975
0.1655
0.0380
0.0002
0.0036
0.0153
0.0824
0.1650
0.1767

0.2469
0.2461
0.0638
0.0040
0.0849
0.5146
0.2944
0.0593
0.0194
0.0137
0.0081
0.0016
British Nuclear Fuels plc, Risley, Warrington, England
Peter R Thorne, Russell L Bowden
MONK Continuous Energy Database, derived from UKNDL and JEF-2
MONK7A Monte Carlo method, using Superhistory powering 1 (continuous energy)
MONK7A on a Sun Sparc Server 1000E, using the Solaris 2.4 operating system
3-D modelling of requested system
No nuclides omitted from requested calculations
Calculations were performed using no convergence limit on the eigenvalue. The calculations were performed using 20 stages of 1000 source neutrons per stage. Typical eigenvalue uncertainty on the individual calculations was of the order of 0.0010. Three independent calculations were undertaken for each case requested, giving a mean uncertainty on the combined result of the order of 0.0006.

<Participant F>

April 03 1996 Burnup Credit Benchmark Phase IIIA
Institut f. Kernenergetik und Energiesysteme (IKE) Univ. of Stuttgart, Germany
Wolfgang Bernnat
w.bernnat@ike.uni-stuttgart.de Telefax: 0711 685 2118
Code: MCNP, Version 4A

1.40446	0.00081
1.19175	0.00091
1.10764	0.00087
1.02504	0.00089
1.17857	0.00087
1.08578	0.00087
0.99425	0.00088
0.98543	0.00085
1.18316	0.00086
1.06708	0.00084
0.95287	0.00077
1.19595	0.00086
1.09085	0.00086
0.99266	0.00086
1.09663	0.00087
1.10041	0.00078
1.25300	0.00079
1.16265	0.00083
1.06730	0.00074
1.26622	0.00080
1.18890	0.00086
1.10954	0.00091

region: 1 F.F.D.= 0.0017 err= 0.024
region: 2 F.F.D.= 0.0596 err= 0.006
region: 3 F.F.D.= 0.4434 err= 0.002
region: 4 F.F.D.= 0.3482 err= 0.002
region: 5 F.F.D.= 0.0859 err= 0.005
region: 6 F.F.D.= 0.0287 err= 0.008
region: 7 F.F.D.= 0.0196 err= 0.010

```
region: 8 F.F.D.=    0.0113 err=   0.014
region: 9 F.F.D.=    0.0016 err=   0.024
region: 1 F.F.D.=    0.0009 err=   0.051
region: 2 F.F.D.=    0.0210 err=   0.015
region: 3 F.F.D.=    0.0778 err=   0.007
region: 4 F.F.D.=    0.2556 err=   0.004
region: 5 F.F.D.=    0.2219 err=   0.004
region: 6 F.F.D.=    0.1470 err=   0.005
region: 7 F.F.D.=    0.1445 err=   0.005
region: 8 F.F.D.=    0.1091 err=   0.006
region: 9 F.F.D.=    0.0223 err=   0.010
region: 1 F.F.D.=    0.0004 err=   0.081
region: 2 F.F.D.=    0.0037 err=   0.035
region: 3 F.F.D.=    0.0107 err=   0.020
region: 4 F.F.D.=    0.1428 err=   0.005
region: 5 F.F.D.=    0.2249 err=   0.004
region: 6 F.F.D.=    0.1814 err=   0.005
region: 7 F.F.D.=    0.2128 err=   0.004
region: 8 F.F.D.=    0.1818 err=   0.005
region: 9 F.F.D.=    0.0414 err=   0.008
region: 1 F.F.D.=    0.0002 err=   0.123
region: 2 F.F.D.=    0.0015 err=   0.054
region: 3 F.F.D.=    0.0045 err=   0.030
region: 4 F.F.D.=    0.0759 err=   0.007
region: 5 F.F.D.=    0.1813 err=   0.005
region: 6 F.F.D.=    0.1879 err=   0.004
region: 7 F.F.D.=    0.2464 err=   0.004
region: 8 F.F.D.=    0.2390 err=   0.004
region: 9 F.F.D.=    0.0634 err=   0.007
region: 1 F.F.D.=    0.0045 err=   0.024
region: 2 F.F.D.=    0.1026 err=   0.007
region: 3 F.F.D.=    0.5009 err=   0.003
region: 4 F.F.D.=    0.2665 err=   0.004
region: 5 F.F.D.=    0.0670 err=   0.008
region: 6 F.F.D.=    0.0262 err=   0.013
region: 7 F.F.D.=    0.0189 err=   0.015
region: 8 F.F.D.=    0.0111 err=   0.020
region: 9 F.F.D.=    0.0023 err=   0.032
```

Institut f. Kernenergetik und Energiesysteme (IKE)
University of Stuttgart
pfaffenwaldring 31
70550 Stuttgart
Dr. Ing. Wolfgang Bernnat , Dipl. Phys. Margarete Mattes

Neutron data library: JEF-2.2
Neutron data processing code: NJOY/CRAY-C94 Version, most actual updates
The MCNP cross-section library was prepared at IKE Stuttgart, the point data were reconstructed by an accuracy of 0.1 %, practical no thinning was used for the MCNP-library
Geometry modeling: exact representation of geometry as specified (pin-wise)
Omitted nuclides: none
Statistical errors are given together with the multiplication factors and the fractional fission densities in the different regions.

<Participant G>

April 01 1996 Burnup Credit Benchmark Phase IIIA
Inst. f. Kernenergetik u. Energiesysteme (IKE)/Bundesamt f. Strahlenschutz (BFS)
Wolfgang Bernnat
w.bernnat@ike.uni-stuttgart.de Telefax: 0711 685 2118
Computer code: RESMOD (Cell-calculation) / KENOVA (transport calculation)

```
    1.4067          .0005
    1.1949          .0005
    1.1107          .0005
    1.0270          .0005
    1.1823          .0005
    1.0902          .0005
     .9984          .0004
     .9880          .0004
    1.1852          .0005
    1.0698          .0004
     .9564          .0004
    1.1964          .0005
    1.0941          .0004
     .9971          .0004
    1.0991          .0005
    1.1031          .0005
    1.2564          .0005
    1.1629          .0005
    1.0696          .0004
    1.2689          .0005
    1.1895          .0005
    1.1119          .0005
region: 1 F.F.D.=    .00128 err=   2.86000
region: 2 F.F.D.=    .04798 err=   1.48000
region: 3 F.F.D.=    .40241 err=    .51000
region: 4 F.F.D.=    .39487 err=    .46000
region: 5 F.F.D.=    .08921 err=    .89000
region: 6 F.F.D.=    .03083 err=   1.16000
region: 7 F.F.D.=    .02076 err=   1.33000
region: 8 F.F.D.=    .01103 err=   1.59000
region: 9 F.F.D.=    .00163 err=   2.39000
region: 1 F.F.D.=    .00030 err=   9.78000
region: 2 F.F.D.=    .00588 err=   7.31000
region: 3 F.F.D.=    .04910 err=   3.07000
region: 4 F.F.D.=    .27190 err=    .48000
region: 5 F.F.D.=    .23714 err=    .42000
region: 6 F.F.D.=    .15113 err=    .50000
region: 7 F.F.D.=    .14986 err=    .52000
region: 8 F.F.D.=    .11096 err=    .58000
region: 9 F.F.D.=    .02371 err=    .79000
region: 1 F.F.D.=    .00002 err=  26.32000
region: 2 F.F.D.=    .00050 err=  13.80000
region: 3 F.F.D.=    .01027 err=   4.76000
region: 4 F.F.D.=    .14347 err=    .70000
region: 5 F.F.D.=    .22419 err=    .30000
region: 6 F.F.D.=    .18328 err=    .31000
region: 7 F.F.D.=    .21346 err=    .31000
region: 8 F.F.D.=    .18071 err=    .38000
region: 9 F.F.D.=    .04408 err=    .53000
region: 1 F.F.D.=    .00004 err=  25.30000
region: 2 F.F.D.=    .00034 err=  26.12000
region: 3 F.F.D.=    .00268 err=  10.16000
region: 4 F.F.D.=    .07319 err=    .87000
region: 5 F.F.D.=    .17849 err=    .36000
region: 6 F.F.D.=    .18450 err=    .26000
region: 7 F.F.D.=    .24900 err=    .26000
region: 8 F.F.D.=    .24349 err=    .29000
region: 9 F.F.D.=    .06827 err=    .45000
region: 1 F.F.D.=    .00293 err=   2.37000
region: 2 F.F.D.=    .06645 err=   1.31000
region: 3 F.F.D.=    .46606 err=    .46000
region: 4 F.F.D.=    .33067 err=    .49000
region: 5 F.F.D.=    .07577 err=   1.07000
```

region: 6 F.F.D.= .02773 err= 1.41000
region: 7 F.F.D.= .01838 err= 1.56000
region: 8 F.F.D.= .00968 err= 2.00000
region: 9 F.F.D.= .00231 err= 2.60000

Institut f. Kernenergetik und Energiesysteme (IKE)
University of Stuttgart
Pfaffenwaldring 31
70550 Stuttgart
Germany
Dr. Ing. Wolfgang Bernnat , Dipl. Phys. Margarete Mattes

Bundesamt fuer Strahlenschutz (BFS)
P.O.box 100149
38201 Salzgitter
Dr. Ing. H. H. Schweer
Germany

Neutron data library: JEF-2.2
Neutron data processing code: NJOY/CRAY-C94 Version, most actual updates.
The transport calculations were performed by means of KENO-VA-version using 127 group cross-sections prepared by the spectral code RESMOD.
The spectral calculations are based on a 292 group library in AMPX-format prepared by IKE Stuttgart, based on JEF-2.2. This library is compatible to SCALE modules, but contains no resonance parameters. The resonance-range is treated by a 1D-transport method based on first collision probabilities.
13.000 energy-points were used for the resonance range from 3 eV to 1.2 keV.
For the range above 1.2 keV the Bondarenko method is used for shielding of resonances (SCALE-module BONAMI-2). For every of the specified material specification a cell calculation was performed. The neutron spectra, calculated by the spectral code were used to collaps the 292 group cross sections to 127 groups for the final KENO-calculations. The program system RSYST was used to collaps fission product cross sections and for the calculation of macroscopic data for the different fuel specifications as well as for cladding and moderator.
Geometry modeling: exact representation of geometry as specified (pin-wise)
Omitted nuclides: none
Statistical errors are given together with the multiplication factors and the fractional fission densities in the different regions. The statistical errors for the fractional fission densities are given in percent.

<Participant H>

Date : 1996-04-04
Japan Atomic Energy Research Institute (JAERI)
Hiroki Sakamoto
E-mail : sakamoto@cyclone.tokai.jaeri.go.jp
Computer Code : MCNP4A
1.40817 +/- 0.00030
1.19395 +/- 0.00036
1.11151 +/- 0.00044
1.02634 +/- 0.00042
1.18355 +/- 0.00035
1.09082 +/- 0.00038
0.99882 +/- 0.00043
0.98878 +/- 0.00033
1.18716 +/- 0.00032
1.07096 +/- 0.00030
0.95824 +/- 0.00031
1.19877 +/- 0.00032
1.09650 +/- 0.00030
0.99822 +/- 0.00028
1.10048 +/- 0.00042

```
1.10498    +/- 0.00032
1.25837    +/- 0.00031
1.16607    +/- 0.00029
1.07235    +/- 0.00029
1.27161    +/- 0.00032
1.19217    +/- 0.00031
1.11416    +/- 0.00029
1.71612e-03
6.38803e-02
4.48843e-01
3.51662e-01
7.70547e-02
2.67537e-02
1.82803e-02
1.02826e-02
1.52960e-03
6.95952e-04
1.59266e-02
7.43913e-02
2.57032e-01
2.28786e-01
1.46227e-01
1.45220e-01
1.09433e-01
2.22796e-02
3.06023e-04
6.12795e-03
2.09482e-02
1.42861e-01
2.17303e-01
1.79287e-01
2.12081e-01
1.79641e-01
4.14331e-02
1.98712e-04
2.22862e-03
6.99222e-03
7.96110e-02
1.76859e-01
1.82547e-01
2.45828e-01
2.41361e-01
6.43848e-02
2.68603e-03
6.34147e-02
4.45697e-01
3.38444e-01
8.17662e-02
3.02914e-02
2.19513e-02
1.29371e-02
2.82528e-03

"MY ANALISYS ENVIRONMENT"

Institute : Japan Atomic Energy Research Institute (JAERI), Japan.
Participants : Hiroki Sakamoto
Neutron Data Library : FSXLIB-J3R2 (A Continuous Energy Cross Section Library
                                for MCNP Based on JENDL-3.2 except for U-234)
                       ENDL85 (U-234 only)
Neutron Data Processing Code or Method : NJOY
Neutron Energy Groups : Continuous Energy
Comments :
```

A Monte Carlo neutron and photon transport code system ,MCNP4A, is able to analyze both critical and shielding problems. The code is able to simulate physical phenomena quite accurately since it has the capabilities of treating energy continuously and modeling 3-dimensional geometries accurately.

The number of neutron histories was 3,000,000 for each case.

<Participant I>

Date: Apr. 02, 1996

Institute: CE-SACLAY DMT/SERMA/LEPP, France

Contact person: Y. K. Lee

E-mail: lee@soleil.saclay.cea.fr (Telefax: 331 69 08 45 72)

Computer codes: APOLLO2 + TRIMARAN2 (cf: ICNC'95, pp.3.12)

Keff and standard deviations:

Case 1 :	1.4119	0.0011
Case 2 :	1.1925	0.0010
Case 3 :	1.1083	0.0012
Case 4 :	1.0241	0.0013
Case 5 :	1.1806	0.0010
Case 6 :	1.0856	0.0013
Case 7 :	0.9961	0.0012
Case 8 :	0.9884	0.0010
Case 9 :	1.1882	0.0015
Case 10 :	1.0733	0.0015
Case 11 :	0.9588	0.0015
Case 12 :	1.2007	0.0015
Case 13 :	1.0973	0.0015
Case 14 :	0.9982	0.0006
Case 15 :	1.0964	0.0013
Case 16 :	1.1018	0.0011
Case 17 :	1.2616	0.0015
Case 18 :	1.1673	0.0015
Case 19 :	1.0726	0.0015
Case 20 :	1.2714	0.0015
Case 21 :	1.1926	0.0015
Case 22 :	1.1155	0.0015

Fractional fission densities from region 1 to 9 (%) :

Case 1:	0.25	8.42	42.62	31.11	9.24	3.72	2.75	1.63	0.24
Case 5:	0.23	5.15	20.09	26.23	17.65	10.81	10.44	7.75	1.66
Case 6:	0.18	3.14	7.76	16.82	19.89	15.76	17.92	14.92	3.62
Case 7:	0.01	0.17	2.37	10.72	16.64	17.17	23.47	22.97	6.48
Case 14:	0.25	5.60	41.40	35.57	9.26	3.48	2.51	1.50	0.36

Institute and country: CE-SACLAY DMT/SERMA/LEPP, France

Participants: Y. K. Lee, C. Diop

Neutron data library: CEA93 (JEF2.2)

Neutron data processing code: THEMIS/NJOY

Neutron energy groups: 99

Computer codes: APOLLO2 + TRIMARAN2 (cf: ICNC'95, pp.3.12)

<Participant J>

1	Date		1996/3/6 (Revised 1996/5/14)		
2	Institute		PNC Tokai works, Japan		
3	Contact Person		Ichiro Nojiri		
4	E-mail address and Telefax		nojiri@tokai.pnc.go.jp Tel. 029 - 287 - 1111 (Extension 2639) Fax. 029 - 287 - 1841		
5	Computer Code		SCALE 4.2 (CSASIX - WAX - KENO V.a)		
	Multiplication Factors				
6	1.3979	14	1.1843	22	1.2539
7	1.1919	15	1.0689	23	1.1616
8	1.1049	16	0.9579	24	1.0705
9	1.0247	17	1.1947	25	1.2662
10	1.1759	18	1.0929	26	1.1900
11	1.0881	19	0.9975	27	1.1128
12	0.9956	20	1.0989		
13	0.9870	21	1.1029		
	Fractional fission densities				
	Case1		Case5		case6
28	0.0015	37	0.0012	46	0.0009
29	0.0643	38	0.0249	47	0.0152
30	0.4731	39	0.1252	48	0.0387
31	0.3489	40	0.2983	49	0.1756
32	0.0663	41	0.2046	50	0.2211
33	0.0212	42	0.1230	51	0.1665
34	0.0147	43	0.1167	52	0.1876
35	0.0087	44	0.0871	53	0.1531
36	0.0013	45	0.0190	54	0.0413
	Case7		Case14		
55	0.0005	64	0.0032		
56	0.0067	65	0.0721		
57	0.0154	66	0.4221		
58	0.0909	67	0.3505		
59	0.1837	68	0.0830		
60	0.1788	69	0.0309		
61	0.2345	70	0.0223		
62	0.2261	71	0.0128		
63	0.0634	72	0.0032		

73	Neutron Data Library	27 Burnup Library (ENDF/B-IV, V)
74	Neutron data processing code	BONAMI-S and NITAWL-II
75	Neutron Groups	27Groups
76	Geometory ModelSee	Fig1.1 and 1.2
77	Statistical errors	7E-04

<Participant K>

April 9, 1996 (Revised April 21, 1996)
E Mennerdahl Systems (EMS)
Dennis Mennerdahl
dennis.mennerdahl@ems.se (fax: +46 - 8 - 756 58 72)
SCALE 4.1

1,3906	0.00052
1,1881	0.00052
1,1075	0.00052
1,0241	0.00050
1,1770	0.00051
1,0885	0.00050
0,9936	0,00048

```
 0,9865     0,00044
 1,1802     0,00049
 1,0667     0,00048
 0,9537     0,00043
 1,1895     0,00050
 1,0900     0,00046
 0,9952     0,00046
 1,0971     0,00050
 1,1004     0,00049
 1,2484     0,00050
 1,1593     0,00050
 1,0685     0.00048
 1,2615     0,00051
 1,1850     0.00056
 1,1099     0,00050
  0,002
  0,063
  0,474
  0,350
  0,066
  0,022
  0,015
  0,008
  0,001
  0,000
  0,002
  0,042
  0,274
  0,236
  0,153
  0,152
  0,115
  0,025
  0,000
  0,000
  0,004
  0,126
  0,226
  0,190
  0,222
  0,187
  0,045
  0,000
  0,000
  0,002
  0,075
  0,183
  0,183
  0,248
  0,241
  0,068
  0,002
  0,041
  0,408
  0,388
  0,091
  0,032
  0,023
  0,013
  0,003
E Mennerdahl Systems, Sweden
Dennis Mennerdahl
SCALE 4.1.  27 group burnup library.  CSAS25 cross section preparation sequence.
```

The installation of SCALE 4.1 is based on the IBM mainframe version. The assembler language subroutines were replaced with FORTRAN subroutines.
Some minor changes were made to work with the Silicon Valley Systems FORTRAN compiler. The installation was carried out on various personal computers. The reported results were calculated on a standard personal computer based on the Intel Pentium (133 MHz) processor.
No modifications of the SCALE installation have been made since the OECD/NEA benchmark study on Burnup Credit started (early 1992).
All calculations were run using 1003 batches with 1000 neutrons per batch. 103 batches were skipped.

<Participant L>

1996/4/12
Institute of Nuclear Safety, NUPEC
Susumu Mitake
kyf01415@niftyserve.or.jp
KENO-V.a with MGCJL-JINS (137-group) library

1.39893	+-0.00055	
1.19179	+-0.00055	
1.10823	+-0.00058	
1.02563	+-0.00068	
1.17940	+-0.00059	
1.08827	+-0.00068	
0.99697	+-0.00070	
0.98765	+-0.00050	
1.18384	+-0.00053	
1.07063	+-0.00045	
0.95788	+-0.00047	
1.19586	+-0.00053	
1.09425	+-0.00052	
0.99731	+-0.00043	
1.10001	+-0.00064	
1.10440	+-0.00052	
1.25548	+-0.00055	
1.16563	+-0.00054	
1.07145	+-0.00047	
1.26865	+-0.00054	
1.19097	+-0.00052	
1.11411	+-0.00048	
1	0.001736	+-0.000053
2	0.059918	+-0.000977
3	0.439771	+-0.002683
4	0.352131	+-0.002571
5	0.081420	+-0.000708
6	0.030023	+-0.000333
7	0.021338	+-0.000316
8	0.011952	+-0.000261
9	0.001711	+-0.000053
1	0.000867	+-0.000078
2	0.018347	+-0.001405
3	0.098094	+-0.004532
4	0.273340	+-0.001093
5	0.216196	+-0.002184
6	0.134805	+-0.001510
7	0.134805	+-0.001402
8	0.101570	+-0.001077
9	0.021976	+-0.000259
1	0.000378	+-0.000057
2	0.006570	+-0.000899
3	0.030370	+-0.003110
4	0.159337	+-0.002199

5	0.216308	+-0.001493
6	0.172753	+-0.001520
7	0.202157	+-0.001880
8	0.170731	+-0.001656
9	0.041396	+-0.000418
1	0.000405	+-0.000075
2	0.005026	+-0.000869
3	0.014885	+-0.002483
4	0.082932	+-0.001941
5	0.181150	+-0.001014
6	0.181351	+-0.001215
7	0.240029	+-0.001824
8	0.230098	+-0.001887
9	0.064125	+-0.000558
1	0.002795	+-0.000084
2	0.060533	+-0.001065
3	0.399876	+-0.002279
4	0.376313	+-0.002220
5	0.088789	+-0.000977
6	0.032026	+-0.000445
7	0.023072	+-0.000341
8	0.013446	+-0.000250
9	0.003150	+-0.000080

```
****************************
* Our analysis environment *
****************************
```

Institute and Country : Institute of Nuclear Safety, NUPEC, Japan
Participants : Susumu Mitake (INS) and Osamu Sato (MRI)

Neutron data library :
(1) MGCL-JINS library
Neutron cross section : ENDF/B-IV (JENDL-3.2 for FP nuclides)
No. of energy groups : 137
Resonance self-shielding correction : Bondarenko method
Effective cross sections were prepared with MAIL-JINS code.
Heterogenity effect correction was made by Dancoff factor.

Code for k-effective calculation : KENO-V.a

Geometry modelings :
Fuel pellets, clads and moderators were modeled individually base on their configurations and dimensions given in the problem specification. No smeared technique was applied.
Length of channel box was assumed to be equal to the length of fuel assembly (from bottom end plug to top end plug).

Monte Carlo parameters (for all cases) :

No. of generations	:	403
No. of initially skipped generations	:	3
No. of histories / geneation	:	2400
Total No. of histories	:	960,000

These parameters were determined based on the results of the Phase IIB problem where we had investigated the statistical errors of the axial fission distributions varying the number of generations and the number of histories/ generation.

<Participant M>

1996/4/12
Institute of Nuclear Safety, NUPEC
Susumu Mitake

```
kyf01415@niftyserve.or.jp
SCALE 4.2 (with 27-group burnup library)
1.39021 +-0.00052
1.18758 +-0.00052
1.10467 +-0.00057
1.02094 +-0.00069
1.17419 +-0.00053
1.08532 +-0.00060
0.99518 +-0.00068
0.98586 +-0.00048
1.17991 +-0.00048
1.06579 +-0.00046
0.95600 +-0.00042
1.18903 +-0.00048
1.09075 +-0.00046
0.99477 +-0.00043
1.09412 +-0.00058
1.09848 +-0.00052
1.24802 +-0.00050
1.15917 +-0.00050
1.06854 +-0.00044
1.26059 +-0.00050
1.18524 +-0.00049
1.10964 +-0.00046
1       0.001634        +-0.000049
2       0.063875        +-0.000996
3       0.453595        +-0.001814
4       0.343829        +-0.001994
5       0.075887        +-0.000668
6       0.027585        +-0.000323
7       0.020249        +-0.000310
8       0.011602        +-0.000229
9       0.001744        +-0.000047
1       0.000764        +-0.000075
2       0.017237        +-0.001227
3       0.105430        +-0.003996
4       0.281033        +-0.001096
5       0.214352        +-0.002015
6       0.130979        +-0.001349
7       0.131319        +-0.001418
8       0.097766        +-0.001056
9       0.021120        +-0.000253
1       0.000434        +-0.000074
2       0.006908        +-0.001082
3       0.023901        +-0.002610
4       0.149079        +-0.001774
5       0.216340        +-0.001428
6       0.177826        +-0.001369
7       0.208416        +-0.001667
8       0.174694        +-0.001485
9       0.042402        +-0.000416
1       0.000347        +-0.000067
2       0.004950        +-0.000917
3       0.014441        +-0.002360
4       0.085850        +-0.001760
5       0.181387        +-0.000889
6       0.182191        +-0.001239
7       0.240878        +-0.001734
8       0.227412        +-0.001797
9       0.062546        +-0.000557
1       0.003369        +-0.000084
2       0.072347        +-0.000846
3       0.417578        +-0.002130
```

4	0.344899	+-0.001621
5	0.090080	+-0.000829
6	0.032510	+-0.000367
7	0.022990	+-0.000324
8	0.013118	+-0.000235
9	0.003109	+-0.000075

****
* Our analysis environment *
****

Institute and Country : Institute of Nuclear Safety, NUPEC, Japan
Participants : Susumu Mitake (INS) and Osamu Sato (MRI)

Neutron data library :
(2) SCALE 27G library
Neutron cross section : ENDF/B-IV
No. of energy groups : 27
Resonance self-shielding correction : Nordheim integral method
Effective cross sections were calculated with the correction of heterogenity effect for each axial fuel region, and CSASI module of SCALE 4.2 was used to calculate the macroscopic effective cross sections at each region.
These cross sections are combined using WAX code.

Code for k-effective calculation : KENO-V.a

Geometry modelings :
Fuel pellets, clads and moderators were modeled individually base on their configurations and dimensions given in the problem specification. No smeared technique was applied.
Length of channel box was assumed to be equal to the length of fuel assembly (from bottom end plug to top end plug).

Monte Carlo parameters (for all cases) :

No. of generations	:	403
No. of initially skipped generations	:	3
No. of histories / geneation	:	2400
Total No. of histories	:	960,000

These parameters were determined based on the results of the Phase IIB problem where we had investigated the statistical errors of the axial fission distributions varying the number of generations and the number of histories/ generation.

<Participant N>

APRIL 16 , 1996
INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE - CEA - FRANCE
ALI NOURI, GILLES POULLOT
ali.nouri@cea.fr FAX : 33 1 46 57 29 98
APOLLO-1 + MORET-3
1.4095
1.1985
1.1140
1.0309
1.1845
1.0950
0.9998
0.9927
1.1901
1.0751
0.9621
1.2021

1.0992
1.0077
1.1066
1.1079
1.2625
1.1718
1.0760
1.2738
1.1958
1.1231
1.238261E-03
6.526599E-02
5.665333E-01
3.132938E-01
4.059759E-02
8.627262E-03
3.339862E-03
1.012799E-03
9.132196E-05
9.361174E-05
2.944693E-03
4.494577E-02
3.579679E-01
2.391157E-01
1.312794E-01
1.211599E-01
8.521333E-02
1.727973E-02
2.073245E-06
7.669329E-05
2.785103E-03
1.331775E-01
2.301518E-01
1.887055E-01
2.180637E-01
1.851791E-01
4.185846E-02
0.000000E+01
1.649764E-05
5.356648E-04
7.738470E-02
1.854001E-01
1.865548E-01
2.501257E-01
2.532525E-01
4.673037E-02
8.136290E-04
3.509358E-02
4.638843E-01
4.237985E-01
5.695737E-02
1.195239E-02
5.293620E-03
1.886961E-03
3.197914E-04
INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE - CEA - FRANCE
ALI NOURI, GILLES POULLOT
Calculations have been carried out using APOLLO-1 and MORET-3. APOLLO-1 is an assembly code using the multigroup treatment (99 groups + self-shielding formalism) and the Collision Probability method for the flux and the leakage calculation in the fondamental mode situation. The nuclear data used by APOLLO-1 (the CEA86 library based on JEF-1 and ENDF/B) was processed using the NJOY-THEMIS system. APOLLO-1 uses the calculated flux for homogenisation and cross sections collapsing. The collapsed energy structure is the 16 groups Hansen and Roach. These

cross section are then used by MORET-3 : a 3 D Monte-Carlo code. For stainless stell nuclear data, the Hansen and Roach library was directly used in MORET.
No nuclide has been omitted
1 standard deviation , sigma = 0.00050
*.....
other information
cases 1- 5- 6- 7- 14
number of neutrons : 1000 batches of 500 neutrons
1 sigma = 0.00040
other cases
number of neutrons 600 batches of 500 neutrons
1 sigma = 0.00050

<Participant O>

April 24, 1996
BELGONUCLEAIRE S.A .
MALDAGUE Th.
Phone 32 2 77 40 511 Fax 32 2 774 05 47
Computer codes: WIMS-6 + KENO Va, Library WIMS 95 69groups
Multiplication factors:

Case 1	1.41373
Case 2	1.20019
Case 3	1.11591
Case 4	1.02852
Case 5	1.18760
Case 6	1.09454
Case 7	1.00125
Case 8	0.98865
Case 9	1.19102
Case 10	1.07336
Case 11	0.95804
Case 12	1.20099
Case 13	1.09698
Case 14	0.99829
Case 15	1.10466
Case 16	1.10829
Case 17	1.26384
Case 18	1.17069
Case 19	1.07462
Case 20	1.27627
Case 21	1.19485
Case 22	1.11678

Fission densities by region(%)

Case 1	Region 1	0.164
	Region 2	5.337
	Region 3	44.484
	Region 4	36.173
	Region 5	7.964
	Region 6	2.727
	Region 7	1.871
	Region 8	1.094
	Region 9	0.186
Case 5	Region 1	0.036
	Region 2	0.894
	Region 3	6.221
	Region 4	25.740
	Region 5	23.384
	Region 6	14.913
	Region 7	15.125

	Region 8	11.384
	Region 9	2.303
Case 6	Region 1	0.028
	Region 2	0.430
	Region 3	1.211
	Region 4	13.430
	Region 5	21.655
	Region 6	18.391
	Region 7	21.858
	Region 8	18.674
	Region 9	4.323
Case 7	Region 1	0.013
	Region 2	0.193
	Region 3	0.563
	Region 4	7.414
	Region 5	17.957
	Region 6	18.476
	Region 7	24.674
	Region 8	24.289
	Region 9	6.422
Case14	Region 1	0.349
	Region 2	8.002
	Region 3	49.804
	Region 4	31.891
	Region 5	5.780
	Region 6	1.915
	Region 7	1.331
	Region 8	0.766
	Region 9	0.163

Analysis Environnement:
The code WIMS-E (Version 6 - 1994) from AEA Technology was used to calculate self-shielding and neutoron flux spectrum in 2D geometry for each density set and in 69 energy groups (CACTUS module).
The cross-sections were condensed to 16 energy groups and fuel pin cell smeared.
The library used is WIMS 95 with 69 energy groups, this library is based on JEF 2.2 file.
The 16 - groups cross-sections sets have been input for KENO Va calculations in 3D.
The Keno multiplication factors are calculated with a standard deviation of 0.1%.
The flux deviation is +/- 5% region with large fission densities and +/- 15% in regions with low fission densities (end zones).

<Participant P>

May 9, 1996
Paul Scherrer Institute, CH-5232 Villigen PSI, Switzerland
Peter Grimm
E-mail: Peter.Grimm@psi.ch Fax: +41 56 310 21 99
BOXER
1.4058 k-eff, Cases 1 - 22
1.1991
1.1161
1.0322
1.1867
1.0957
1.0042
0.9936
1.1887
1.0744
0.9622
1.1997
1.0985
1.0021

```
1.1062
1.1089
1.2600
1.1689
1.0755
1.2728
1.1946
1.1178
1.5335E-03                    Fractional fission densities, Case 1
5.7699E-02
4.4081E-01
3.6178E-01
7.8971E-02
2.7804E-02
1.9283E-02
1.0585E-02
1.5329E-03
5.4644E-05                    Fractional fission densities, Case 5
1.5507E-03
3.7660E-02
2.6538E-01
2.4419E-01
1.5613E-01
1.5557E-01
1.1570E-01
2.3762E-02
1.4984E-06                    Fractional fission densities, Case 6
4.3953E-05
5.0259E-03
1.3142E-01
2.2413E-01
1.8790E-01
2.2123E-01
1.8698E-01
4.3272E-02
1.1127E-07                    Fractional fission densities, Case 7
3.2518E-06
1.1881E-03
7.2104E-02
1.7943E-01
1.8685E-01
2.5184E-01
2.4380E-01
6.4785E-02
2.4104E-03                    Fractional fission densities, Case 14
5.9061E-02
4.3862E-01
3.5946E-01
7.9038E-02
2.8095E-02
1.9763E-02
1.1144E-02
2.4068E-03
```

PSI Calculations for the Burnup Credit Criticality Benchmark, Phase III-A

Peter Grimm

Paul Scherrer Institute
CH-5232 Villigen PSI
Switzerland

Neutron Data Library:

The BOXLIB cross section library for BOXER used in the present calculations was processed using the ETOBOX code developed at PSI (Ref. 1). It contains cross sections for 34 actinide nuclides (from Th-232 through Cm-248), 55 fission products considered explicitely, and two pseudo fission products. The source of cross section data for all nuclides is JEF-1, except for Gd-155, whose cross sections are taken from JENDL-2, and for Zircalloy-2, which is taken from ENDF/B-4. The fission product yields are taken from JEF-2 for thermal fission and summed over the isobaric chains. The decay data for the fission products originate from the compilation in Reference 2. For the actinides half lives from Ref. 3 are used.

The BOXLIB cross section library contains microscopic neutron cross sections collapsed to 70 groups. The group structure is the 69 group WIMS structure with an extra group between 10 and 15 MeV. However, the upper boundary of the thermal energy range is 1.3 eV instead of 4 eV. P0 and P1 scattering matrices (P2 transport corrected) are given for most isotopes. The (n,2n) and (n,3n) reactions are included in the scattering matrices (no P2 transport correction) and subtracted from the absorption cross section. The weighting spectrum used for the collapsing in ETOBOX is a spectrum calculated in many microgroups for a typical LWR cell in the fast range, a 1/E spectrum at intermediate energies, and a modified Maxwellian spectrum in the thermal range. In the fast range (E > 907 eV) the resonance cross sections (both resolved and unresolved resonances) are Doppler broadened and collapsed to groups for three temperatures and 4 values of the dilution cross section. In the resonance range between 1.3 eV and 907 eV (important low energy resonances of plutonium isotopes are included) pointwise lists of Doppler broadened cross sections are produced for three to seven temperatures (depending on the nuclide). For the unresolved resonances these lists are produced for four dilutions. The spacing of the points depends on the variation of the cross sections with lethargy, so that the cross section values between the points can be accurately reconstructed by interpolation. The minimum spacing of the points is 1.0E-5 lethargy units. Typical numbers of energy points for actinides are 7000 to 8000 between 1.3 and 907 eV. For some fission products which contribute little to the absorption the resonance cross sections are given for infinite dilution only. The thermal scattering matrices for most nuclides are calculated using the free gas model. For the moderator nuclides and especially for hydrogen in water the S(alpha,beta) matrices given in the basic cross section files are used.

Cell and Transport Code:

BOXER cell and two-dimensional transport and depletion code (developed at PSI, Ref. 4)

Resonance self-shielding: Pointwise two-region collision probability calculation (1.3 eV < E < 907 eV), tabular interpolation versus temperature and equivalent dilution cross section for E > 907 eV, Dancoff factor corrected for 2D array geometry by Monte Carlo method

Cell calculation: One-dimensional integral transport calculation in cylindrical geometry employing white boundary conditions or boundary source from a previously calculated cell. Fundamental mode spectrum (k-eff = 1) in 70 groups
by B1 method for homogenized cell

Weighting spectrum for group collapsing in homogeneous (non-cell) materials:
One-dimensional transport calculation in slab geometry, preserving mean chord lengths of materials in two-dimensional grid

Two-dimensional transport calculations: Transmission probability integral transport method, using first-order spherical harmonics expansions for mesh surface currents and linear space dependence of surface currents and source within meshes, P1 anisotropic scattering

Models and Calculational Options Used:

Assembly averaged cross sections for each composition from 2D BOXER calculation with homogenized cells (fuel pin and water rod)

Reflector materials smeared in radial direction

Cladding composition replaced by Zircalloy-2

Axial calculation: One-dimensional fine mesh BOXER transport calculation using assembly averaged macroscopic cross sections, 1 to 1.5 cm mesh widths (12 meshes per axial node, total 352 meshes)

Energy group structure for assembly homogenization and axial calculation:
11 groups, upper boundaries 15, 6.07, 2.23 MeV, 821 keV, 907, 75.5, 16, 4, 1.3, 0.625, and 0.14 eV

Convergence criterion: 1.0E-5 for flux in axial calculation

References
---

1. J.M. Paratte, K. Foskolos, P. Grimm, J.M. Hollard,
"ELCOS, The PSI Code System for LWR Core Analysis,
Part I: User's Manual for the Library Preparation Code ETOBOX,"
PSI Report 96-02, January 1996

2. M.E. Meek, B.F. Rider,
"Compilation of Fission Product Yields",
NEDO-12154-1, 74NED6 (1974)

3. W. Seelmann-Eggebert et al.,
"Karlsruhe Chart of the Nuclides, 5th edition",
Kernforschungszentrum Karlsruhe (1981)

4. J.M. Paratte, P. Grimm, J.M. Hollard,
"ELCOS, The PSI Code System for LWR Core Analysis,
Part II: User's Manual for the Fuel Assembly Code BOXER,"
PSI Report 96-08, February 1996

<Participant Q>

May 22, 1996 (July 24, 1996)
Oak Ridge National Laboratory (USA)
Mark D. DeHart, Cecil V. Parks
DeHartMD@ornl.gov, ParksCV@ornl.gov (Fax: 423-576-3513)
SCALE 4.3 (BONAMI/NITAWL-II and KENO V.a)
1.4038 +/- 0.0005
1.1982 +/- 0.0005
1.1149 +/- 0.0005
1.0294 +/- 0.0005
1.1852 +/- 0.0005
1.0949 +/- 0.0005
1.0034 +/- 0.0005

0.9918 +/- 0.0004
1.1885 +/- 0.0005
1.0740 +/- 0.0005
0.9617 +/- 0.0004
1.1997 +/- 0.0005
1.0983 +/- 0.0005
1.0008 +/- 0.0004
1.1045 +/- 0.0005
1.1069 +/- 0.0005
1.2590 +/- 0.0005
1.1676 +/- 0.0005
1.0734 +/- 0.0004
1.2711 +/- 0.0005
1.1932 +/- 0.0005
1.1168 +/- 0.0005
0.0015 +/- 3.03%
0.0609 +/- 1.73%
0.4637 +/- 0.46%
0.3484 +/- 0.66%
0.0722 +/- 0.90%
0.0247 +/- 1.21%
0.0173 +/- 1.49%
0.0099 +/- 1.72%
0.0014 +/- 2.53%
0.0001 +/- 9.87%
0.0029 +/- 5.39%
0.0501 +/- 1.62%
0.2867 +/- 0.39%
0.2374 +/- 0.32%
0.1466 +/- 0.40%
0.1450 +/- 0.42%
0.1076 +/- 0.50%
0.0234 +/- 0.79%
0.0000 +/- 0.00%
0.0000 +/- 97.83%
0.0047 +/- 2.92%
0.1352 +/- 0.64%
0.2275 +/- 0.31%
0.1879 +/- 0.29%
0.2186 +/- 0.29%
0.1821 +/- 0.33%
0.0441 +/- 0.53%
0.0000 +/- 0.00%
0.0000 +/- 0.00%
0.0010 +/- 6.17%
0.0729 +/- 0.95%
0.1803 +/- 0.33%
0.1866 +/- 0.28%
0.2514 +/- 0.24%
0.2399 +/- 0.27%
0.0679 +/- 0.44%
0.0027 +/- 2.62%
0.0617 +/- 1.58%
0.4532 +/- 0.62%
0.3414 +/- 0.65%
0.0789 +/- 1.36%
0.0290 +/- 1.55%
0.0200 +/- 1.79%
0.0106 +/- 2.13%
0.0024 +/- 2.75%
Institute and country: Oak Ridge National Laboratory, USA
Participants: Mark D. DeHart and Cecil V. Parks
Neutron data library: SCALE 44GROUPNDF5, a 44-group neutron

cross-section library based on ENDF/B-V data, with ENDF/B-VI evaluations for O-16, Eu-154 & Eu-155. [Ref. 1]
Neutron data processing code or method: All cross-sections were processed using the CSASN sequence of SCALE 4.3. CSASN uses the BONAMI code to apply the Bondarenko resonance self-shielding method for nuclides with Bondarenko data; it then invokes NITAWL-II to perform Nordheim resonance self-shielding corrections. CSASN calculations are performed assuming an infinite lattice of pins; nominal pin parameters were assumed.
Neutron energy groups: 44 energy groups, selected based on typical LWR spectra.
Description of code system: SCALE 4.3 [Ref. 2], run on an IBM RS/6000-580 workstation. CSASN was used for cross-section procession; the WAX utility module was used to combine cross-section libraries computed for each of 9 axial zones into a single library; k-effective calculations were performed using this library and KENO-V.a
Geometry Modelling: Gas plenum was explicitly modeled as a void region inside the clad and bounded by the blanket fuel and the end plug. Channel box was assumed to extend the full length of reflector (i.e., 459.76cm). Reflective BCs were assumed axially at either end of reflector.
Omitted or replace nuclides: None.
Statistical errors for k-eff: As reported in lines 6-27 above, ranging from +/-0.0004 to +/-0.0005, for a 1-sigma confidence level.
Statistical errors for fission densities: given in lines 28-72 as a percentage of reported value, for a 1-sigma confidence level.
Other information: Calculations were based on a total of 2000 generations of 1000 neutrons each. The first 1000 generations were discarded for the purpose of calculating the fission densities. Fission density profiles are thus considered 'converged'; however, these profiles will show zero fission densities in regions distant from the dominant source region.
References:

1) M. D. DeHart and S. M. Bowman, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross-Section Library for Use in Criticality Safety Analyses," NUREG/CR-6102 (ORNL/TM-12460), Oak Ridge Natl. Lab., Martin Marietta Energy Systems, Inc., 1994.
2) "SCALE- A Modular Code System for Performing Standardize Computer Analyses for Licensing Evaluation," NUREG/CR-200, Rev 5 (ORNL/NUREG/CSD-2/R5), Vols. I, II, and III (Draft, Sept. 1995).

<Participant R>

Case Identifier	k-effective RUN1	
1	1.4056	(0.0007)
2	1.1932	(0.0007)
3	1.1091	(0.0007)
4	1.0269	(0.0008)
5	1.1778	(0.0007)
6	1.0900	(0.0008)
7	0.9994	(0.0008)
8	0.9897	(0.0006)
9	1.1844	(0.0006)
10	1.0699	(0.0006)
11	0.9586	(0.0006)
12	1.1955	(0.0007)

13	1.0948	(0.0006)
14	1.0002	(0.0006)
15	1.0997	(0.0008)
16	1.1091	(0.0007)
17	1.2569	(0.0007)
18	1.1664	(0.0007)
19	1.0757	(0.0006)
20	1.2699	(0.0007)
21	1.1923	(0.0007)
22	1.1176	(0.0007)

Case Identifier	k-effective RUN2	
1	1.4058	(0.0007)
2	1.1915	(0.0007)
3	1.1085	(0.0007)
4	1.0280	(0.0007)
5	1.1787	(0.0007)
6	1.0877	(0.0008)
7	0.9987	(0.0008)
8	0.9894	(0.0006)
9	1.1841	(0.0007)
10	1.0706	(0.0006)
11	0.9573	(0.0006)
12	1.1962	(0.0007)
13	1.0945	(0.0006)
14	0.9985	(0.0006)
15	1.1007	(0.0008)
16	1.1069	(0.0007)
17	1.2585	(0.0007)
18	1.1667	(0.0007)
19	1.0757	(0.0006)
20	1.2703	(0.0007)
21	1.1931	(0.0007)
22	1.1167	(0.0007)

Case Identifier	Mean k-effective	
1	1.4057	(0.0005)
2	1.1924	(0.0005)
3	1.1088	(0.0005)
4	1.0275	(0.0005)
5	1.1783	(0.0005)
6	1.0889	(0.0006)
7	0.9991	(0.0006)
8	0.9896	(0.0004)
9	1.1843	(0.0005)
10	1.0703	(0.0004)
11	0.9580	(0.0004)
12	1.1959	(0.0005)
13	1.0947	(0.0004)
14	0.9994	(0.0004)
15	1.1002	(0.0006)
16	1.1080	(0.0005)
17	1.2577	(0.0005)
18	1.1666	(0.0005)
19	1.0757	(0.0004)
20	1.2701	(0.0005)
21	1.1927	(0.0005)
22	1.1172	(0.0005)

fractional fission densities for Phase IIIA of the BUC Benchmark Exercise:

Case 1

Region	FFD
1	0.002
2	0.064
3	0.432
4	0.355
5	0.085
6	0.029
7	0.020
8	0.011
9	0.002

Case 5

Region	FFD
1	0.001
2	0.034
3	0.131
4	0.261
5	0.199
6	0.128
7	0.129
8	0.097
9	0.020

Case 6

Region	FFD
1	0.001
2	0.010
3	0.025
4	0.151
5	0.216
6	0.178
7	0.208
8	0.172
9	0.040

Case 7

Region	FFD
1	0.000
2	0.005
3	0.011
4	0.084
5	0.186
6	0.185
7	0.240
8	0.228
9	0.060

Case 14

Region	FFD
1	0.003
2	0.066
3	0.467
4	0.335
5	0.069
6	0.026
7	0.019
8	0.013
9	0.003

<Participant S>

May 13, 1999
Argonne National Laboratory
Roger Blomquist
RNBlomquist@anl.gov
VIM Continuous Energy Monte Carlo Code
1.4032 (0.0008)
1.1957 (0.0007)
1.1132 (0.0007)
1.0309 (0.0007)
1.1842 (0.0007)
1.0958 (0.0007)
0.9993 (0.0007)
0.9911 (0.0007)
1.1852 (0.0007)
1.0714 (0.0007)
0.9604 (0.0007)
1.1978 (0.0007)
1.0956 (0.0007)
1.0011 (0.0007)
1.1021 (0.0007)
1.1079 (0.0007)
1.2580 (0.0009)
1.1669 (0.0007)
1.0746 (0.0007)
1.2708 (0.0007)
1.1935 (0.0008)
1.1153 (0.0007)
0.0017 (0.0002)
0.0602 (0.0054)
0.4307 (0.0142)
0.3578 (0.0092)
0.0826 (0.0056)
0.0307 (0.0026)
0.0219 (0.0024)
0.0125 (0.0016)
0.0018 (0.0002)
0.0002 (0.0001)
0.0030 (0.0010)
0.0407 (0.0044)
0.2667 (0.0090)
0.2464 (0.0046)
0.1577 (0.0035)
0.1522 (0.0033)
0.1108 (0.0034)
0.0224 (0.0007)
0.0000 (0.0000)
0.0000 (0.0000)
0.0027 (0.0005)
0.1271 (0.0052)
0.2295 (0.0023)
0.1834 (0.0024)
0.2215 (0.0025)
0.1912 (0.0023)
0.0446 (0.0007)
0.0000 (0.0000)
0.0000 (0.0000)
0.0019 (0.0005)
0.0816 (0.0032)
0.1846 (0.0029)
0.1861 (0.0023)

0.2471 (0.0027)
0.2353 (0.0034)
0.0633 (0.0010)
0.0018 (0.0002)
0.0437 (0.0030)
0.4357 (0.0138)
0.3873 (0.0108)
0.0746 (0.0043)
0.0262 (0.0020)
0.0186 (0.0014)
0.0100 (0.0008)
0.0022 (0.0002)
Institute and country: Argonne National Laboratory, USA
Participant: Roger N. Blomquist
Neutron data library: VIM continuous energy data files based on
(1) ENDF/B-V for actinides, structurals, and moderator, and
(2) ENDF/B-VI for fission products.
Neutron data processing code or method: Pointwise energy treatment of smooth and resolved resonance data, probability tables for unresolved resonance cross sections with conditional means for scattering and fission, probability tables or analytical distributions for secondary energy and angle distributions. All codes are entirely independent of NJOY.
Neutron energy groups: Continuous energy, with linear-linear interpolation errors less than 0.005.
Description of code system: VIM (Ref. 1) continuous-energy Monte Carlo neutron transport code (ENDF/B-V available at RSICC and OECD/NEA Data Bank) run on Sun Sparcstations.
* power iteration on fission source distribution
* no variance reduction used, except optimal combined eigenvalue estimator
Geometry modelling: Gas plenum was smeared, all other geometry represented in detail with combinatorial geometry.
Omitted or replaced nuclides: none, other than as prescribed in specifications, paragraph 2.
Estimated absolute 1-sigma statistical k-eff uncertainties: see lines 6-27.
Estimated absolute 1-sigma statistical fractional fission rate uncertainties: see lines 28-72.
Other information: 2000 histories per generation; skipped 1000 generations, then collected tallies for 500 generations. Reaction rate tallies accumulated for 20 each set of 20 generations to reduce the influence of serial correlation due to loose coupling.
References:
1. "VIM Continuous Energy Monte Carlo Transport Code", Proceedings of the International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses, April 30 - May 4, 1995, Portland, OR, p1133.

<Participant T>

01/07/97
"DETR, UK"
Jim Stewart
Jim@rmtd.demon.co.uk
MONK6
1.4159
1.1997
1.1151
1.0283
1.1896

1.0952
0.9987
0.9887
1.1904
1.0732
0.9589
1.1996
1.0964
0.9993
1.1068
1.1087
1.2641
1.1713
1.076
1.2759
1.1968
1.1163
0.0014
0.0516
0.4841
0.3458
0.0701
0.0219
0.0155
0.0085
0.001
0.0002
0.0015
0.0319
0.2353
0.2542
0.1637
0.1648
0.1234
0.0249
0.0000
0.0000
0.0062
0.1442
0.2241
0.1836
0.2148
0.1853
0.0418
0.0123
0.1789
0.0499
0.0628
0.1370
0.1413
0.1885
0.1821
0.0472
0.0029
0.0794
0.5345
0.3022
0.0504
0.0150
0.0097
0.0046
0.0013
"Results by Jim Stewart Department of the Environment Transport and the Regions, UK"

UKNDL library (with standard data adjustments) with some JEFF 2 extensions
8220 groups (pseudo-point)
MONK6B run on SCO UNIX PC
Top and bottom plugs of water rod modelled as water rather than zircalloy
All nuclides included
Converged to standard deviation of 0.0005-0.001
Typical standard deviations on fission densities 1% to 4% (some results up to 15%)

<Participant U>

01/07/97
"DETR, UK"
Jim Stewart
Jim@rmtd.demon.co.uk
MONK6
1.3919
1.1901
1.1067
1.0247
1.1751
1.0869
0.9936
0.9842
1.1799
1.0662
0.9521
1.1903
1.0885
0.9912
1.0999
1.1024
1.251
1.1623
1.0698
1.2644
1.189
1.1126
0.0018
0.0612
0.4242
0.3683
0.0803
0.0305
0.0207
0.0116
0.0016
0.0007
0.0129
0.0636
0.2606
0.2231
0.1447
0.1521
0.1184
0.0239
0.0000
0.0002
0.0051
0.1233
0.2209
0.1907

```
0.2255
0.1912
0.0430
0.0005
0.0134
0.0043
0.0802
0.1755
0.1822
0.2450
0.2348
0.0639
0.0035
0.0827
0.5076
0.3140
0.0587
0.0166
0.0098
0.0057
0.0014
"Results by Jim Stewart Department of the Environment Transport and the Regions, UK"
UKNDL library (with no data adjustments) with some JEFF 2 extensions
8220 groups (pseudo-point)
MONK6B run on SCO UNIX PC
Top and bottom plugs of water rod modelled as water rather than zircalloy
All nuclides included
Converged to standard deviation of 0.0005-0.001
Typical standard deviations on fission densities 1% to 4% (some results up to 10%)
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Appendix III

A Proposal for Phase IIIA Benchmarks Made at 1995 Meeting in Albuquerque

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Burnup Credit Criticality in BWRs

- Study on the Proposed Benchmark Problem -

1. Introduction
2. Seclection of Typical BWR Fuel Assembly
3. Modeling of Fuel Assembly
4. Operating History in BWRs
5. Necessary Data Sets of Fuel Composition
6. Method in Calculating the Fuel Compostions

September 1995

Yoshitaka Naito (JAERI)
Yoshihira Ando (Toshiba NEL)
Ishi Mitsuhashi (Toshiba NEL)

1. Introduction

We proposed a burnup credit benchmark problem for BWR spent fuel at the last meeting held in Paris and got several comments for our problem. So, we modify our problem and propose a modified benchmark problem.

Modified items are two items as follows.

(1) estimate fuel composition based on the burnup calculation for typical BWR fuel assembly

(2) consider blanket regions at each side of the fuel

2. Selection of Typical BWR Fuel Assembly

We select a typical BWR fuel assembly shown in Fig.1. This type of fuel assembly has been used widely in the USA and Japanese commercial BWR plants. The fuel enrichment distribution shown in Fig.1 is a typical example (called as reference assembly hereafter) in which fuel rod average enrichment is 3.80 w/o. This enrichment is set up based on the fuel design to achieve a batch average exposure of about 40 GWd/Mt. A fuel assembly in blanket region is shown in Fig.2. All fuel rods in this fuel assembly contain natural uranium and the assembly has a large center water rod same as the typical fuel assembly shown in Fig.1.

3. Modeling of Fuel Assembly

Our proposed benchmark is the infinite fuel assembly array in water. The depletion behavior of fuel composition in each rod is different because of the heterogeneity of fuel enrichment distribution and neutron spectrum distribution in BWR fuel assembly. If we calculate the real geometry of spent fuel, the large number data sets of fuel composition corresponding to the number of fuel rods in the reference assembly will be needed. The number is too large, so we think that the homogenization of the fuel composition after the depletion in the reference assembly is necessary. We show the homogenization of fuel composition in Fig.3. We call the right assembly in Fig.3 as the homogeneous assembly hereafter.

We discuss on the two effects arisen from the homogenization of fuel composition.

(1) Gd reactivity worth

Gd reactivity worth shown in Fig.4 is the reactivity difference of the infinite multiplication factors in cold state after the burnup between the reference assembly and the fuel assembly excluding Gd isotope from the reference assembly (called as the heterogeneous assembly hereafter).

Gd reactivity worth decreases with the fuel burnup and becomes constant near about 2 % dK/K after about 20 GWd/Mt. So, we concluded that the exclusion of Gd isotope from the fuel brings no problem in the burnup credit benchmark for BWR spent fuel.

(2) Reactivity difference due to the homogenization of fuel composition

To estimate the reactivity effect arisen from the homogenization of fuel composition, we compare the infinite multiplication factors between the heterogeneous assembly and the homogeneous assembly. The reactivity difference due to the homogenization of fuel composition shown in Fig.5 is almost constant near about 2 % dK/K after 20 GWd/Mt and vary little with the void fraction and the cooling period. Therefore, we concluded that the effect due to the homogenization of fuel composition is negligible in the estimation of burnup credit effect for

4. Operating History in BWRs

To calculate the fuel composition of spent fuel, void profile and burnup profile in BWRs are needed. So, we calculated these operating histories on the assumption of Hailing distribution for 4 batch core in which average enrichment is 3.80 w/o in fuel region and a node fuel with natural uranium is set up as the blanket in each side of the fuel. Void profile is shown in Fig.6, and burnup profile is shown in Fig.7.

5. Necessary Data Sets of Fuel Composition

We studied on the necessary division of fuel composition along the axis.

We compared the multiplication factors and axial power distributions in studied cases shown in Fig.8, in which axial assembly model was determined referred on BWR axial assembly model of ORIGEN2 cross section library shown in Fig.9.

First, we calculated the macro cross sections for fuel assembly lattice shown in Fig.1 and Fig.2 through burnup calculation and those for external regions ,i.e gas plenum, end plug, tie plate, reflector through fixed source calculation using BWR lattice analysis code. Second, the calculated macro cross sections were tabulated as function of void fraction and burnup. Finally we calculated the multiplication factors and axial power distributions based on the macro cross sections corresponding to the void profile and burnup profile shown in Fig.8 and Fig.9 through 1D-diffusion calculations, in which the macro cross sections are homogenized through the heterogeneity in assembly.

In the comparison, 24 node calculation is the reference calculation because 24 node calculation is standard in BWR 3D simulation.

Comparison of relative power distributions in reference calculation is shown in Fig.10. This shows that power distribution shifts from the center to the top in the fuel as the burnup progress and the detailed division in the top is required to get an accurate result in the high burnup cases.

Comparison of reactivity difference in studied cases is shown in Fig.11 and results are shown in Table 1 and Table 2.

Above results on multiplication factors show that the CASE 9 (9 axial division) is the best averaging scheme of axial fuel composition and the difference between the reference and the CASE 9 is small enough to study on burnup credit effect for BWR spent fuel.

On the other side, the compared results on power distributions shown in Fig.12 and Table 3 also shows that the CASE 9 is the best.

Therefore, we concluded that we can calculate fuel composition data based on the 9 axial zoning corresponding to the CASE 9.

Our proposal benchmark cases and case-wise priority are shown in Table 4 and Table 5.

If we adopt the 9 axial zoning, necessary data sets of fuel composition for benchmark problems are as follows.

Very Important	: 50 data sets
Important	: 120 data sets
Rather Important	: 96 data sets
Little Important	: 138 data sets
not Important	: 66 data sets

6. Method in Calculating the Fuel Composition

We use SPINOZA (Spectrum Indued Nuclear OrganiZation Analysis) system in calculating the fuel composition.

SPINOZA system has been developed in Toshiba. In this system, we can execute ORIGEN2 calculation with cross section corresponding to neutron spectrum in BWR lattice.The schematic flow diagram of SPINOZA system is shown in Fig.13. SPINOZA system consists of four modules TGBLA, OLIB, TLIB, ORIGEN2. TGBLA is BWR fuel design code and generates FGS file that includes fine group neutron spectrum for every region in BWR fuel assembly. OLIB calculates one group cross section for (n,gamma), (n,fiss), (n,2n), (n,3n), (n,alpha), (n,p) in combination with fine group neutron spectrum and fine group cross section and installs them in public ORIGEN2 cross section library, where user can also select neutron spectrum in specified region : fuel / cladding / moderator in a cell , channel, out-channel moderator, control blade. User can optionally update fission yield data in ORIGEN2 cross section library through OLIB. But, cross sections in SPINOZA cross section library are created in infinite dilution condition. Therefore, TLIB calculates one group cross sections for (n,gamma), (n,fiss) in combination with burnup dependent self-shielded fine group cross sections and a burnup dependent FGS file.

Self-shielded fine group cross sections are prepared through TGBLA burnup calculation in which important resonant nuclides are treated explicitly. ORIGEN2 has burnup dependent cross section as block data to consider that cross sections for important resonant nuclides change in burnup. So, TLIB makes up burnup dependent cross sections for all reactions and nuclides included in ORIGEN2 burnup dependent cross section data. Therefore, user can execute ORIGEN2 calculation with the cross sections which corresponds to neutron spectrum in the region specified by user and are prepared as a function of fissioned nuclide amount until the highest burnup in TGBLA burnup calculation.

First, fuel composition data will be estimated by ORIGEN2 using the assembly average one group cross section as a function of void fraction and burnup under several rated power conditions. Second, the fuel composition data will be arranged on the void and burnup history in combination with benchmark problem case and operating history shown in Fig.6 and Fig.7. Finally, fuel composition data in the benchmark problem will be made out from the above arranged fuel composition.

Table 1 Comparison of Multiplication Factors in Studied Casess at No Cooling Condition

STATE	REFERENCE	CASE-3	CASE-4	CASE-5	CASE-6A	CASE-6B	CASE-7	CASE-8A	CASE-8B	CASE-8C	CASE-9	CASE-10	CASE-12	CASE-14
BOL	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289	1.40289
EOC1	1.21672	1.21840	1.22091	1.22121	1.22015	1.21576	1.21588	1.21795	1.21559	1.21543	1.21616	1.21723	1.21657	1.21689
EOC2	1.12210	1.10178	1.11490	1.12220	1.12587	1.11289	1.11737	1.12433	1.11719	1.11737	1.12132	1.12219	1.12162	1.12176
EOC3	1.04966	1.00476	1.03005	1.04271	1.04910	1.03877	1.04545	1.05175	1.04425	1.04545	1.04883	1.04909	1.04908	1.04840
EOC4	1.00433	0.94790	0.97641	0.99170	1.00012	0.99357	1.00081	1.00563	0.99885	1.00081	1.00366	1.00337	1.00382	1.00240

Table 2 Comparison of Multiplication Factor Differences in Studied Casess at No Cooling Condition

STATE	*%DK/K CASE-3	CASE-4	CASE-5	CASE-6A	CASE-6B	CASE-7	CASE-8A	CASE-8B	CASE-8C	CASE-9	CASE-10	CASE-12	CASE-14
BOL	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
EOC1	0.14	0.34	0.37	0.28	-0.08	-0.07	0.10	-0.11	-0.11	-0.05	0.04	-0.01	0.01
EOC2	-1.81	-0.64	0.01	0.34	-0.82	-0.42	0.20	-0.44	-0.42	-0.07	0.01	-0.04	-0.03
EOC3	-4.28	-1.87	-0.66	-0.05	-1.04	-0.40	0.20	-0.52	-0.42	-0.08	-0.05	-0.05	-0.12
EOC4	-5.62	-2.78	-1.26	-0.42	-1.07	-0.35	0.13	-0.55	-0.35	-0.07	-0.10	-0.05	-0.19

Where, %DK/K(Case#) = 100 x {K-eff(Case#)/K-eff(Reference) - 1}

Table 3 Comparison of Power Distributions in EOC4 Core at No Cooling Condition

*Region-wise Relative Fission Density Distribution

NODE#	REFERENCE	CASE-3	CASE-5	CASE-6A	CASE-8A	CASE-9	CASE-14
1	3.6789E-06	6.6557E-02	2.6604E-05	8.4464E-06	3.8530E-06	3.6197E-06	4.4984E-06
2	1.7546E-05	2.7636E-01	1.1962E-04	3.8658E-05	1.7956E-05	1.6810E-05	2.1383E-05
3	2.9632E-05	4.7646E-01	2.6861E-04	8.7925E-05	3.9143E-05	3.6423E-05	3.5640E-05
4	5.6740E-05	6.7159E-01	5.4846E-04	1.8210E-04	7.6471E-05	7.0564E-05	7.1898E-05
5	1.1665E-04	8.5653E-01	1.0496E-03	3.5321E-04	1.3370E-04	1.2342E-04	1.4212E-04
6	2.6676E-04	1.0249E+00	2.0962E-03	7.1574E-04	2.8486E-04	2.5387E-04	3.1551E-04
7	5.9830E-04	1.1754E+00	4.1544E-03	1.4287E-03	6.1954E-04	5.3060E-04	7.1352E-04
8	1.3017E-03	1.3053E+00	8.2044E-03	2.8770E-03	1.3517E-03	1.1130E-03	1.5117E-03
9	2.7383E-03	1.4124E+00	1.6360E-02	5.7970E-03	3.0090E-03	2.3366E-03	3.2121E-03
10	5.5818E-03	1.4949E+00	3.0726E-02	1.1682E-02	6.1977E-03	4.9064E-03	6.4482E-03
11	1.1006E-02	1.5514E+00	5.6931E-02	2.3539E-02	1.2596E-02	1.0302E-02	1.2739E-02
12	2.2405E-02	1.5808E+00	1.0506E-01	4.7380E-02	2.5497E-02	2.1599E-02	2.5514E-02
13	4.5341E-02	1.5826E+00	1.9364E-01	9.7890E-02	5.2346E-02	4.6798E-02	5.1769E-02
14	9.0063E-02	1.5569E+00	3.5678E-01	1.8090E-01	1.0089E-01	8.8444E-02	1.0052E-01
15	1.7464E-01	1.5041E+00	6.5729E-01	3.2635E-01	1.9200E-01	1.6257E-01	1.9601E-01
16	3.2974E-01	1.4250E+00	1.2091E+00	5.8436E-01	3.6385E-01	2.9636E-01	3.6134E-01
17	6.0527E-01	1.3211E+00	2.4166E+00	1.0439E+00	7.1215E-01	5.3892E-01	6.6776E-01
18	1.0770E+00	1.1941E+00	3.2527E+00	1.8613E+00	1.2091E+00	9.7863E-01	1.1522E+00
19	1.8466E+00	1.0463E+00	3.7301E+00	3.6271E+00	1.9793E+00	1.8450E+00	2.0001E+00
20	3.0097E+00	8.8018E-01	3.7950E+00	4.5671E+00	3.1931E+00	2.9391E+00	3.0914E+00
21	4.5495E+00	6.9703E-01	3.3913E+00	4.5807E+00	5.4259E+00	4.5869E+00	4.7639E+00
22	5.7031E+00	5.0330E-01	2.6592E+00	3.8651E+00	5.6362E+00	5.8055E+00	5.3499E+00
23	5.1772E+00	3.0372E-01	1.6587E+00	2.5019E+00	4.0281E+00	5.2910E+00	4.9268E+00
24	1.3477E+00	9.3180E-02	4.5416E-01	6.6933E-01	1.0571E+00	1.3794E+00	1.2876E+00
TOTAL	1.0000E+00	1.0000E+00	1.0000E+00	1.0000E+00	1.0000E+00	1.0000E+00	1.0000E+00

*%DP/P

NODE#	CASE-3	CASE-5	CASE-6A	CASE-8A	CASE-9	CASE-14
1	1809055	623	130	4.7	-1.6	22.3
2	1574960	582	120	2.3	-4.2	21.9
3	1607824	806	197	32.1	22.9	20.3
4	1183527	867	221	34.8	2.4	26.7
5	734173	800	203	14.6	5.8	21.8
6	384103	686	168	6.8	-4.8	18.3
7	196357	594	139	3.6	-11.3	19.3
8	100177	530	121	3.8	-14.5	16.1
9	51479	497	112	9.9	-14.7	17.3
10	26682	450	109	11.0	-12.1	15.5
11	13996	417	114	14.4	-6.4	15.7
12	6956	369	111	13.8	-3.6	13.9
13	3390	327	116	15.4	3.2	14.2
14	1629	296	101	12.0	-1.8	11.6
15	761	276	87	9.9	-6.9	12.2
16	332	267	25	10.3	-10.1	9.6
17	118	299	72	17.7	-11.0	10.3
18	11	202	73	12.3	-9.1	7.0
19	-43	102	96	7.2	-0.1	8.3
20	-71	26	52	6.1	-2.3	2.7
21	-85	-25	1	19.3	0.8	4.7
22	-91	-53	-32	-1.2	1.8	-6.2
23	-94	-68	-52	-22.2	2.2	-4.8
24	-93	-66	-50	-21.6	2.4	-4.5

Where, %DP/P(Case#) = 100 x {P(Case#)/P(Reference) - 1}

Table 4 Parameters and Case Numbers

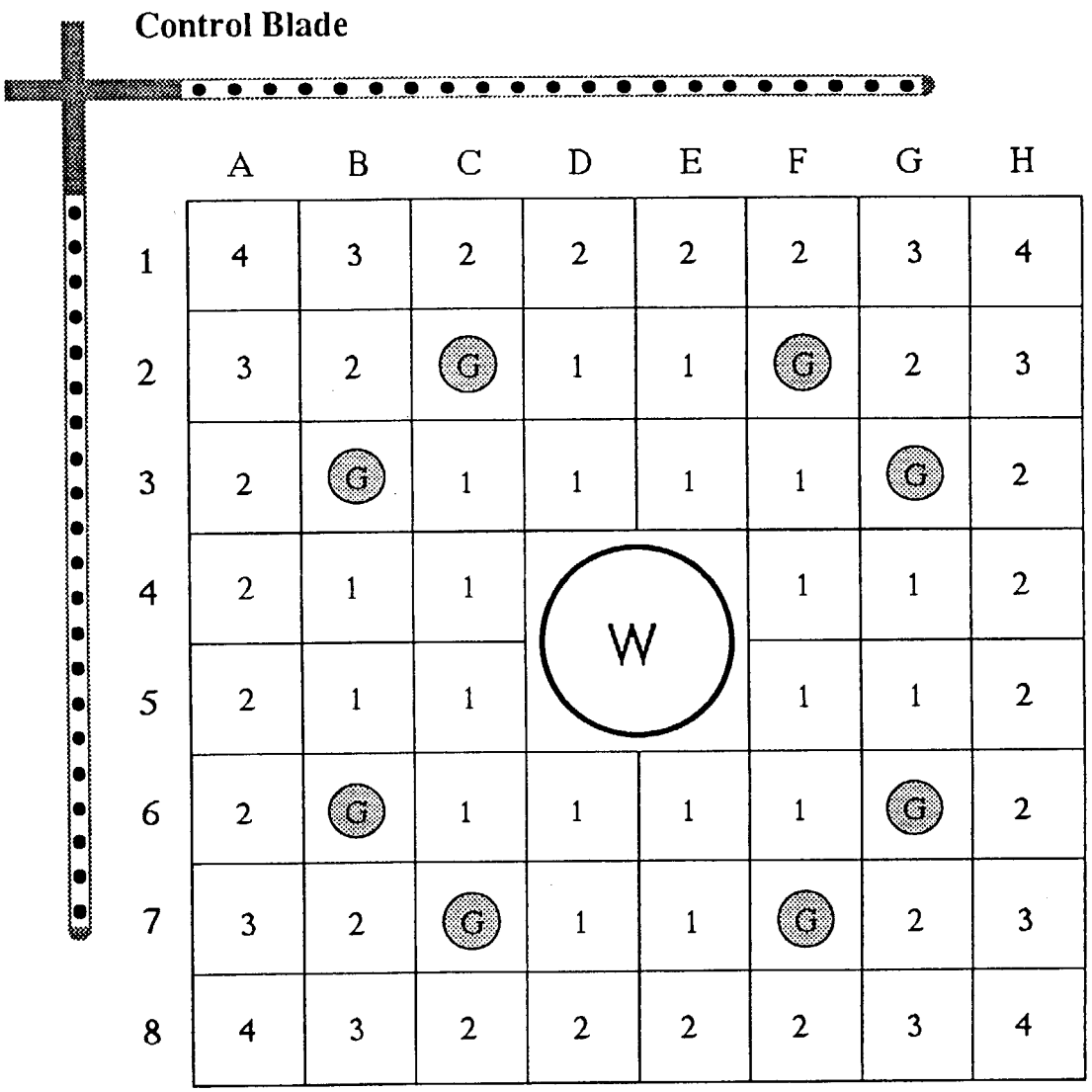
				Initial Enrichment = 3.8 w/o			
				Burnup			
				Fresh	10GWD/Mt	30 GWd/Mt	40GWd/Mt
Cooling Time	FPs	Burnup Profile	Void Profile	Case#			
1 year	Yes	Yes	Yes	1	2	3	4
			40%Uniform		5	6	7
			70%Uniform		8	9	10
		No	Yes		11	12	13
			40%Uniform		14	15	16
			70%Uniform		17	18	19
	No	Yes	Yes		20	21	22
			40%Uniform		23	24	25
			70%Uniform		26	27	28
		No	Yes		29	30	31
			40%Uniform		32	33	34
			70%Uniform		35	36	37
5 years	Yes	Yes	Yes		38	39	40
			40%Uniform		41	42	43
			70%Uniform		44	45	46
		No	Yes		47	48	49
			40%Uniform		50	51	52
			70%Uniform		53	54	55
	No	Yes	Yes		56	57	58
			40%Uniform		59	60	61
			70%Uniform		62	63	64
		No	Yes		65	66	67
			40%Uniform		68	69	70
			70%Uniform		71	72	73

Note ; 40% or 70% Uniform Void Cases are Considered as Void Profile = No

Table 5 Importance of Benchmark Problem Cases

				Initial Enrichment = 3.8 w/o			
				Burnup			
				Fresh	10GWD/Mt	30 GWd/Mt	40GWd/Mt
Cooling Time	FPs	Burnup Profile	Void Profile	Case			
1 year	Yes	Yes	Yes	☆	×	△	◎
			40%Uniform		×	△	○
			70%Uniform		×	△	○
		No	Yes		×	△	○
			40%Uniform		×	△	○
			70%Uniform		×	△	◎
	No	Yes	Yes		×	△	◎
			40%Uniform		×	△	○
			70%Uniform		×	△	○
		No	Yes		×	△	○
			40%Uniform		×	△	○
			70%Uniform		×	△	◎
5 years	Yes	Yes	Yes		○	◎	☆
			40%Uniform		△	○	◎
			70%Uniform		△	◎	◎
		No	Yes		△	◎	☆
			40%Uniform		△	○	☆
			70%Uniform		○	◎	☆
	No	Yes	Yes		△	◎	☆
			40%Uniform		△	○	◎
			70%Uniform		△	◎	◎
		No	Yes		△	◎	☆
			40%Uniform		△	○	☆
			70%Uniform		△	◎	☆

Note --- ☆ : Very Important、(9 Cases) ◎ : Important、(14Cases) ○ : Rather Important、(16Cases) △ : Little Important、(22 Cases) × : not Important (12 Cases)



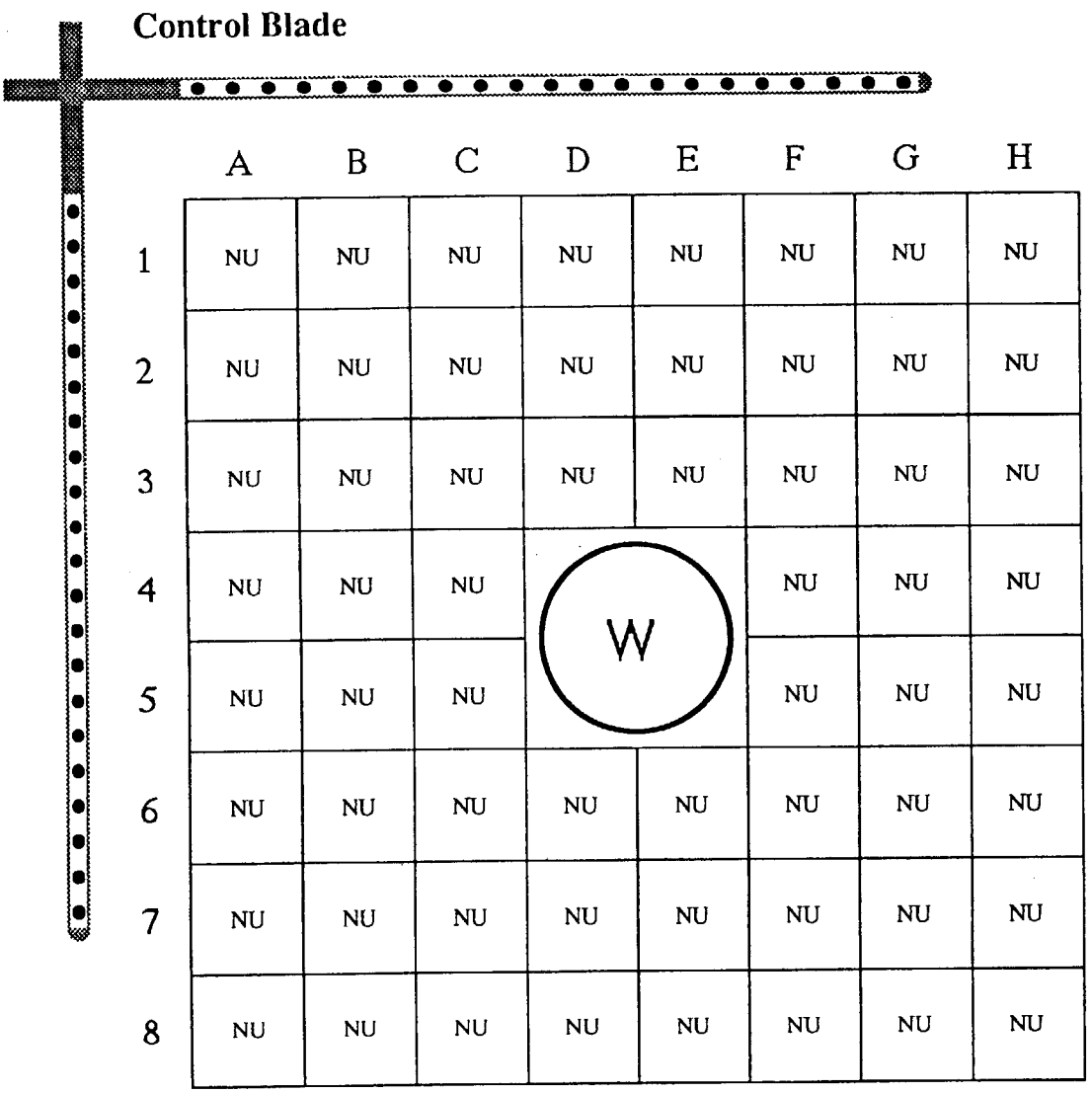
Gd Rod

(W) Large Water Rod

Rod Type#	U-235 Enrichment	Gadolinia Content	No. of Rods
1	4.9 w/o		20
2	3.6		20
3	3.0		8
4	2.3		4
G	3.0	7 w/o	8
W	Water Rod		1

Assembly Average U-235 Enrichment = 3.80 w/o

Fig. 1 Typical BWR Fuel Assembly



Rod Type#	U-235 Enrichment	No. of Rods
NU	0.71 w/o	60
W	Water Rod	1

Assembly Average U-235 Enrichment = 0.71 w/o

Fig. 2 Blanket Fuel Assembly

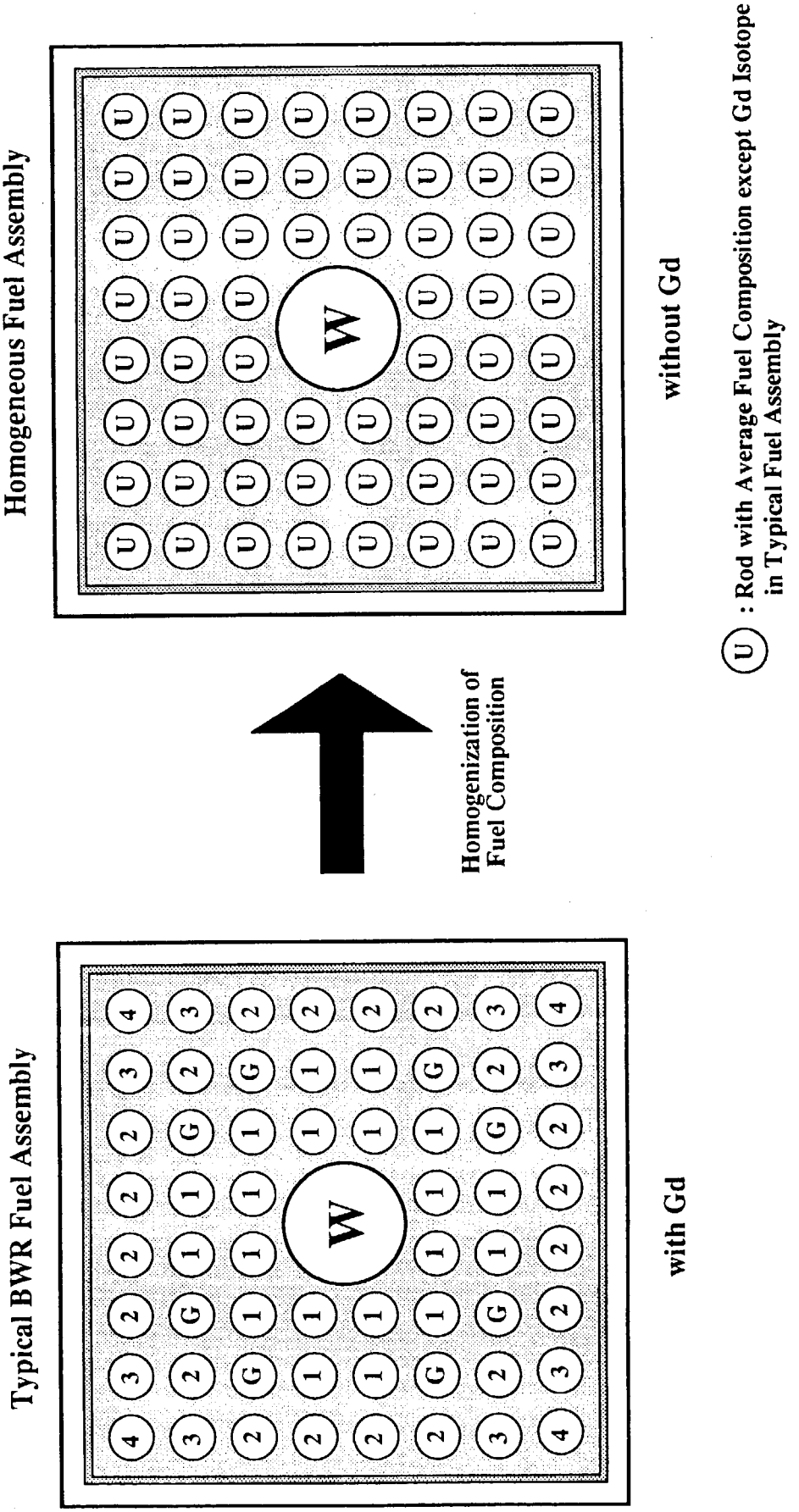


Fig. 3 Homogenization of Fuel Composition in Typical Fuel Assembly

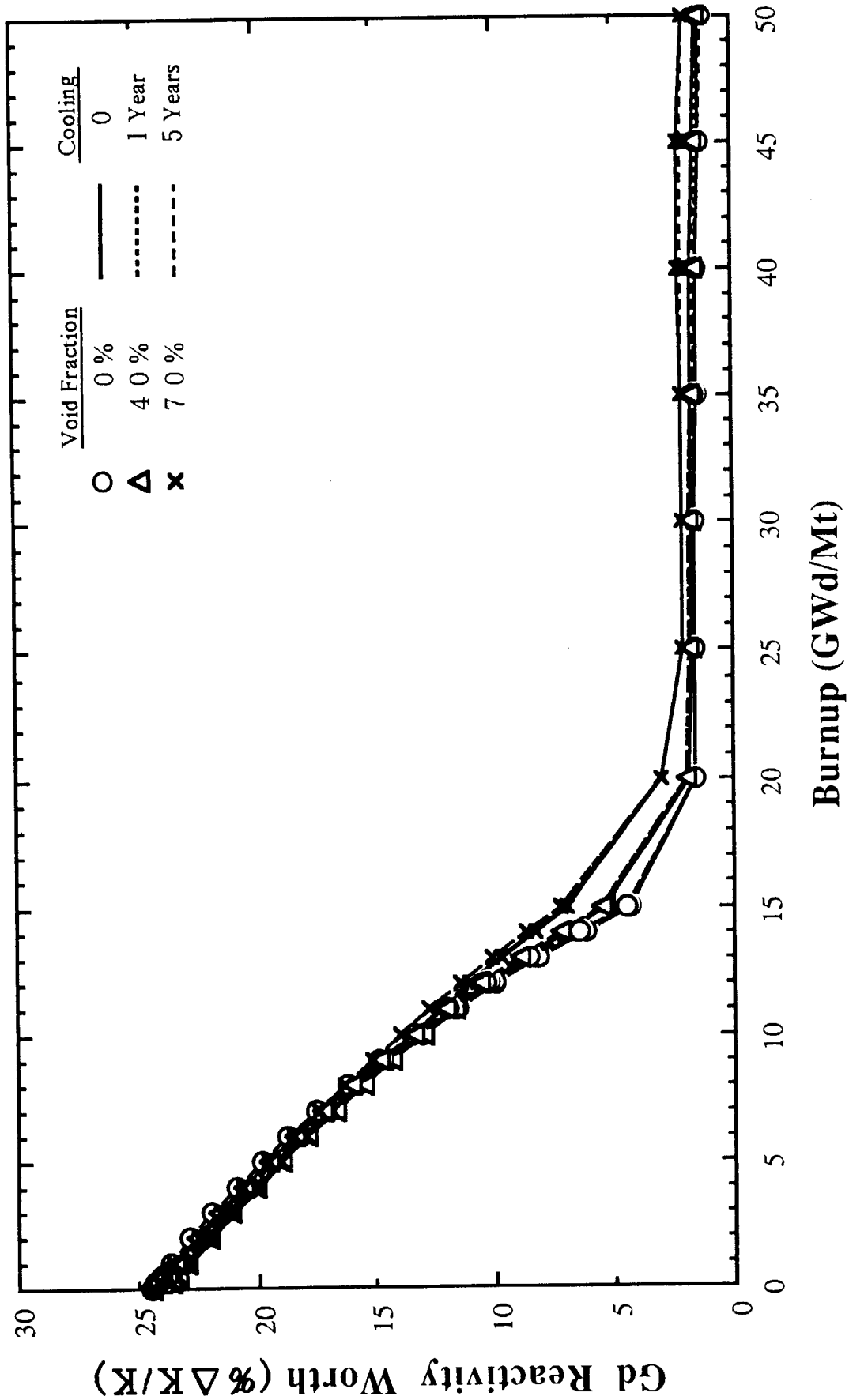


Fig. 4 Gd Reactivity Worth in Fuel Assembly

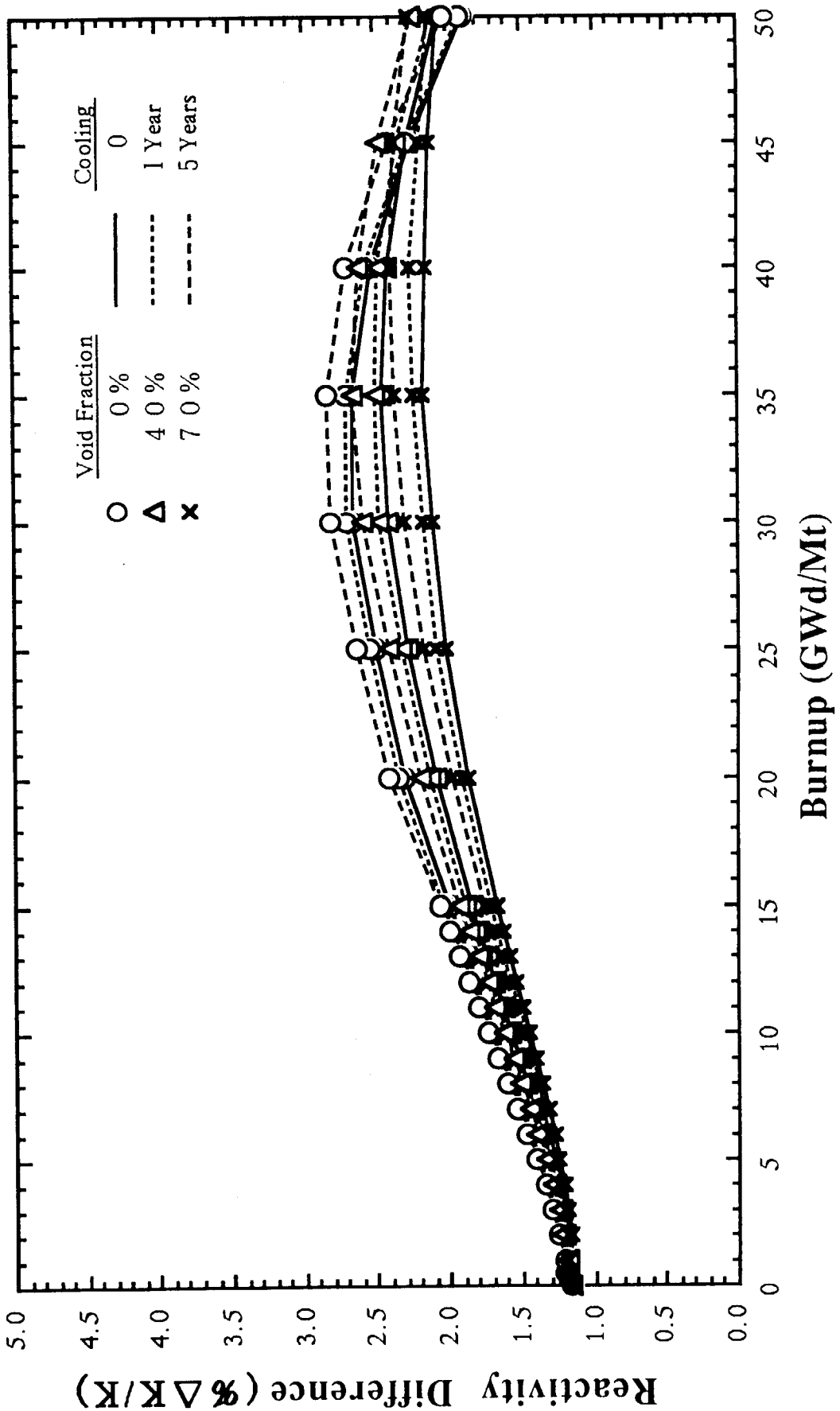


Fig. 5 Reactivity Difference due to the Homogenization of Fuel Composition

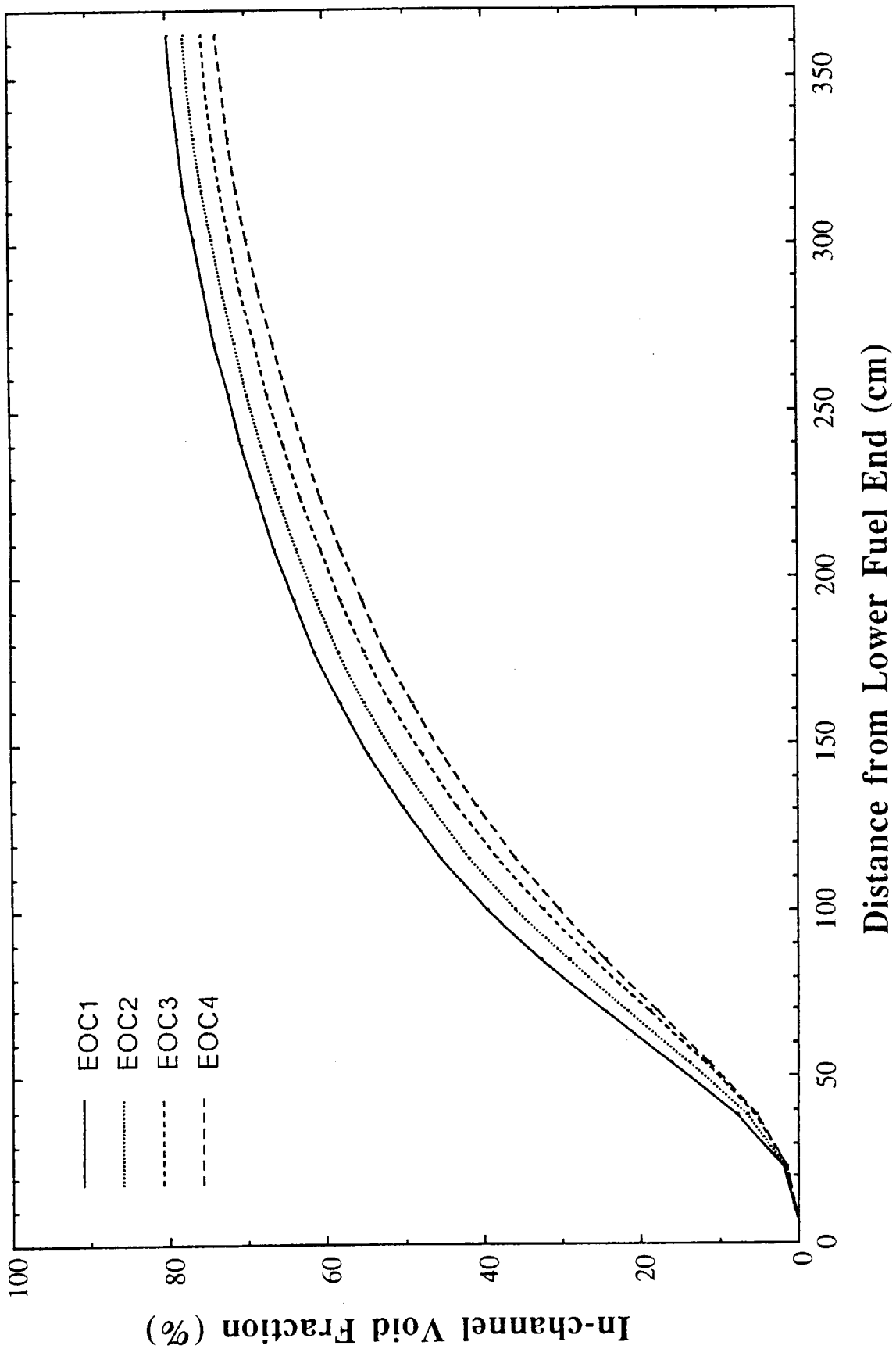


Fig. 6 Void Profile in Typical Fuel Core Based on Haling Calculation

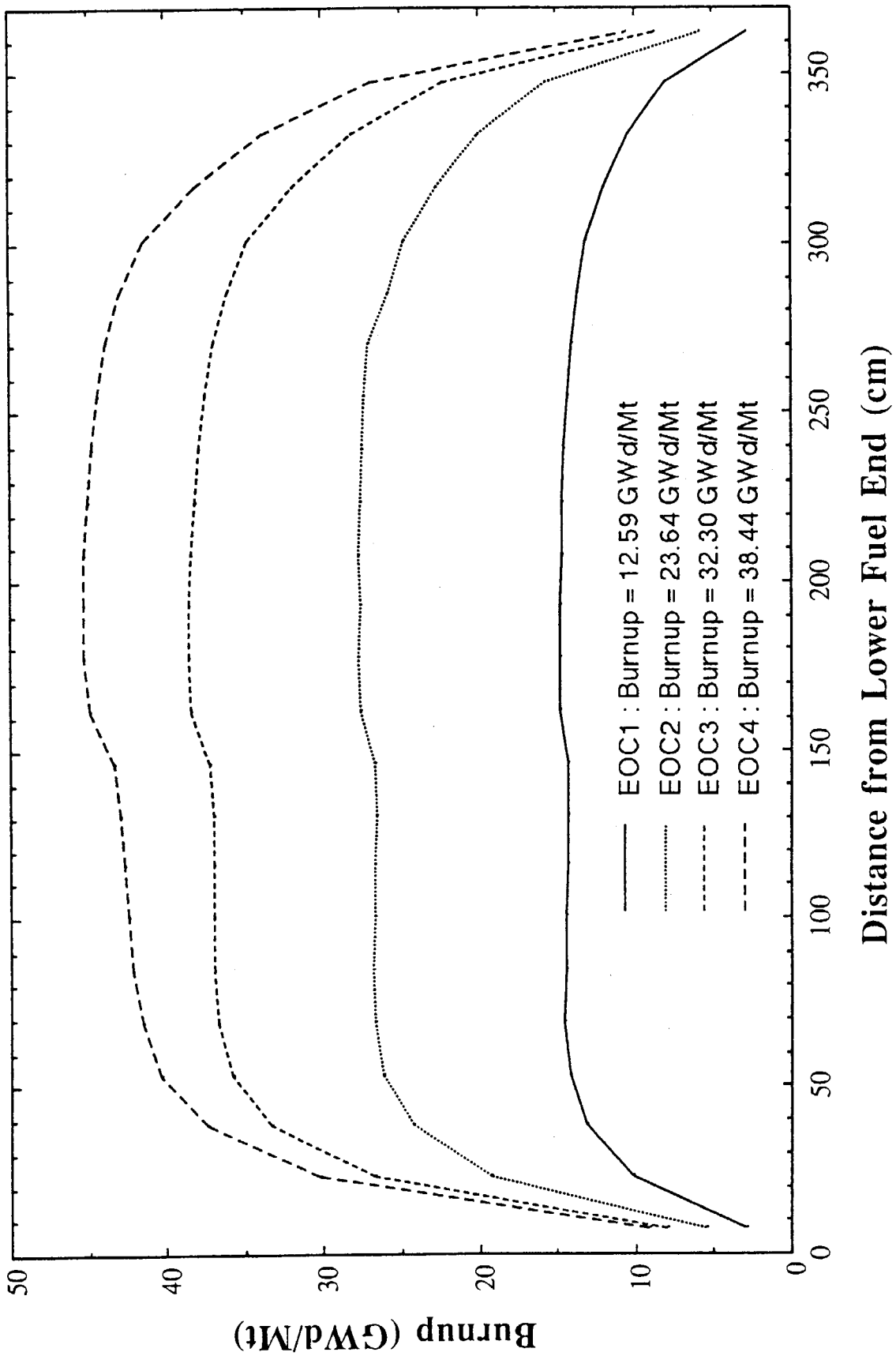


Fig.7 Burnup Profile in Typical Fuel Core Based on Hailing Calculation

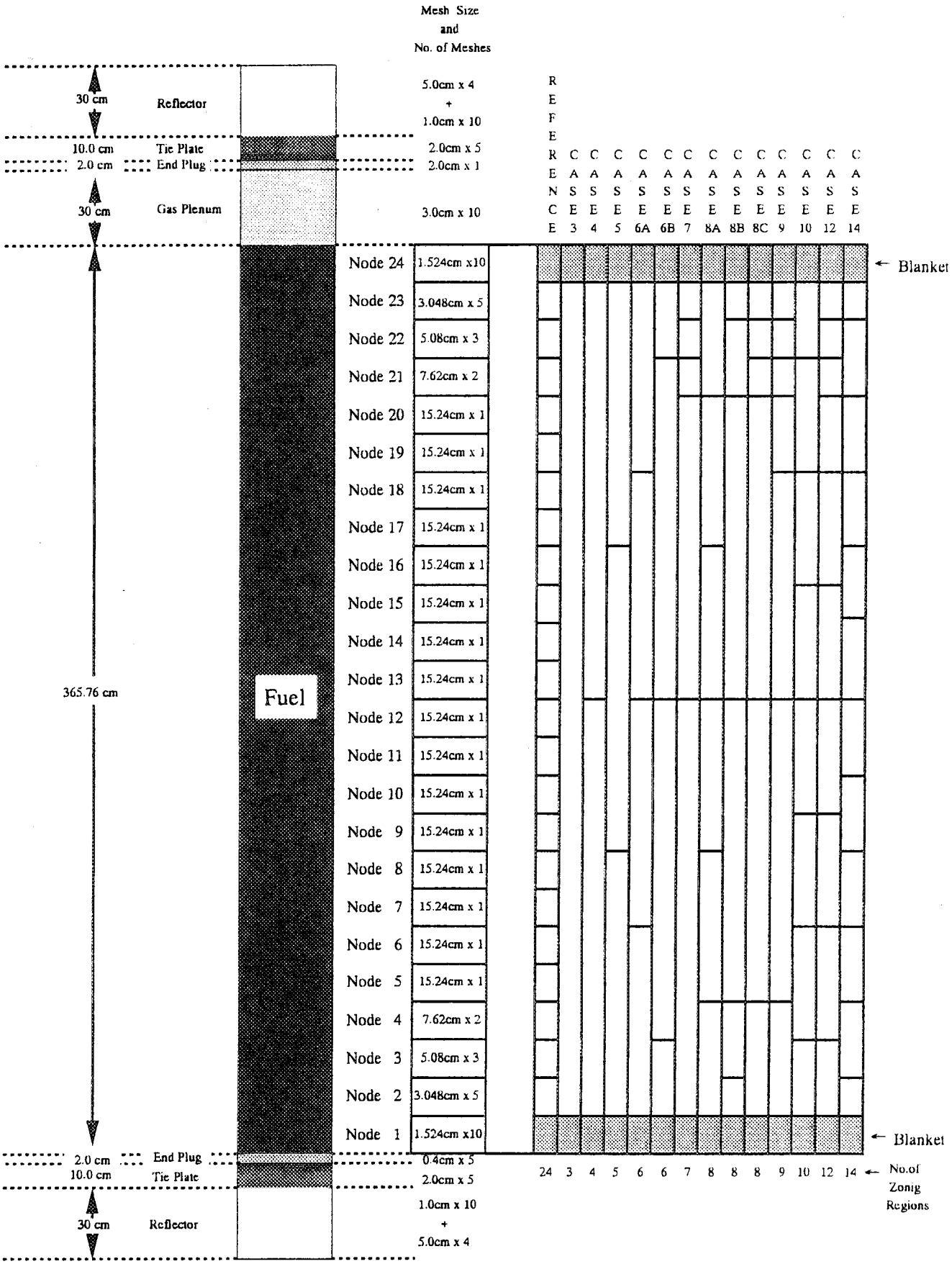
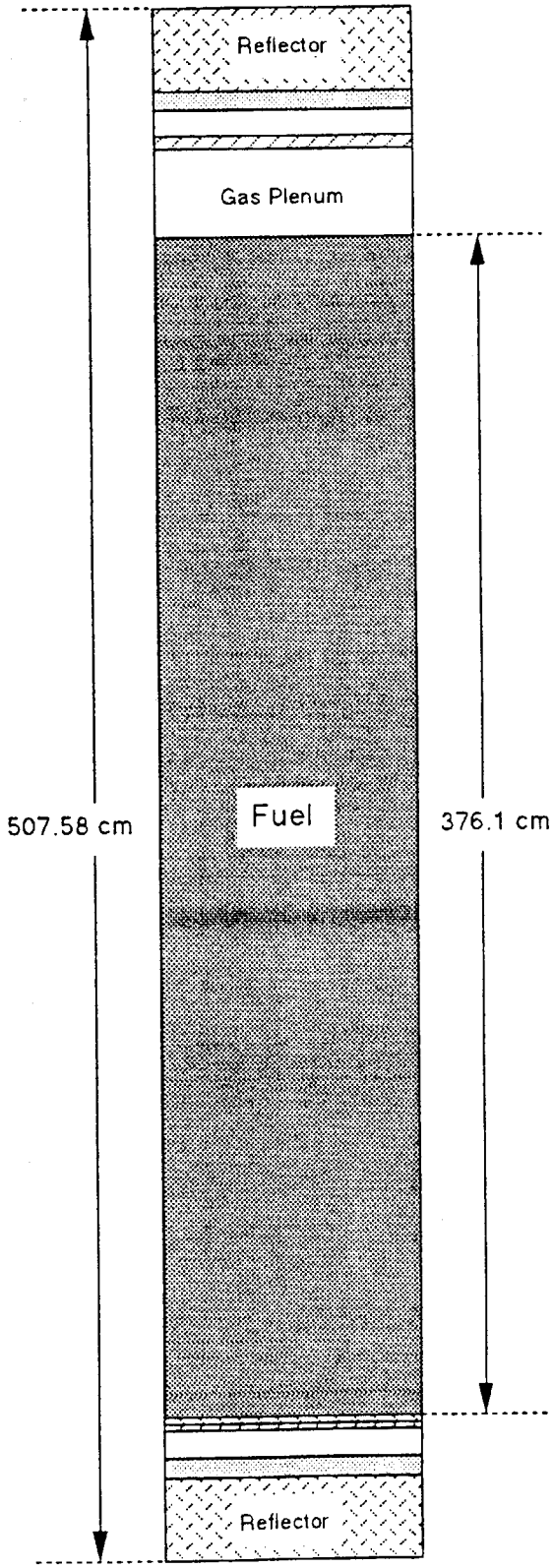


Fig. 8 Studied Cases on the Averaging of Axial Fuel Composition



Zone No.	Region	Length (cm)	Materials (Volume%)
1	Reflector	25.4	Water (100)
2	Handle Plate	8.4	Water (93.6) SS-304 (6.4)
3	Upper Tie Plate	8.5	Water (81.0) SS-304 (19.0)
4	Upper End Plug	2.1	Water (59.7) Zr-2 (40.5) IX-750 (15.4)
5	Gas Plenum and Spring	30.5	Water (59.7) Zr-2 (7.8) SS-302 (2.0) Void (30.5)
6-11	Fuel	376.1 (total)	Water (59.6) Fuel+Zr-2 (40.6)
12	Lower End Plug	1.6	Water (63.5) Zr-2 (36.5)
13	Lower Tie Plate	11.8	Water (81.9) SS-304 (18.1)
14	Oriffice	12.7	Water (75) SS-304 (25)
15	Lower Core Plate	5.08	Water (25) SS-304 (75)
16	Reflector	25.4	Water (100)

Fig. 9 BWR Axial Assembly Model in generating ORIGEN2 Cross Section Library

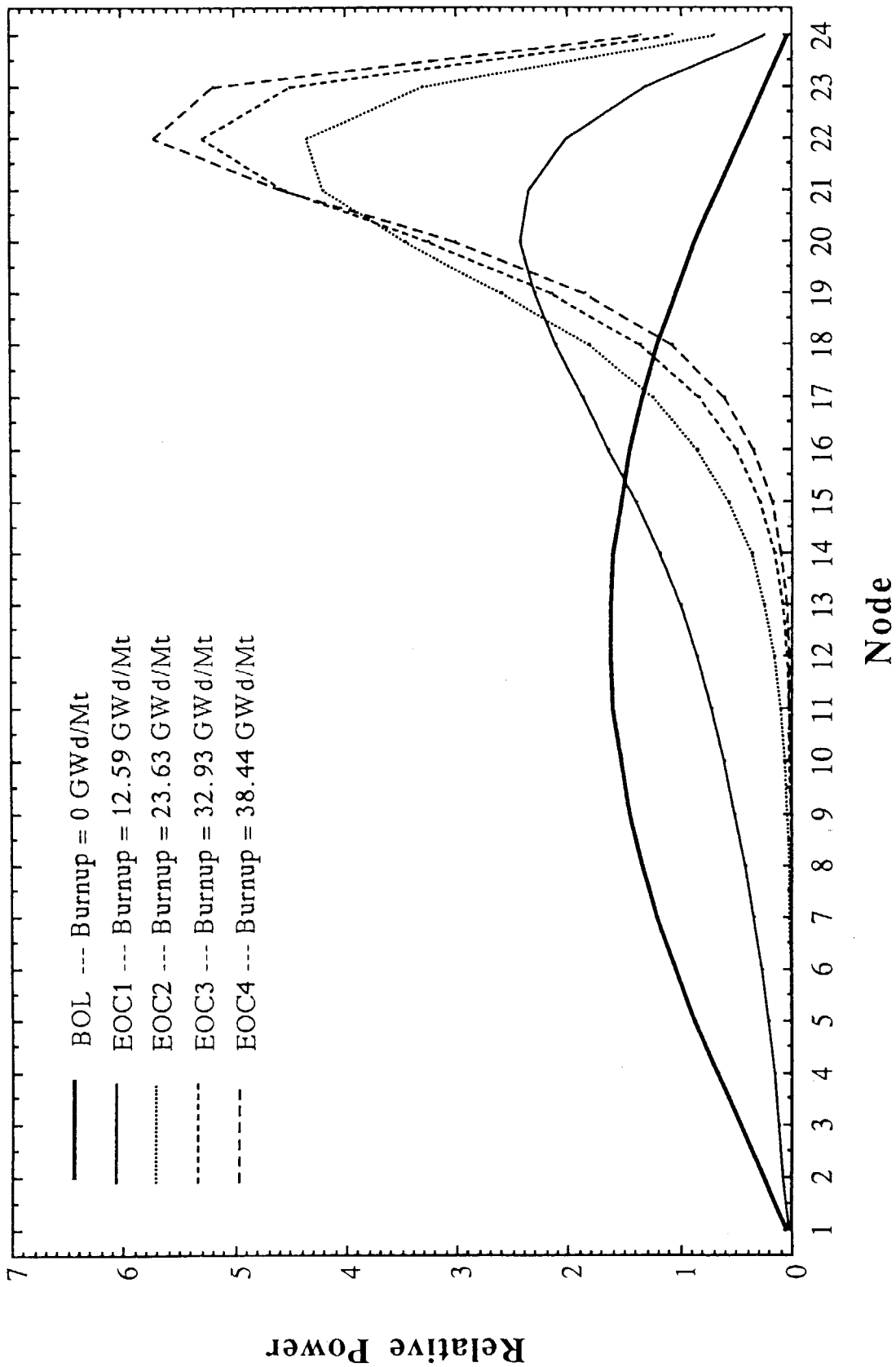


Fig. 10 Comparison of Relative Power Distributions in Reference Calculation

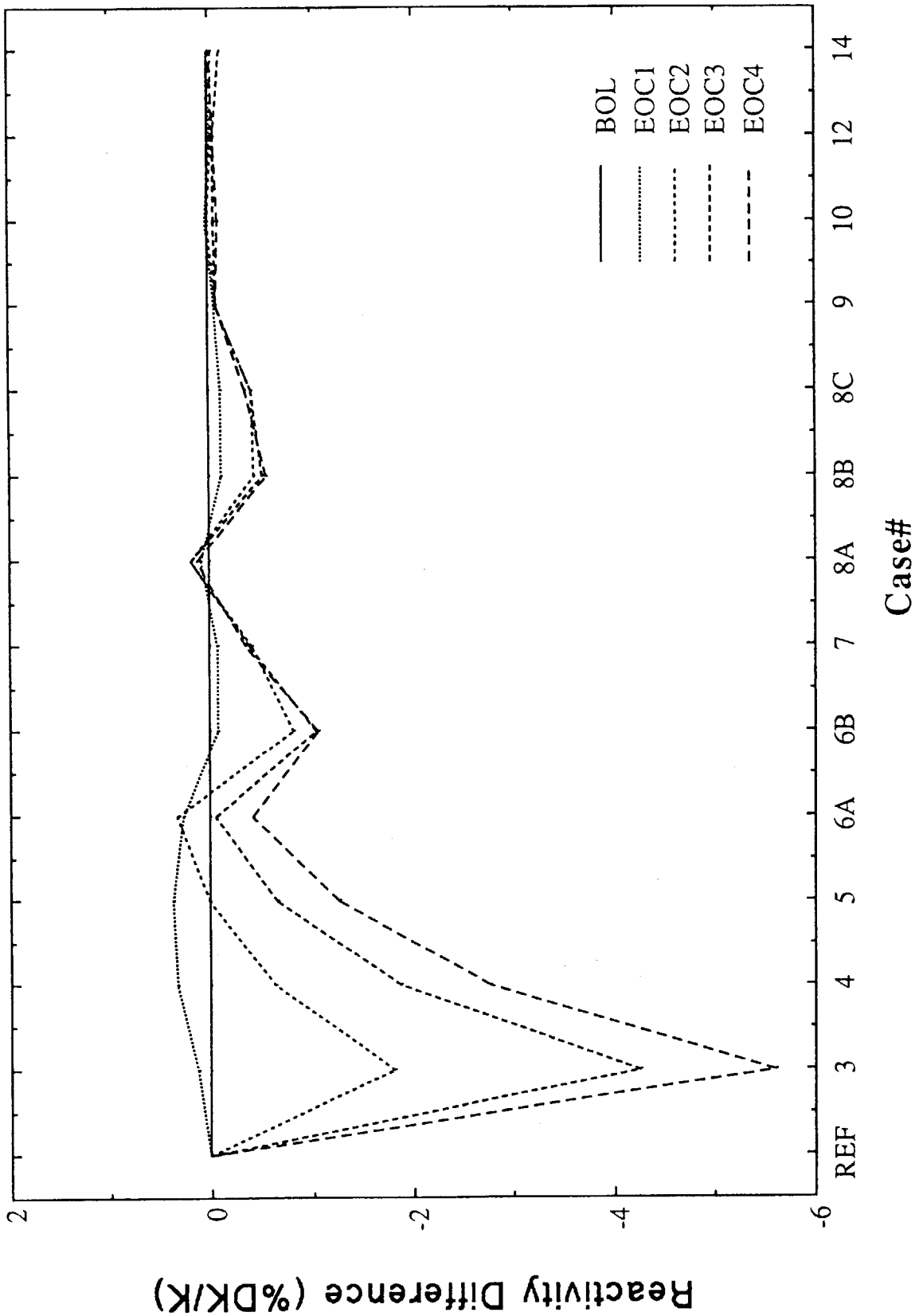


Fig.11 Comparison of Reactivity Differences in Studied Cases

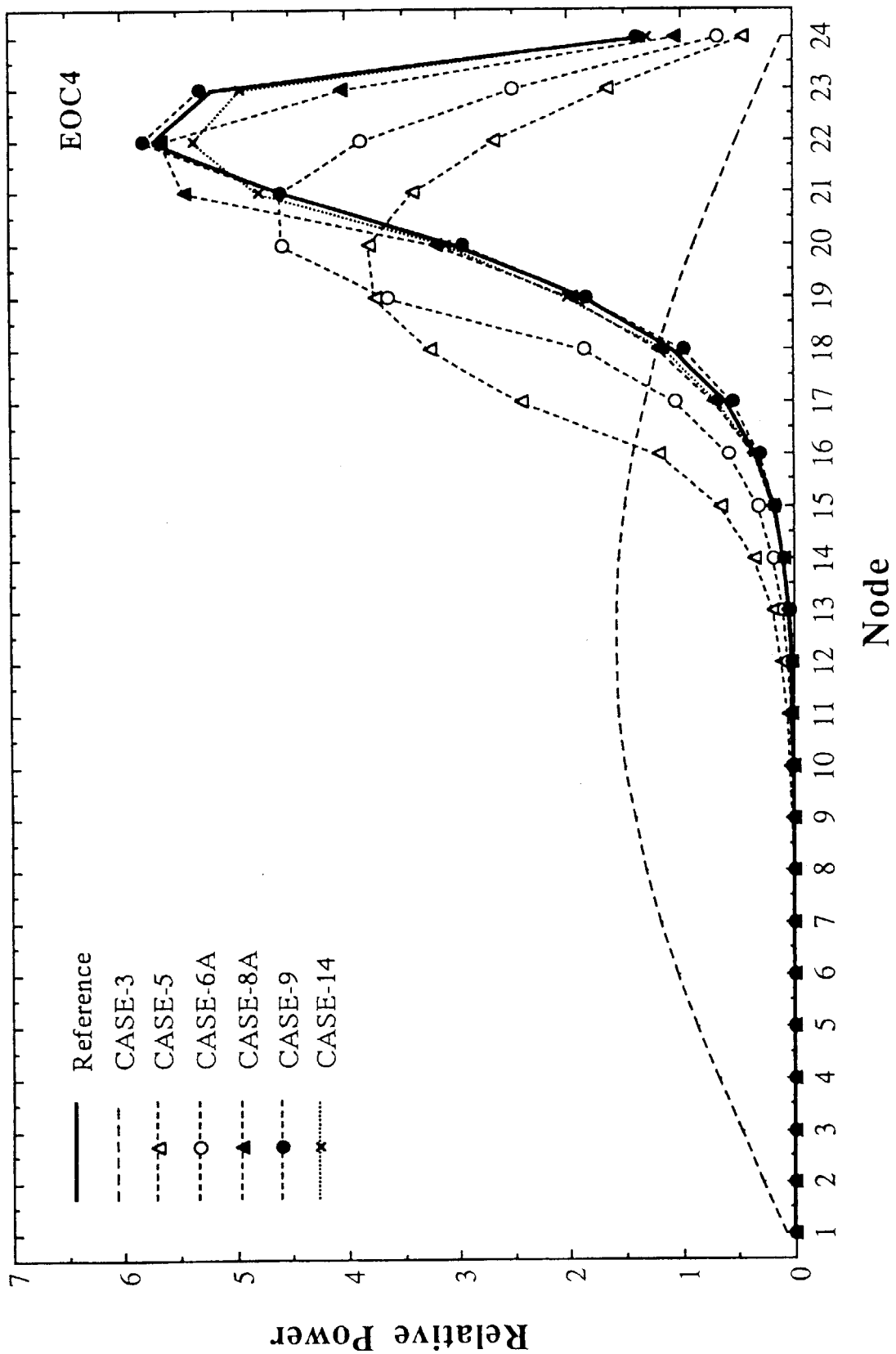


Fig.12 Comparison of Relative Power Distributions in Studied Cases

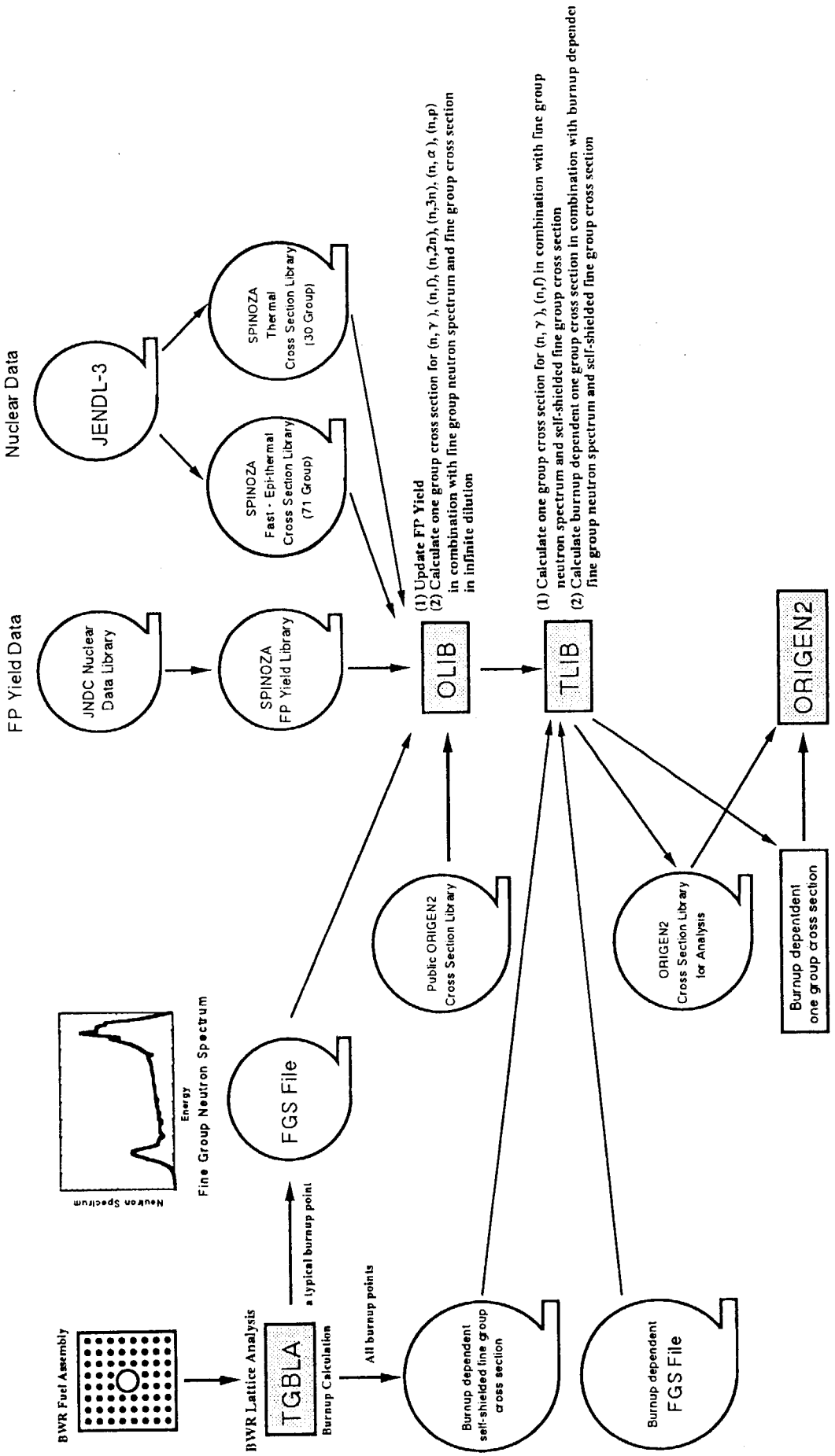


Fig. 13 Schematic Flow Diagram in SPINOZA System

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Appendix IV

Documents on Phase IIIA Benchmarks Presented at 1996 Meeting in Paris

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Appendix IV.1

Results from CEA/IPSN

[CEA/IPSN]

Institute : IPSN (France)

Participants : G. POULLOT, X. BOUDIN

Neutron Data Library : CEA 86 library issued from jef 1.

Neutron Code : APOLLO I and MORET III. MORET needs a 16 group structure, using the self-shielded cross-sections from APOLLO with an isotropic scattering (P0).

Statistical error : $3\sigma = 150$ pcm

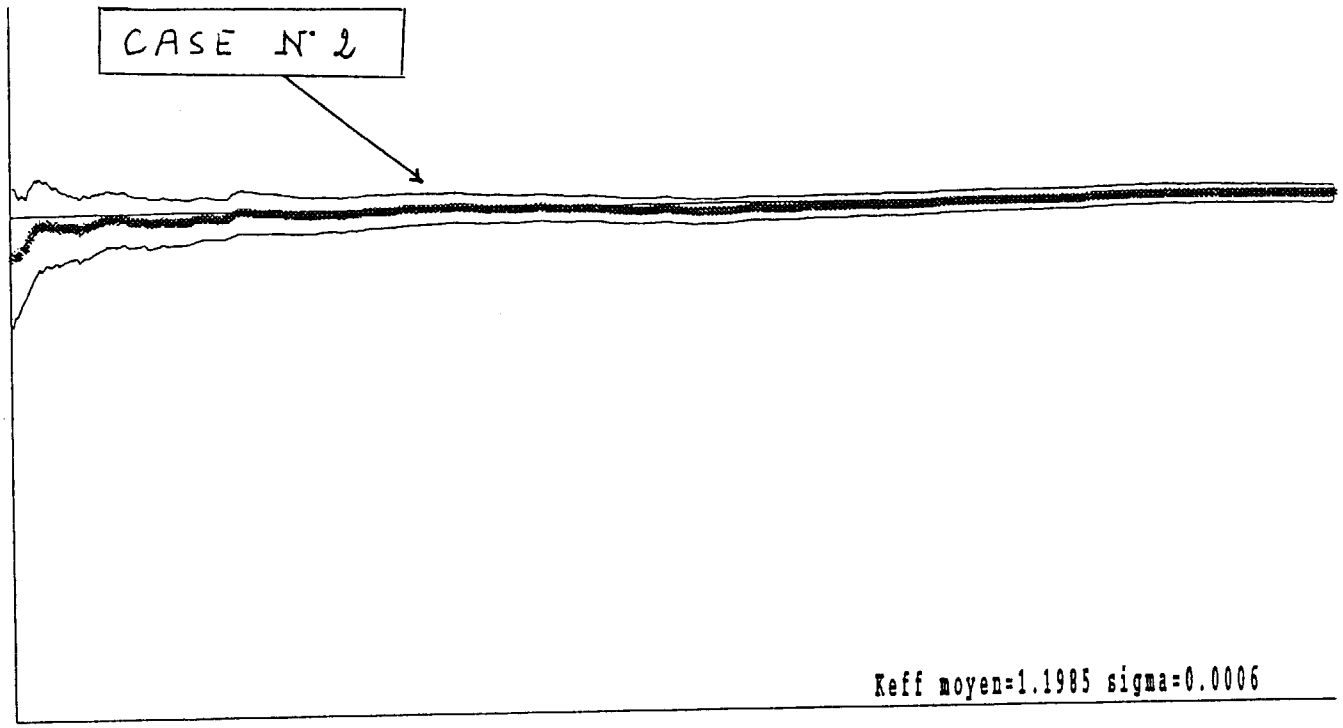
Other information : we performed, in each case, 2 calculations.

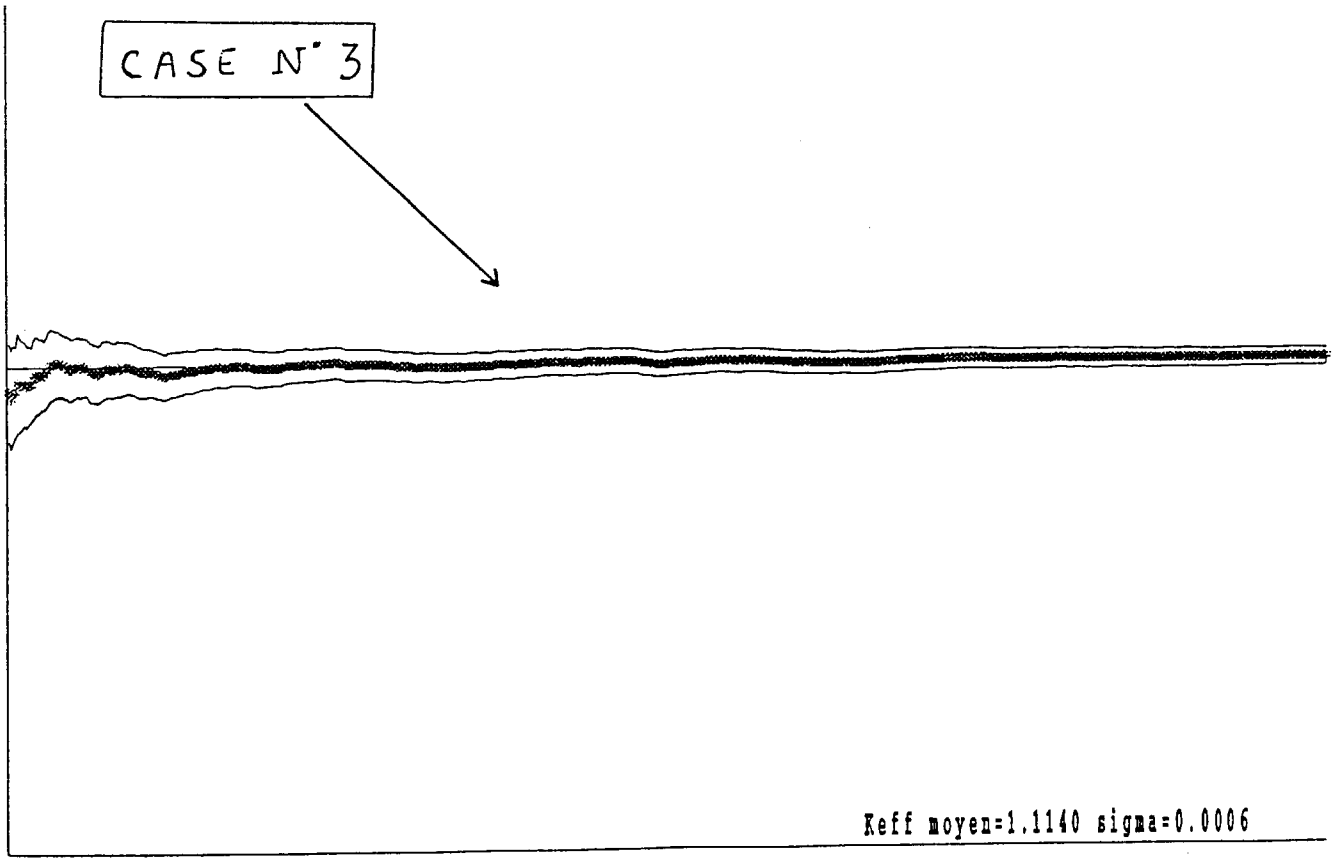
The first one is used to find the best neutron distribution source for the second calculation. In each volume, the neutron distribution, given by the first calculation (from the eigen vector), is considered as source for the second calculation. With an additionnal hypothesis : the neutron distribution is located at the center of the volumes.

Multiplications factors (keff) for the studied cases

Cooling time	Fission Products	Burn up Profile	Void Profile		Burn up		
				0 GWd/t	20 GWd/t	30 GWd/t	40 GWd/t
1 year	yes	yes	yes	**1** 1,4095	**2** 1,1985	**3** 1,1140	**4** 1,0339
5 years	yes	yes	yes		**5** 1,1845	**6** 1,0950	**7** 0,9998
		no	yes				**8** 0,9927
		no	40% (uniform)		**9** 1,1901	**10** 1,0751	**11** 0,9621
		no	70% (uniform)		**12** 1,2021	**13** 1,0992	**14** 1,0077
	no	yes	yes				**15** 1,1066
		no	yes				**16** 1,1079
		no	40% (uniform)		**17** 1,2625	**18** 1,1718	**19** 1,0760
		no	70% (uniform)		**20** 1,2738	**21** 1,1958	**22** 1,1231

2 convergence curves for the cases 2 and 3 (for the second calculation)





Results from CEA/IPSN

Fractionnal fission densities

Case	Region	F.F.D.
	1	1,2383 E-03
	2	6,5266 E-02
	3	0,56653
	4	0,313294
1	5	4,0597 E-02
	6	8,62726 E-03
	7	3,3398 E-02
	8	1,0128 E-03
	9	9,1322 E-05
	1	9,36117 E-05
	2	2,94469 E-03
	3	4,49458 E-02
	4	0,357968
5	5	0,239116
	6	0,1312794
	7	0,1211599
	8	8,52133 E-02
	9	1,727973 E-02
	1	2,0732 E-06
	2	7,6693 E-05
	3	2,7351 E-03
	4	0,13318
6	5	0,23015
	6	0,18870
	7	0,21806
	8	0,18518
	9	4,1858 E-02
	1	0,0
	2	1,64976 E-05
	3	5,3566 E-04
	4	0,077385
7	5	0,18540
	6	0,18655
	7	0,25012
	8	0,2532
	9	4,6735 E-02
	1	8,1363 E-04
	2	3,5093 E-02
	3	0,46388
	4	0,42379
14	5	5,6957 E-02
	6	1,19524 E-02
	7	5,2936 E-03
	8	1,8869 E-03
	9	3,1979 E-04

Appendix IV.2

Results of NEA/NSC Burnup Credit Benchmark Phase IIIA: Comparative Calculation with SCALE27 and MGCL-JINS Cross Section libraries

[NUPEC/INS]

Fifth Burnup Credit Benchmarks Meeting (Paris, June 25-27, 1996)

Results of NEA/NSC Burnup Credit Benchmark Phase IIIA: Comparative Calculation with SCALE27 and MGCL-JINS Cross-section Libraries

Susumu Mitake and Osamu Sato
Institute of Nuclear Safety, NUPEC

1. Method of Calculation

The following two sets of cross-section libraries are applied in the study of burnup credit criticality benchmark Phase IIIA in which the effect of axial burnup profile in a BWR fuel assembly is investigated.

(a) SCALE 27 group library

Prepared in the SCALE-4.1 code systems, based on ENDF/B-IV and processed by NITAWL-II (Nordheim method).

(b) MGCL-JINS 137 group library

Based on JENDL-3.2 (for fission product nuclides) and ENDF/B-IV (for U, Pu and other nuclides), and processed by MAIL-JINS (Bondarenko method).

Monte Carlo code KENO-V.a is used to calculate the multiplication factors and the fractional fission densities in each axial region. The statistical parameters for Monte Carlo calculations are set as follows:

Number of histories per generation	2,400
Number of generations	403
Initial skipped generations	3
Total number of histories	960,000

These values of parameters were selected and adopted from our results of the phase IIB calculations.

2. Comparison of Results with Two Cross-section Libraries

The calculated multiplication factors are comparatively listed in Table 1. And, Table 2 shows the reactivity changes ($\Delta k/kk'$) due to calculation parameters such as: (a) consideration of fission product (FP) nuclides, (b) consideration of burnup profile, (c) modeling of void profile, and (d) cooling time. The reactivity decreases from initial fresh fuel by burnup are

summarized in Table 3. The axial distributions of fission densities (normalized to unity) are compared in Fig.1.

(a) Consideration of FP nuclides

Calculations with MGCL-JINS brings larger values in the reactivity changes due to consideration of FP nuclides. The largest difference is found for the case in which both burnup profile and void profile are taking into account in the calculation conditions.

(b) Consideration of burnup profile

The reactivity changes due to consideration of burnup profile are well agreed for two libraries.

(c) Modeling of void profile

Smaller multiplication factors are calculated when the uniform void profiles of 40 % void are assumed, compared to the case in which exact void profiles are considered. The reactivity changes are calculated to be 2 to 3 %Δk/kk'. But, when the uniform void profiles of 70 % void are assumed, the larger multiplication factors, up to 1 %Δk/kk' in the reactivity change to the exact profile case, are resulted. Thus, the 70 % uniform void cases may be on the safety sides. The largest difference between the SCALE27 and MGCL-JINS results is found for the case without FP nuclides. For the cases of the uniform void profile assumption, MGCL-JINS brings greater multiplication factors compared to SCALE27.

(d) Cooling time

Smaller multiplication factors are calculated when the cooling time increases from 1 year to 5 years. Reduction in the multiplication factors indicates to be dependent on the burnup, but its tendency is different among the sets of the cross-section libraries. MGCL-JINS shows smaller reduction at the burnup of 20 GWD/tU, and nearly equal reduction at 30 GWD/tU, but greater at 40 GWD/tU, compared to those calculated with SCALE27.

Table 1 Results of NEA/NSC burnup credit benchmark Phase IIIA

case	FP	Burnup Profile	Void Profile	Burnup [GWd/tU]	Cooling time [yr]	keff±1σ	
						SCALE27 lib.	MGCL-JINS lib.
1	Fresh fuel			0		1.3902 ±0.0005	1.3989 ±0.0006
2	Yes	Yes	Yes	20	1	1.1876 ±0.0005	1.1918 ±0.0006
3				30		1.1047 ±0.0006	1.1082 ±0.0006
4				40		1.0209 ±0.0007	1.0256 ±0.0007
5	Yes	Yes	Yes	20	5	1.1742 ±0.0005	1.1794 ±0.0006
6				30		1.0853 ±0.0006	1.0883 ±0.0007
7				40		0.9952 ±0.0007	0.9970 ±0.0007
8	Yes	No	Yes	40	5	0.9859 ±0.0005	0.9877 ±0.0005
9	Yes	No	40% uniform	20	5	1.1799 ±0.0005	1.1838 ±0.0005
10				30		1.0658 ±0.0005	1.0706 ±0.0005
11				40		0.9560 ±0.0004	0.9579 ±0.0005
12	Yes	No	70% uniform	20	5	1.1890 ±0.0005	1.1959 ±0.0005
13				30		1.0908 ±0.0005	1.0943 ±0.0005
14				40		0.9948 ±0.0004	0.9973 ±0.0004
15	No	Yes	Yes	40	5	1.0941 ±0.0006	1.1000 ±0.0006
16	No	No	Yes	40	5	1.0985 ±0.0005	1.1044 ±0.0005
17	No	No	40% uniform	20	5	1.2480 ±0.0005	1.2555 ±0.0006
18				30		1.1592 ±0.0005	1.1656 ±0.0005
19				40		1.0685 ±0.0004	1.0715 ±0.0005
20	No	No	70% uniform	20	5	1.2606 ±0.0005	1.2687 ±0.0005
21				30		1.1852 ±0.0005	1.1910 ±0.0005
22				40		1.1096 ±0.0005	1.1141 ±0.0005

Table 2 Reactivity change due to various parameters

Parameter	Condition	Burnup [GWD/tU]	%Δk/kk'	
			SCALE27 lib.	MGCL-JINS lib.
FP (No-Yes)	Burnup profile=Yes Void profile=Yes Cooling time =5years	40	9.087% ±0.084%	9.396% ±0.088%
	Burnup profile=No Void profile=40% uni. Cooling time =5years	20	4.625% ±0.047%	4.820% ±0.051%
		30	7.558% ±0.055%	7.612% ±0.056%
		40	11.017% ±0.060%	11.066% ±0.066%
	Burnup profile=No Void profile=70% uni. Cooling time =5years	20	4.774% ±0.046%	4.798% ±0.050%
		30	7.309% ±0.052%	7.422% ±0.057%
		40	10.406% ±0.057%	10.512% ±0.058%
Burnup profile (No-Yes)	FP=Yes Void profile=Yes Cooling time =5years	40	-0.950% ±0.085%	-0.947% ±0.087%
	FP=No Void profile=Yes Cooling time =5years	40	0.363% ±0.065%	0.361% ±0.068%
Void profile (40%-Yes)	FP=Yes Burnup profile=No Cooling time=5years	40	-3.139% ±0.083%	-3.117% ±0.087%
	FP=No Burnup profile=No Cooling time=5years	40	-2.551% ±0.058%	-2.785% ±0.059%
Void profile (70%-Yes)	FP=Yes Burnup profile=No Cooling time=5years	40	0.909% ±0.057%	0.981% ±0.058%
	FP=No Burnup profile=No Cooling time=5years	40	0.916% ±0.057%	0.789% ±0.058%
Cooling time (5-1year)	FP=Yes Burnup profile=Yes Void profile=Yes	20	-0.960% ±0.053%	-0.881% ±0.057%
		30	-1.614% ±0.069%	-1.655% ±0.074%
		40	-2.535% ±0.095%	-2.803% ±0.096%

Table 3 Reactivity change from fresh fuel

(SCALE 27 group lib.) %Δk/kk'

FP	Yes	Yes	Yes	Yes	No	No
Burnup Profile	Yes	Yes	No	No	No	No
Void Profile	Yes	Yes	40% uniform	70% uniform	40% uniform	70% uniform
Cooling time [yr]	1	5	5	5	5	5
0GWd/tU	0%	0%	0%	0%	0%	0%
20GWd/tU	-12.27%	-13.23%	-12.82%	-12.17%	-8.20%	-7.40%
30GWd/tU	-18.59%	-20.21%	-21.90%	-19.75%	-14.34%	-12.44%
40GWd/tU	-26.02%	-28.55%	-32.67%	-28.59%	-21.65%	-18.19%

(MGCL 137 group lib.) %Δk/kk'

FP	Yes	Yes	Yes	Yes	No	No
Burnup Profile	Yes	Yes	No	No	No	No
Void Profile	Yes	Yes	40% uniform	70% uniform	40% uniform	70% uniform
Cooling time [yr]	1	5	5	5	5	5
0GWd/tU	0%	0%	0%	0%	0%	0%
20GWd/tU	-12.42%	-13.31%	-12.99%	-12.14%	-8.17%	-7.34%
30GWd/tU	-18.75%	-20.41%	-21.92%	-19.90%	-14.31%	-12.48%
40GWd/tU	-26.02%	-28.82%	-32.91%	-28.79%	-21.85%	-18.27%

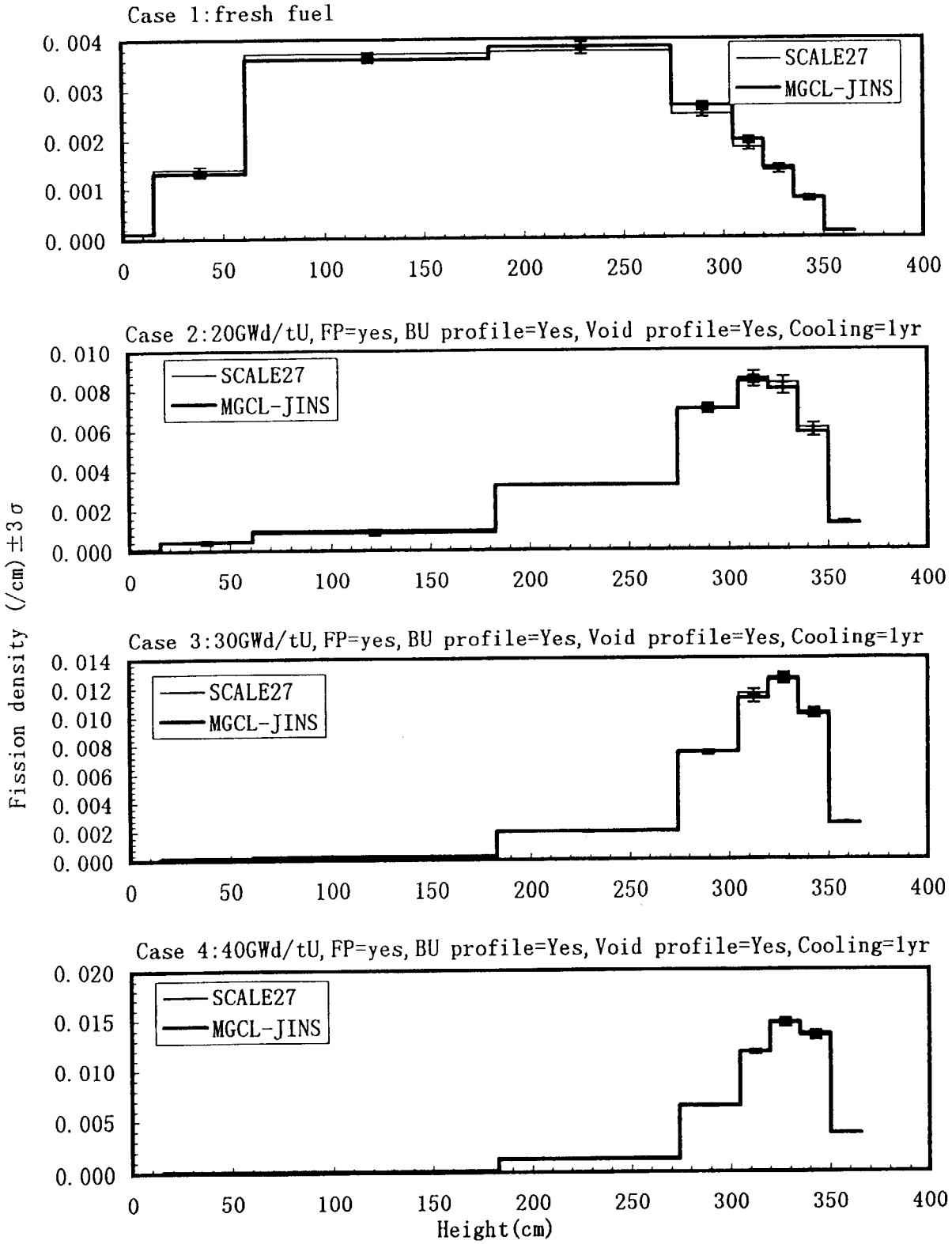


Fig. 1 Axial distribution of normalized fission densities (1/6)

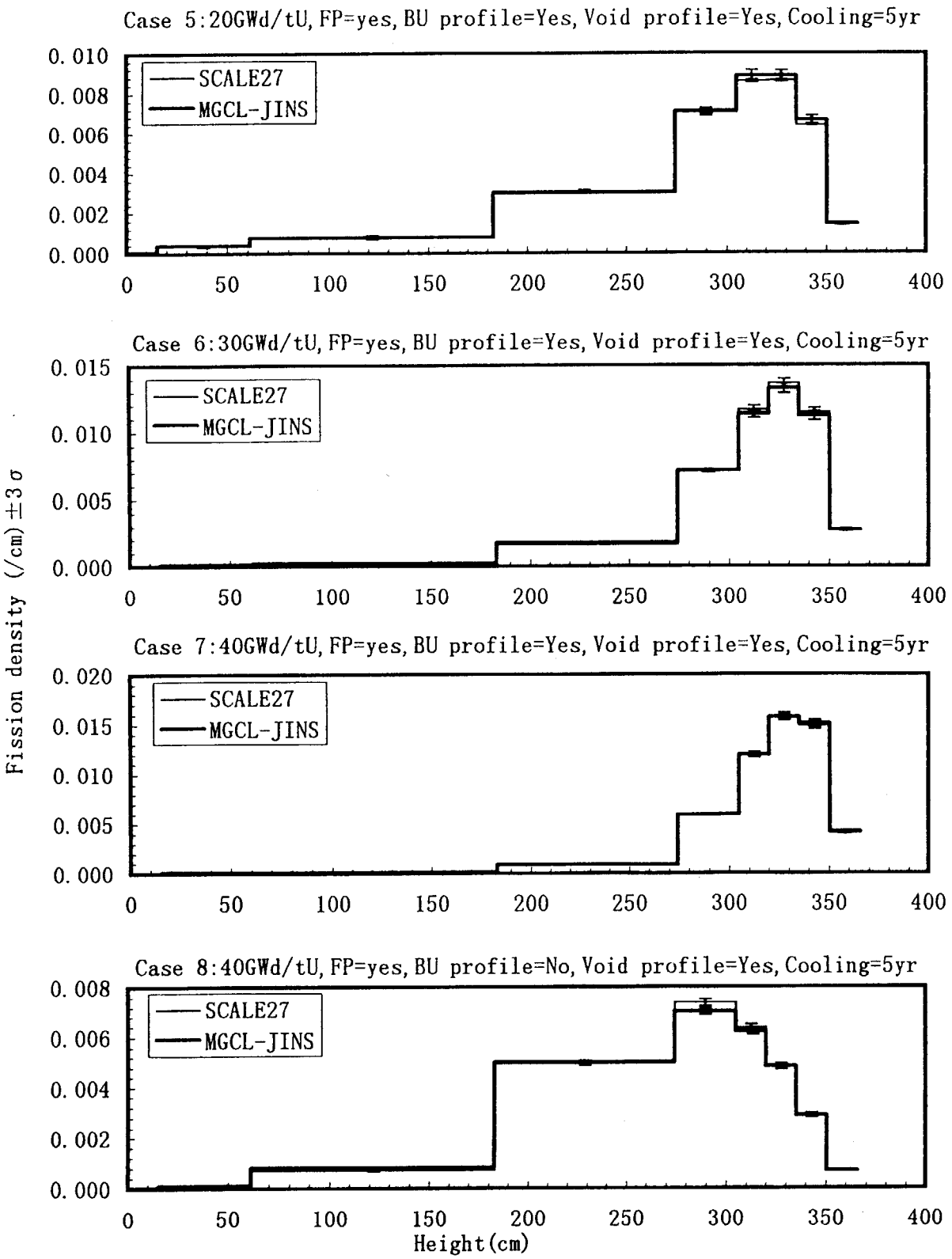


Fig. 1 Axial distribution of normalized fission densities (2/6)

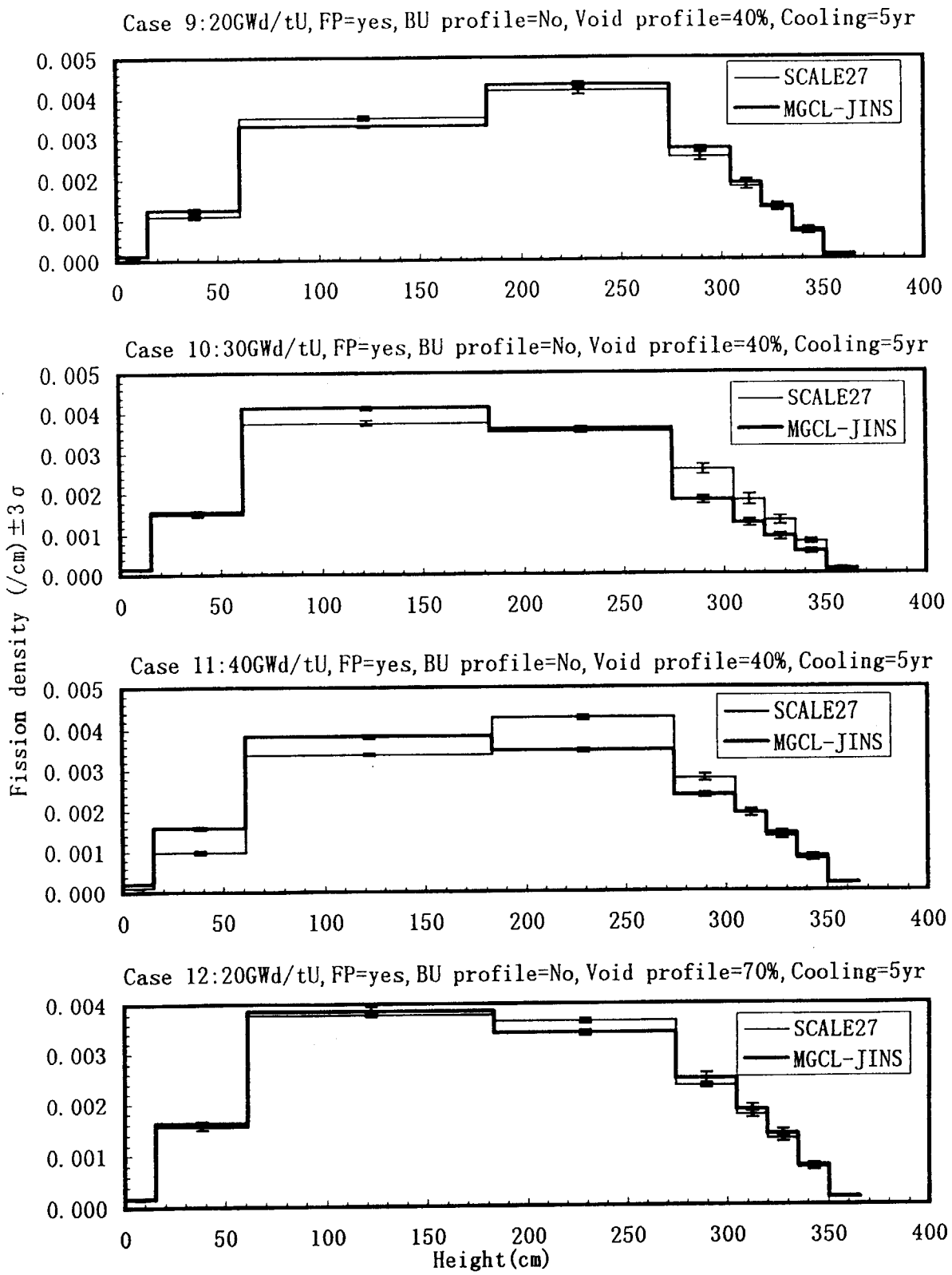


Fig. 1 Axial distribution of normalized fission densities (3/6)

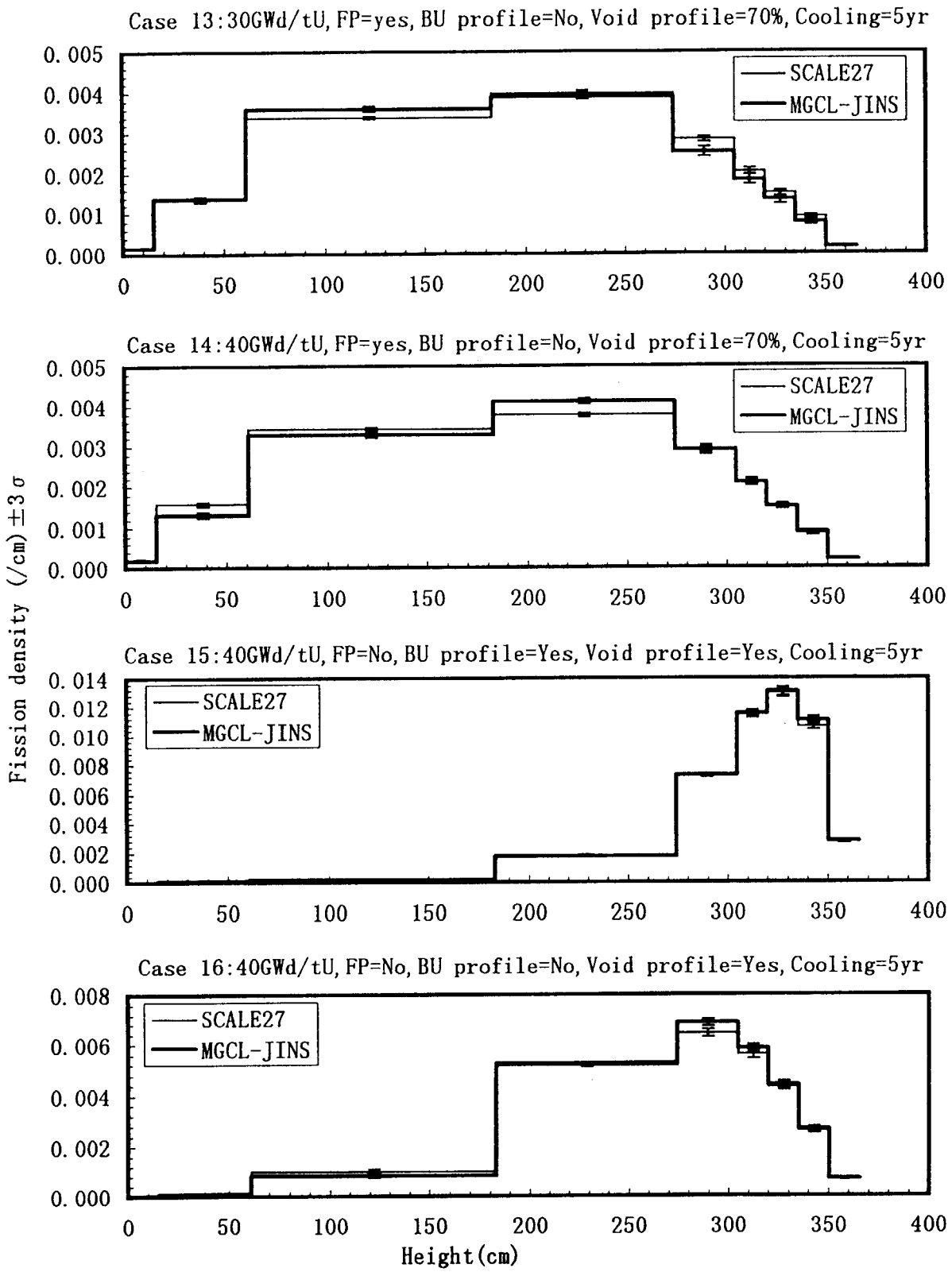


Fig. 1 Axial distribution of normalized fission densities (4/6)

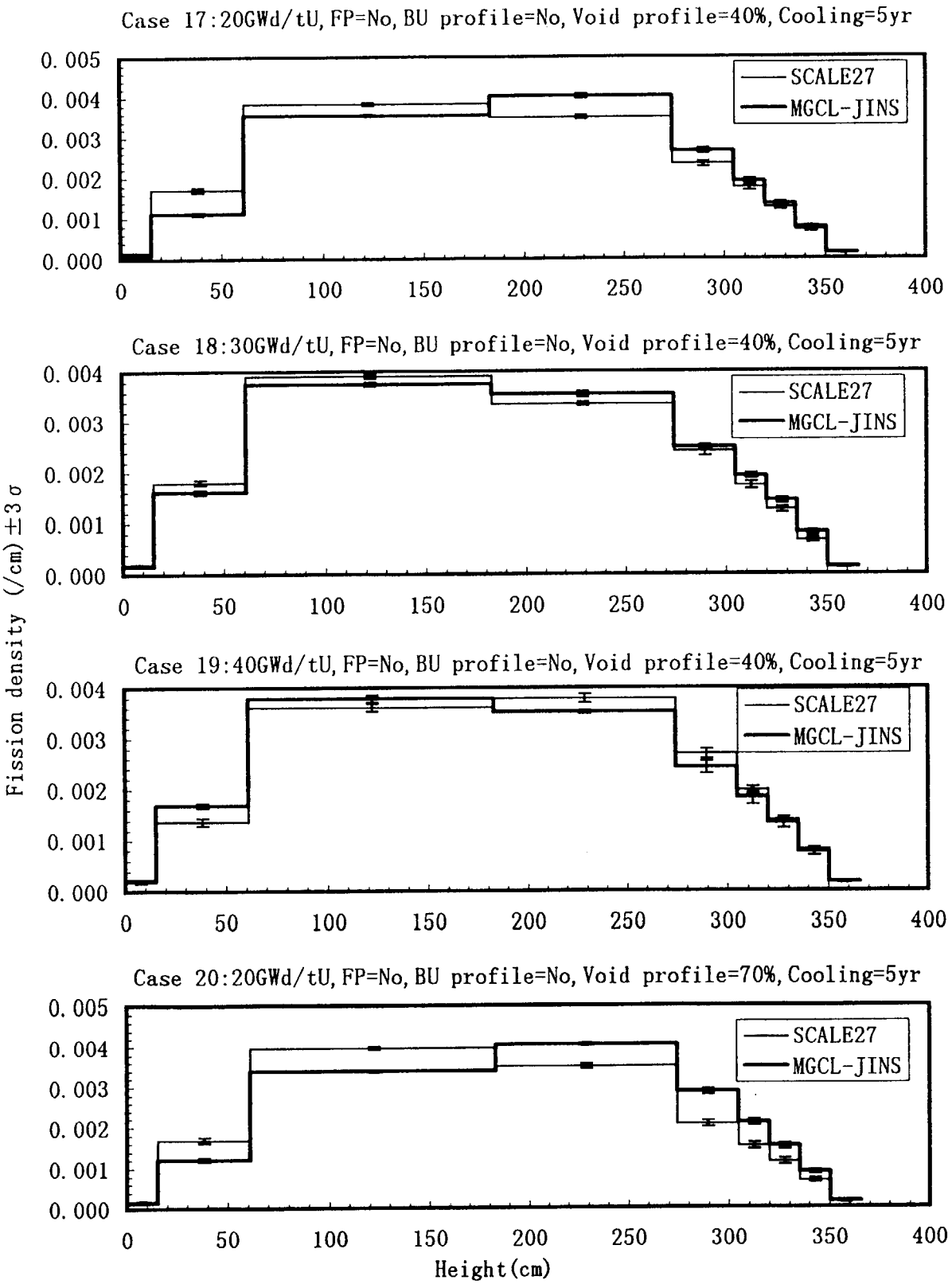


Fig. 1 Axial distribution of normalized fission densities (5/6)

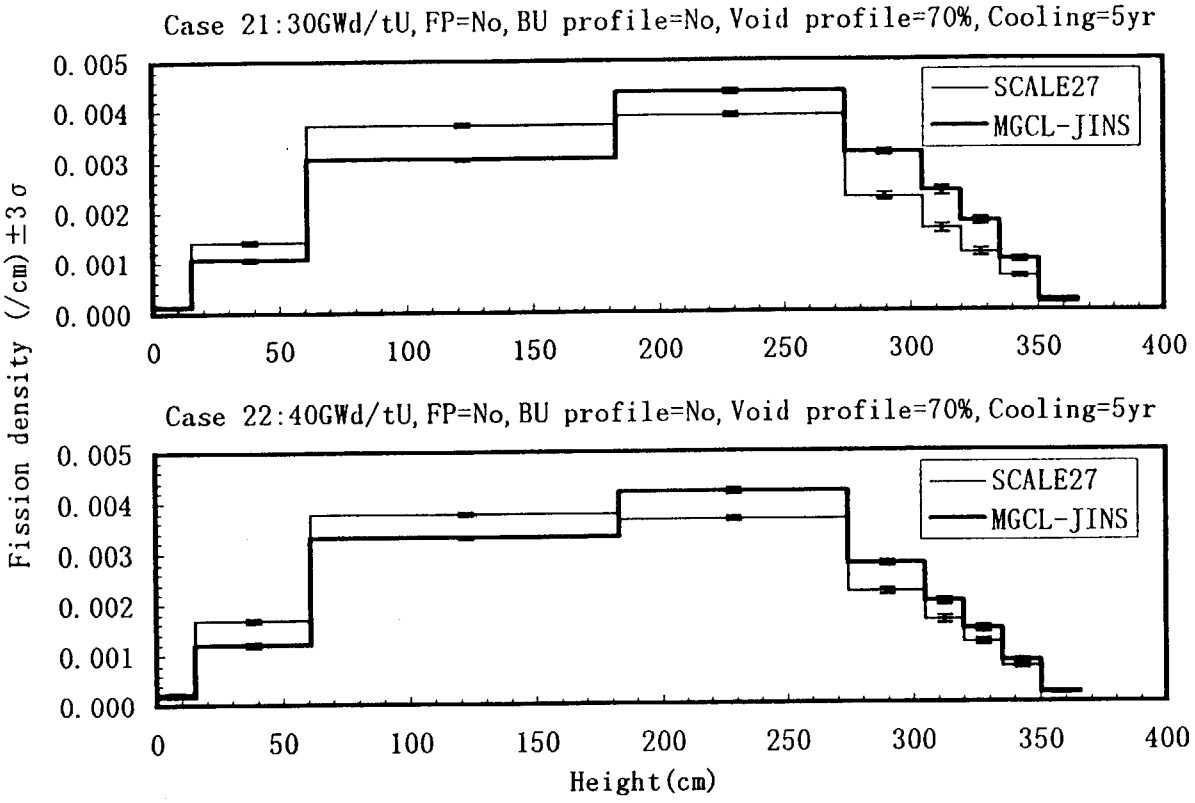


Fig. 1 Axial distribution of normalized fission densities (6/6)

Appendix IV.3

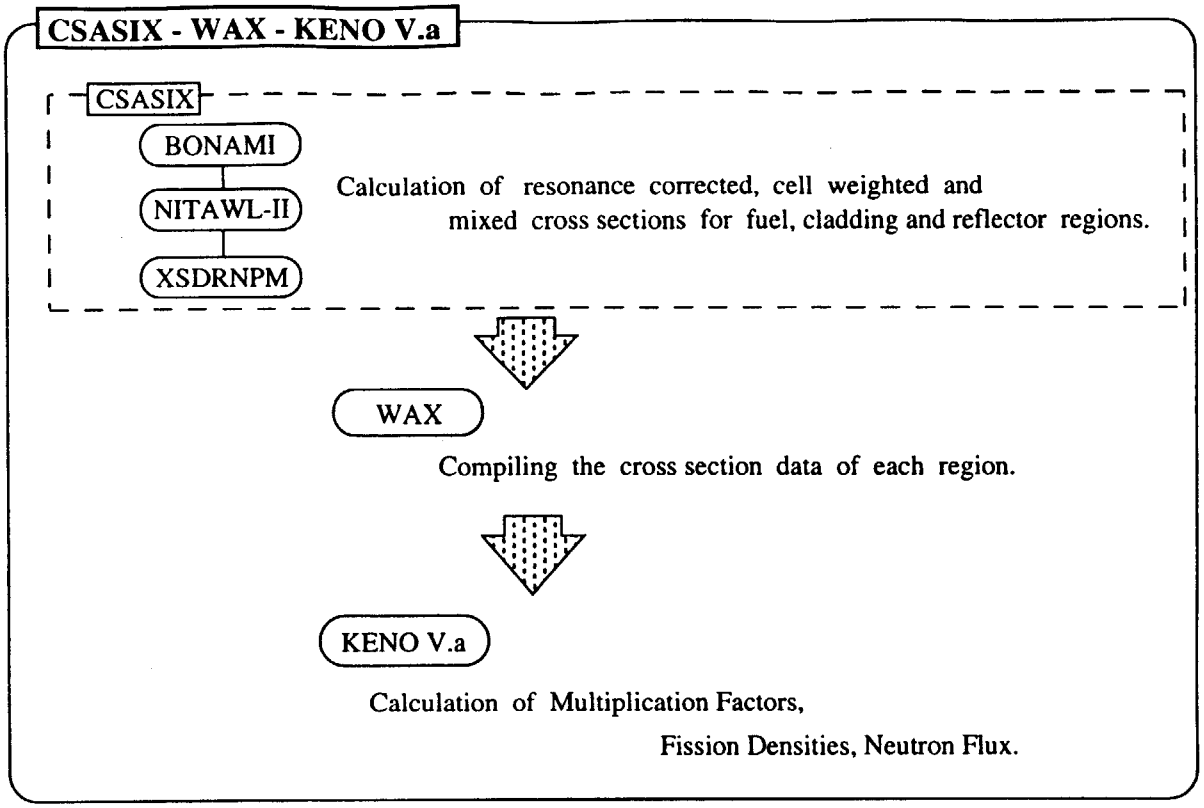
Result of Burnup Credit Criticality Benchmark Phase IIIA
— Cross Comparison of SCALE 4 —

[PNC]

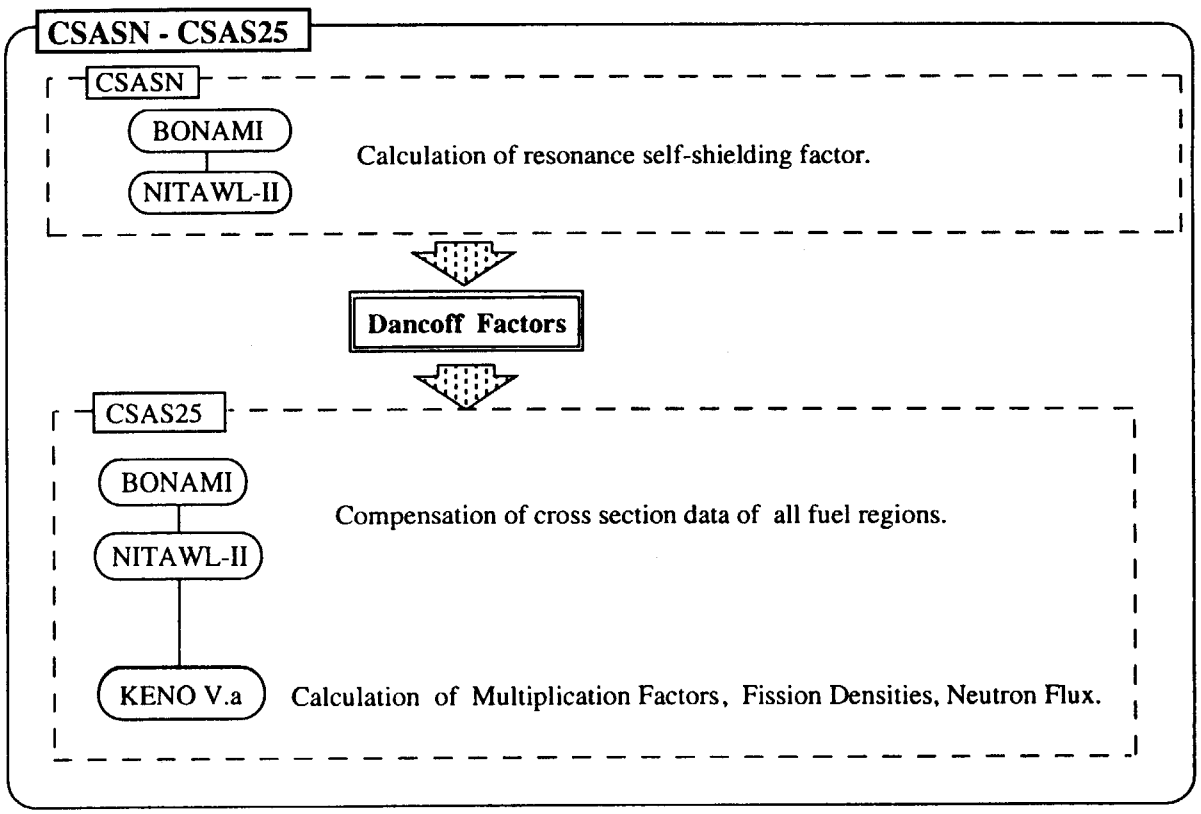
Result of Burnup Credit Criticality Benchmark Phase IIIA

- Cross Comparison of SCALE 4 -

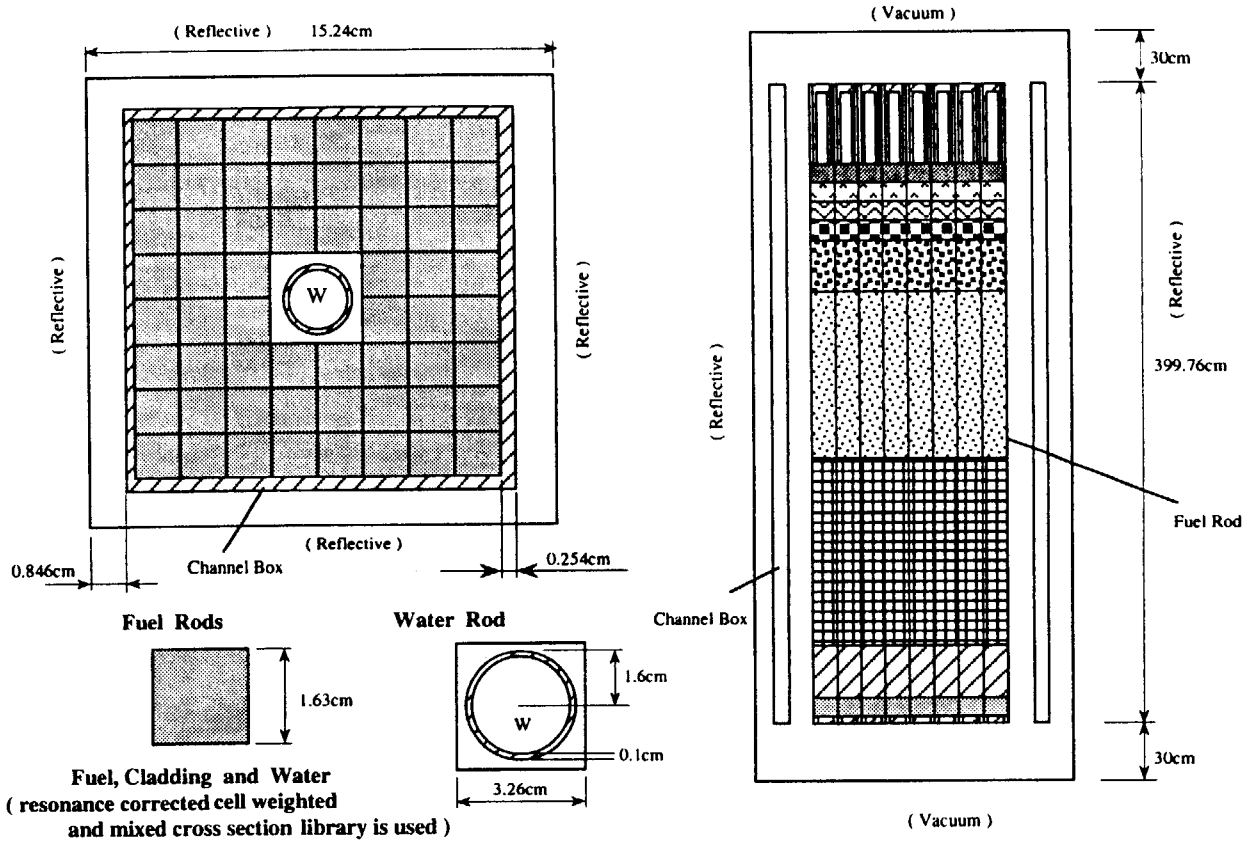
PNC Tokai Works
Ichiro Nojiri, Yasuhiro Fukasaku



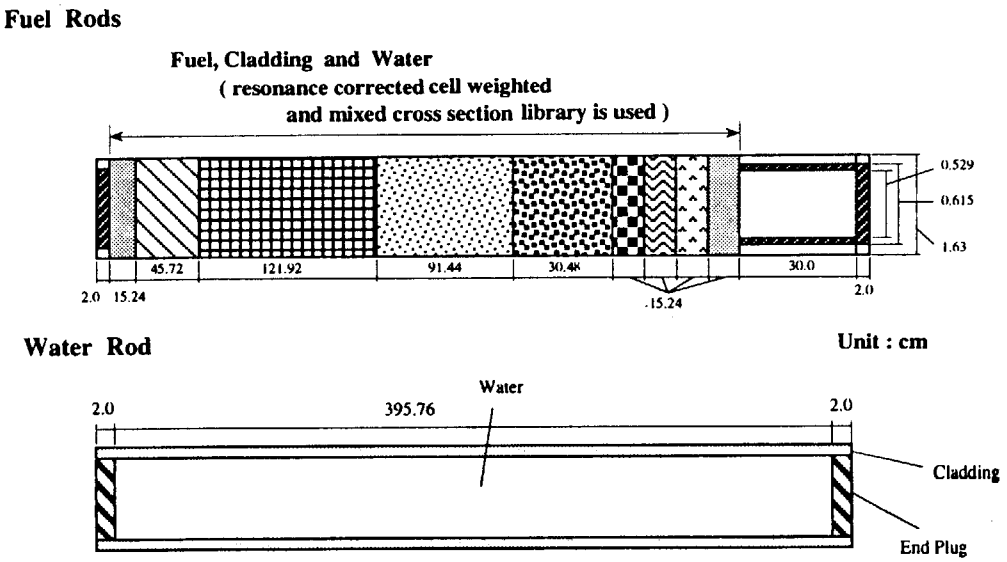
Calculation flow of CSASIX - WAX - KENO V.a Module



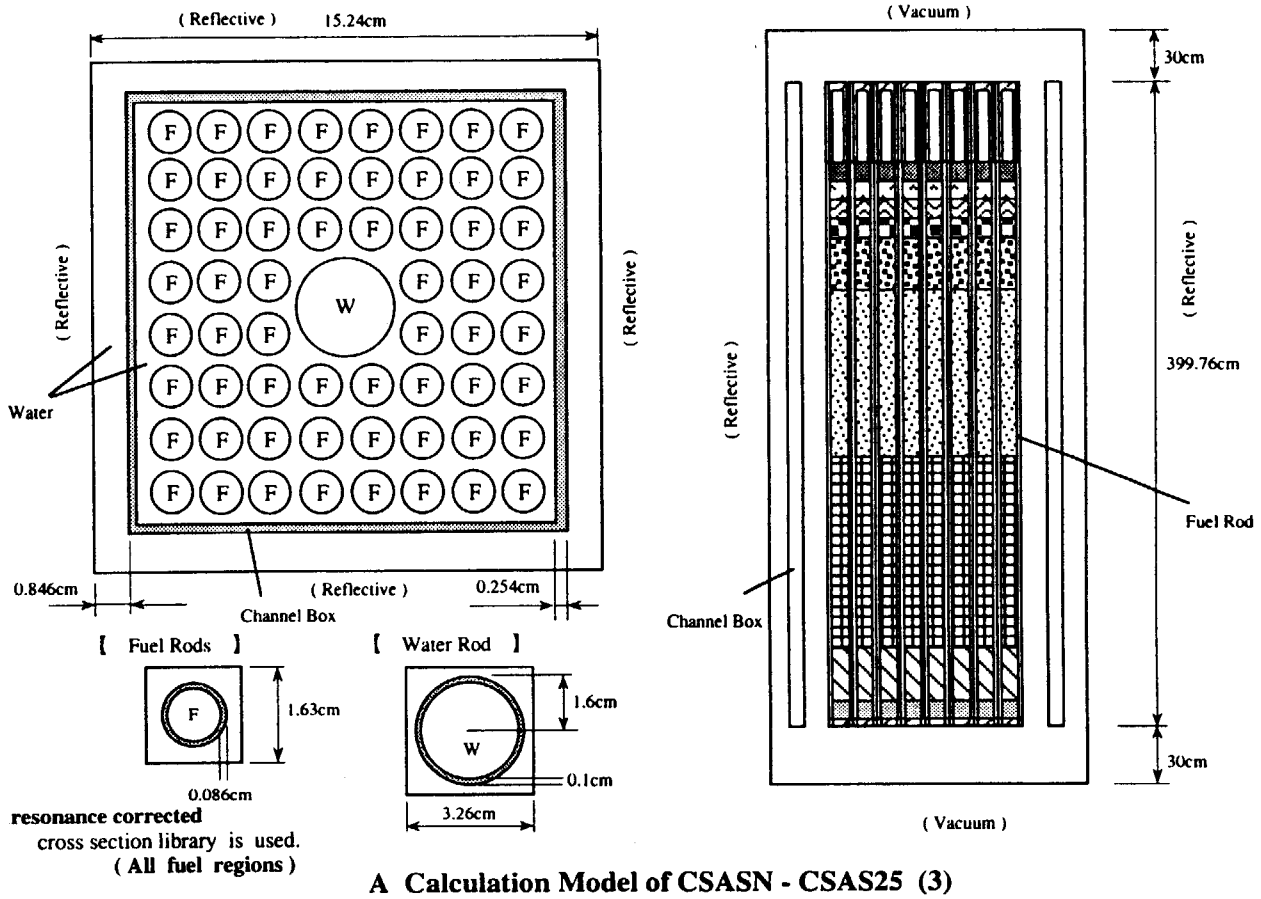
Calculation flow of CSASN - CSAS25 Module



A Calculation Model of CSASIX - WAX - KENO V.a (1)

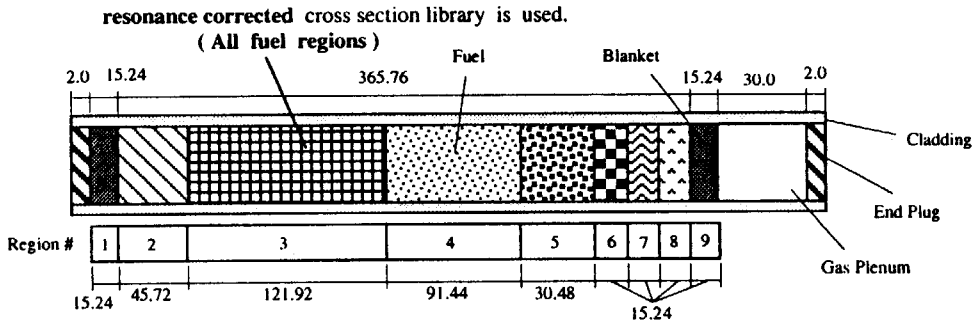


A Calculation Model of CSASIX - WAX - KENO V.a (2)

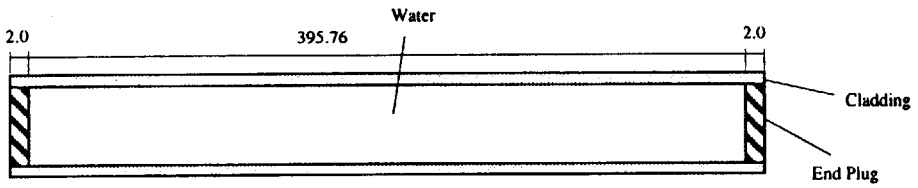


A Calculation Model of CSASN - CSAS25 (3)

Fuel Rods



Water Rod



A Calculation Model of CSASN - CSAS25 (4)

OECD/NEA Criticality Benchmark (Phase-IIIA) Calculation Results

Case No.	Calculation Parameter					CSASIX-WAX-KENO V.a		CSASN - CSAS25									
	C.T*1 (yr)	F.P.s	B.P*2	V.P*3	Burnup (GWd/tU)	27BURNUPLIB		27BURNUPLIB				27GROUPNDF4				44GROUPNDF5	
						SCALE 4.2		SCALE 4.2		SCALE 4.3		SCALE 4.2		SCALE 4.3		SCALE 4.3	
						K_{eff}	σ	K_{eff}	σ	K_{eff}	σ	K_{eff}	σ	K_{eff}	σ	K_{eff}	σ
1	Fresh Fuel					1.3979	0.0007	1.3898	0.0007	1.3910	0.0007	1.3900	0.0007	1.3914	0.0007	1.4050	0.0008
2					20	1.1919	0.0007	1.1936	0.0008	1.1958	0.0008					1.2044	0.0007
3	1	Yes	Yes	Yes	30	1.1049	0.0010	1.1088	0.0006	1.1135	0.0008					1.1203	0.0009
4					40	1.0247	0.0009	1.0258	0.0009	1.0298	0.0009					1.0363	0.0010
5					20	1.1759	0.0007	1.1830	0.0007	1.1829	0.0007					1.1922	0.0007
6		Yes	Yes	Yes	30	1.0881	0.0009	1.0901	0.0009	1.0901	0.0009					1.0984	0.0008
7					40	0.9956	0.0009	1.0007	0.0008	1.0007	0.0009					1.0068	0.0009
8		Yes	No	Yes	40	0.9870	0.0006	0.9922	0.0006	0.9944	0.0007					1.0002	0.0006
9					20	1.1843	0.0006	1.1896	0.0007	1.1878	0.0007					1.1981	0.0007
10		Yes	No	40%	30	1.0689	0.0006	1.0740	0.0007	1.0766	0.0007					1.0812	0.0006
11					40	0.9579	0.0006	0.9635	0.0006	0.9654	0.0006					0.9691	0.0006
12					20	1.1947	0.0006	1.1983	0.0007	1.2004	0.0006					1.2097	0.0006
13	5	Yes	No	70%	30	1.0929	0.0006	1.0992	0.0007	1.1004	0.0006					1.1054	0.0006
14					40	0.9975	0.0005	1.0032	0.0007	1.0041	0.0006					1.0104	0.0006
15		No	Yes	Yes	40	1.0989	0.0008	1.1022	0.0008	1.1047	0.0008	1.1028	0.0009	1.1029	0.0008	1.1100	0.0008
16		No	No	Yes	40	1.1029	0.0006	1.1094	0.0007	1.1093	0.0008	1.1096	0.0007	1.1093	0.0007	1.1131	0.0007
17					20	1.2539	0.0007	1.2584	0.0007	1.2583	0.0007	1.2583	0.0007	1.2589	0.0007	1.2694	0.0007
18		No	No	40%	30	1.1616	0.0006	1.1689	0.0007	1.1703	0.0008	1.1693	0.0007	1.1713	0.0007	1.1758	0.0007
19					40	1.0705	0.0006	1.0782	0.0007	1.0783	0.0007	1.0780	0.0006	1.0790	0.0007	1.0817	0.0006
20					20	1.2662	0.0006	1.2693	0.0007	1.2709	0.0007	1.2708	0.0007	1.2724	0.0007	1.2826	0.0007
21		No	No	70%	30	1.1900	0.0006	1.1949	0.0007	1.1957	0.0007	1.1941	0.0007	1.1949	0.0007	1.2025	0.0007
22					40	1.1128	0.0006	1.1184	0.0007	1.1196	0.0006	1.1195	0.0007	1.1208	0.0008	1.1253	0.0007

*1 C.T : Cooling Times
*2 B.P : Burnup Profile
*3 V.P : Void Profile
40% and 70% uniform void cases are considered as "Void Profile = NO".

Appendix V

Document on Phase IIIA Benchmarks Presented at 1997 Meeting in Paris

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Appendix V.1

Analyses of All Combination of Parameter Values

[JAERI & Toshiba]

OECD/NEA/NSC/BUC WG

Analyses of All Combination of Parameter Values

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July 9, 1997

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1 Introduction

In OECD/NEA/NSC/BUC benchmark "PHASE3A", effects of several parameters about spent BWR assembly for neutron multiplication factor are analyzed through twenty two cases. However, ninety seven cases were considered in that draft. This report presents the results of analysis of ninety seven cases (including dis-selected cases) for supplement.

2 Problem Specification

Table 2.1 shows a relation between parameters and case numbers. Numbers of cases selected in PHASE3A are indicated in parentheses.

Table 2.1: A List of Parameters and Case Numbers

Cooling Time	FPs	Burnup Profile	Void Profile	FRESH	10GWD/T	20 GWD/T	30 GWD/T	40 GWD/T
1 Year	YES	YES	YES	1(1)	2	3(2)	4(3)	5(4)
			40% UNIF		6	7	8	9
			70% UNIF		10	11	12	13
	YES	NO	YES		14	15	16	17
			40% UNIF		18	19	20	21
			70% UINF		22	23	24	25
	NO	YES	YES		26	27	28	29
			40% UNIF		30	31	32	33
			70% UINF		34	35	36	37
	NO	NO	YES		38	39	40	41
			40% UNIF		42	43	44	45
			70% UINF		46	47	48	49
5 Years	YES	YES	YES		50	51(5)	52(6)	53(7)
			40% UNIF		54	55	56	57
			70% UINF		58	59	60	61
	YES	NO	YES		62	63	64	65(8)
			40% UNIF		66	67(9)	68(10)	69(11)
			70% UINF		70	71(12)	72(13)	73(14)
	NO	YES	YES		74	75	76	77(15)
			40% UNIF		78	79	80	81
			70% UINF		82	83	84	85
	NO	NO	YES		86	87	88	89(16)
			40% UNIF		90	91(17)	92(18)	93(19)
			70% UINF		94	95(20)	96(21)	97(22)

40% or 70% void cases are considered as void profile = NO
Numbers in ( ) are case number of PHASE3A

3 Codes and Library

Table 3.1 shows codes and libraries which were used in this calculation. Two deterministic codes and MCNP4A were used. Since MCNP4A is three dimensional monte carlo code using continuous energy cross section library, results of MCNP4A are used as reference.

Table 3.1: Codes and Libraries

Codes	Library	Energy Groups
SRAC/CITATION	JENDL-3.2	107/20*
MCNP4A	JENDL-3.2	Cont.
TGBLA/ALEX	ENDF/B-4,5	95/3

* It means 107 group for SRAC, and 20 Group for CITATION

4 Results

In Tables 4.1, 4.2 ,4.3, infinite neutron multiplication factors (k_{inf}) by each code are shown. Tables 4.4 and 4.5 present ratios of k_{inf} to MCNP4A. These tables reveal that the averaged deference is 0.3 % between MCNP4A and SRAC/CITATION, and 1.3 % between MCNP4A and TGBLA/ALEX.

4.1 Effect of Decay

Differences of k_{inf} (%Δk) between one year and five year decay period are shown in Table 4.6. Larger decrease of reactivity is observed when FP is included and in higher burnup.

4.2 Burnup Profile

Difference of k_{inf} (%Δk) between cases with and without axial burnup profile (end effect)are shown in Table 4.7 and Figure 4.1.

End effect is enhanced when FP are considered and in higher burnup like case of PWR[4].

4.3 Void Profile

Differences of k_{inf} (%Δk) between cases with and without considering void profile are shown in Table 4.8.

Whether considering void profile or not has larger effect than burnup profile. Especially for 40 % uniform void, about 4 % difference are observed for the case of 40

[4]PHASE2A

GWd/tHM. Cases of 70 % constant void have smaller effect than the cases of 40 % uniform void.

Table 4.1: K_{inf} Values (MCNP4A)

Cooling	FP	Burnup	Void	FRESH	10GWd/t	20GWd/t	30GWd/t	40GWd/t
1	YES	YES	YES	1.4082	1.2905	1.1940	1.1115	1.0263
1	YES	YES	40 UNIF		1.2874	1.1823	1.0871	0.9876
1	YES	YES	70 UNIF		1.2925	1.1947	1.1081	1.0205
1	YES	NO	YES		1.3025	1.2061	1.1111	1.0234
1	YES	NO	40 UNIF		1.3006	1.1994	1.0939	0.9911
1	YES	NO	70 UNIF		1.3034	1.2114	1.1191	1.0323
1	NO	YES	YES		1.3380	1.2613	1.1900	1.1180
1	NO	YES	40 UNIF		1.3353	1.2490	1.1636	1.0749
1	NO	YES	70 UNIF		1.3407	1.2628	1.1881	1.1135
1	NO	NO	YES		1.3466	1.2737	1.1973	1.1230
1	NO	NO	40 UNIF		1.3437	1.2646	1.1789	1.0901
1	NO	NO	70 UNIF		1.3488	1.2786	1.2053	1.1332
5	YES	YES	YES		1.2846	1.1836	1.0908	0.9988
5	YES	YES	40 UNIF		1.2823	1.1706	1.0684	0.9625
5	YES	YES	70 UNIF		1.2863	1.1818	1.0868	0.9935
5	YES	NO	YES		1.2974	1.1937	1.0876	0.9888
5	YES	NO	40 UNIF		1.2954	1.1872	1.0710	0.9582
5	YES	NO	70 UNIF		1.2987	1.1988	1.0965	0.9982
5	NO	YES	YES		1.3354	1.2549	1.1776	1.1005
5	NO	YES	40 UNIF		1.3320	1.2415	1.1527	1.0598
5	NO	YES	70 UNIF		1.3375	1.2543	1.1751	1.0946
5	NO	NO	YES		1.3453	1.2667	1.1850	1.1050
5	NO	NO	40 UNIF		1.3426	1.2584	1.1661	1.0724
5	NO	NO	70 UNIF		1.3467	1.2716	1.1922	1.1142

Table 4.2: K_{inf} Values (SRAC/CITATION)

Cooling	FP	Burnup	Void	FRESH	10GWd/t	20GWd/t	30GWd/t	40GWd/t
1	YES	YES	YES	1.3963	1.2844	1.1928	1.1091	1.0250
1	YES	YES	40 UNIF		1.2811	1.1800	1.0851	0.9867
1	YES	YES	70 UNIF		1.2853	1.1916	1.1052	1.0203
1	YES	NO	YES		1.2953	1.2033	1.1103	1.0228
1	YES	NO	40 UNIF		1.2932	1.1957	1.0919	0.9902
1	YES	NO	70 UNIF		1.2970	1.2076	1.1171	1.0315
1	NO	YES	YES		1.3317	1.2583	1.1883	1.1155
1	NO	YES	40 UNIF		1.3273	1.2440	1.1604	1.0720
1	NO	YES	70 UNIF		1.3332	1.2587	1.1852	1.1095
1	NO	NO	YES		1.3393	1.2694	1.1951	1.1220
1	NO	NO	40 UNIF		1.3361	1.2605	1.1754	1.0881
1	NO	NO	70 UNIF		1.3415	1.2737	1.2017	1.1309
5	YES	YES	YES		1.2790	1.1802	1.0886	0.9963
5	YES	YES	40 UNIF		1.2762	1.1678	1.0664	0.9608
5	YES	YES	70 UNIF		1.2798	1.1786	1.0851	0.9918
5	YES	NO	YES		1.2906	1.1906	1.0870	0.9873
5	YES	NO	40 UNIF		1.2889	1.1822	1.0695	0.9567
5	YES	NO	70 UNIF		1.2919	1.1944	1.0933	0.9966
5	NO	YES	YES		1.3291	1.2509	1.1759	1.0983
5	NO	YES	40 UNIF		1.3251	1.2369	1.1496	1.0555
5	NO	YES	70 UNIF		1.3304	1.2507	1.1723	1.0926
5	NO	NO	YES		1.3372	1.2623	1.1821	1.1020
5	NO	NO	40 UNIF		1.3342	1.2523	1.1630	1.0695
5	NO	NO	70 UNIF		1.3391	1.2663	1.1885	1.1116

Table 4.3: K_{inf} Value (TGBLA/ALEX)

Cooling	FP	Burnup	Void	FRESH	10GWd/t	20GWd/t	30GWd/t	40GWd/t
1	YES	YES	YES	1.3806	1.2706	1.1815	1.1008	1.0197
1	YES	YES	40 UNIF		1.2672	1.1687	1.0771	0.9824
1	YES	YES	70 UNIF		1.2714	1.1805	1.0977	1.0155
1	YES	NO	YES		1.2813	1.1904	1.0999	1.0158
1	YES	NO	40 UNIF		1.2793	1.1827	1.0816	0.9838
1	YES	NO	70 UNIF		1.2828	1.1947	1.1067	1.0245
1	NO	YES	YES		1.3156	1.2429	1.1749	1.1044
1	NO	YES	40 UNIF		1.3113	1.2289	1.1491	1.0639
1	NO	YES	70 UNIF		1.3171	1.2430	1.1725	1.1005
1	NO	NO	YES		1.3233	1.2530	1.1798	1.1090
1	NO	NO	40 UNIF		1.3204	1.2444	1.1608	1.0759
1	NO	NO	70 UNIF		1.3253	1.2574	1.1866	1.1176
5	YES	YES	YES		1.2654	1.1694	1.0810	0.9924
5	YES	YES	40 UNIF		1.2623	1.1577	1.0591	0.9575
5	YES	YES	70 UNIF		1.2660	1.1683	1.0780	0.9882
5	YES	NO	YES		1.2767	1.1779	1.0774	0.9827
5	YES	NO	40 UNIF		1.2749	1.1710	1.0602	0.9517
5	YES	NO	70 UNIF		1.2780	1.1820	1.0840	0.9911
5	NO	YES	YES		1.3129	1.2353	1.1622	1.0872
5	NO	YES	40 UNIF		1.3088	1.2218	1.1374	1.0480
5	NO	YES	70 UNIF		1.3142	1.2349	1.1596	1.0831
5	NO	NO	YES		1.3210	1.2455	1.1664	1.0897
5	NO	NO	40 UNIF		1.3183	1.2374	1.1479	1.0571
5	NO	NO	70 UNIF		1.3230	1.2499	1.1731	1.0982

Table 4.4: Ratio to MCNP4A (SRAC/CITATION)

Cooling	FP	Burnup	Void	10GWd/t	20GWd/t	30GWd/t	40GWd/t
1	YES	YES	YES	0.995	0.999	0.998	0.999
1	YES	YES	40 UNIF	0.995	0.998	0.998	0.999
1	YES	YES	70 UNIF	0.994	0.997	0.997	1.000
1	YES	NO	YES	0.994	0.998	0.999	0.999
1	YES	NO	40 UNIF	0.994	0.997	0.998	0.999
1	YES	NO	70 UNIF	0.995	0.997	0.998	0.999
1	NO	YES	YES	0.995	0.998	0.999	0.998
1	NO	YES	40 UNIF	0.994	0.996	0.997	0.997
1	NO	YES	70 UNIF	0.994	0.997	0.998	0.996
1	NO	NO	YES	0.995	0.997	0.998	0.999
1	NO	NO	40 UNIF	0.994	0.997	0.997	0.998
1	NO	NO	70 UNIF	0.995	0.996	0.997	0.998
5	YES	YES	YES	0.996	0.997	0.998	0.998
5	YES	YES	40 UNIF	0.995	0.998	0.998	0.998
5	YES	YES	70 UNIF	0.995	0.997	0.998	0.998
5	YES	NO	YES	0.995	0.997	0.999	0.999
5	YES	NO	40 UNIF	0.995	0.996	0.999	0.998
5	YES	NO	70 UNIF	0.995	0.996	0.997	0.998
5	NO	YES	YES	0.995	0.997	0.999	0.998
5	NO	YES	40 UNIF	0.995	0.996	0.997	0.996
5	NO	YES	70 UNIF	0.995	0.997	0.998	0.998
5	NO	NO	YES	0.994	0.997	0.998	0.997
5	NO	NO	40 UNIF	0.994	0.995	0.997	0.997
5	NO	NO	70 UNIF	0.994	0.996	0.997	0.998

Table 4.5: Ratio to MCNP4A (TGBLA/ALEX)

Cooling	FP	Burnup	Void	10GWd/t	20GWd/t	30GWd/t	40GWd/t
1	YES	YES	YES	0.985	0.990	0.990	0.994
1	YES	YES	40 UNIF	0.984	0.989	0.991	0.995
1	YES	YES	70 UNIF	0.984	0.988	0.991	0.995
1	YES	NO	YES	0.984	0.987	0.990	0.993
1	YES	NO	40 UNIF	0.984	0.986	0.989	0.993
1	YES	NO	70 UNIF	0.984	0.986	0.989	0.992
1	NO	YES	YES	0.983	0.985	0.987	0.988
1	NO	YES	40 UNIF	0.982	0.984	0.988	0.990
1	NO	YES	70 UNIF	0.982	0.984	0.987	0.988
1	NO	NO	YES	0.983	0.984	0.985	0.987
1	NO	NO	40 UNIF	0.983	0.984	0.985	0.987
1	NO	NO	70 UNIF	0.983	0.983	0.984	0.986
5	YES	YES	YES	0.985	0.988	0.991	0.994
5	YES	YES	40 UNIF	0.984	0.989	0.991	0.995
5	YES	YES	70 UNIF	0.984	0.989	0.992	0.995
5	YES	NO	YES	0.984	0.987	0.991	0.994
5	YES	NO	40 UNIF	0.984	0.986	0.990	0.993
5	YES	NO	70 UNIF	0.984	0.986	0.989	0.993
5	NO	YES	YES	0.983	0.984	0.987	0.988
5	NO	YES	40 UNIF	0.983	0.984	0.987	0.989
5	NO	YES	70 UNIF	0.983	0.985	0.987	0.989
5	NO	NO	YES	0.982	0.983	0.984	0.986
5	NO	NO	40 UNIF	0.982	0.983	0.984	0.986
5	NO	NO	70 UNIF	0.982	0.983	0.984	0.986

Table 4.6: %Δk $(1Year - 5Year) \times 100$ (MCNP4A)

FP	Burnup	Void	10GWd/t	20GWd/t	30GWd/t	40GWd/t
YES	YES	YES	0.6	1.0	2.1	2.8
YES	YES	40 UNIF	0.5	1.2	1.9	2.5
YES	YES	70 UNIF	0.6	1.3	2.1	2.7
YES	NO	YES	0.5	1.2	2.4	3.5
YES	NO	40 UNIF	0.5	1.2	2.3	3.3
YES	NO	70 UNIF	0.5	1.3	2.3	3.4
NO	YES	YES	0.3	0.6	1.2	1.8
NO	YES	40 UNIF	0.3	0.8	1.1	1.5
NO	YES	70 UNIF	0.3	0.8	1.3	1.9
NO	NO	YES	0.1	0.7	1.2	1.8
NO	NO	40 UNIF	0.1	0.6	1.3	1.8
NO	NO	70 UNIF	0.2	0.7	1.3	1.9

Table 4.7: %Δk Burnup $(Profile(Yes) - Profile(No)) \times 100$ (MCNP4A)

Cooling	FP	Void	10GWd/t	20GWd/t	30GWd/t	40GWd/t
1	YES	YES	-1.2	-1.2	0.0	0.3
1	YES	40 UNIF	-1.3	-1.7	-0.7	-0.4
1	YES	70 UNIF	-1.1	-1.7	-1.1	-1.2
1	NO	YES	-0.9	-1.2	-0.7	-0.5
1	NO	40 UNIF	-0.8	-1.6	-1.5	-1.5
1	NO	70 UNIF	-0.8	-1.6	-1.7	-2.0
5	YES	YES	-1.3	-1.0	0.3	1.0
5	YES	40 UNIF	-1.3	-1.7	-0.3	0.4
5	YES	70 UNIF	-1.2	-1.7	-1.0	-0.5
5	NO	YES	-1.0	-1.2	-0.7	-0.5
5	NO	40 UNIF	-1.1	-1.7	-1.3	-1.3
5	NO	70 UNIF	-0.9	-1.7	-1.7	-2.0

Cooling(y)	FP	VOID	MARK	Cooling(y)	FP	VOID	MARK
1	YES	YES	●	5	YES	YES	△
1	YES	40U	⊙	5	YES	40U	▲
1	YES	70U	□	5	YES	70U	⧗
1	NO	YES	▣	5	NO	YES	▽
1	NO	40U	◇	5	NO	40U	▼
1	NO	70U	◆	5	NO	70U	⋈

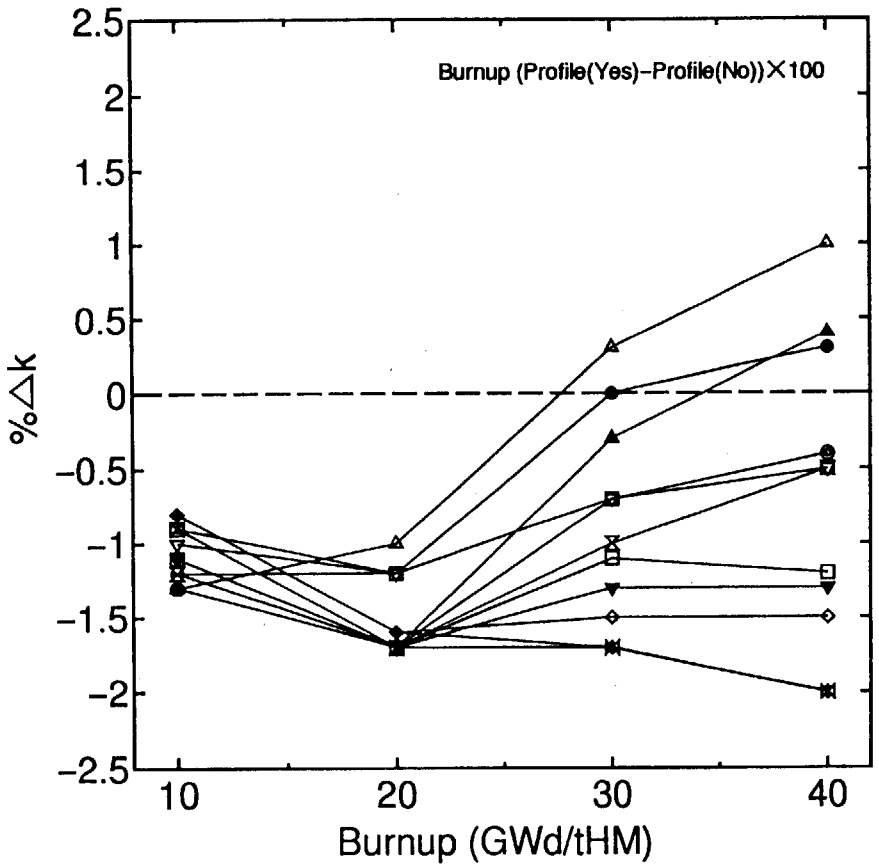

Figure 4.1: End effect Calculated using MCNP4A

Table 4.8: %Δk Void $(Profile(Yes) - Profile(No)) \times 100$ (MCNP4A)

Cooling	FP	Burnup	Void	10GWd/t	20GWd/t	30GWd/t	40GWd/t
1	YES	YES	40 UNIF	0.3	1.2	2.4	3.9
1	YES	YES	70 UNIF	-0.2	-0.1	0.3	0.6
1	YES	NO	40 UNIF	0.2	0.7	1.7	3.2
1	YES	NO	70 UNIF	-0.1	-0.5	-0.8	-0.9
1	NO	YES	40 UNIF	0.3	1.2	2.6	4.3
1	NO	YES	70 UNIF	-0.3	-0.1	0.2	0.5
1	NO	NO	40 UNIF	0.3	0.9	1.8	3.3
1	NO	NO	70 UNIF	-0.2	-0.5	-0.8	-1.0
5	YES	YES	40 UNIF	0.2	1.3	2.2	3.6
5	YES	YES	70 UNIF	-0.2	0.2	0.4	0.5
5	YES	NO	40 UNIF	0.2	0.7	1.7	3.1
5	YES	NO	70 UNIF	-0.1	-0.5	-0.9	-0.9
5	NO	YES	40 UNIF	0.3	1.3	2.5	4.1
5	NO	YES	70 UNIF	-0.2	0.1	0.3	0.6
5	NO	NO	40 UNIF	0.3	0.8	1.9	3.3
5	NO	NO	70 UNIF	-0.1	-0.5	-0.7	-0.9

Appendix VI

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国際単位系（SI）と換算表

表1　SI基本単位および補助単位

量	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd
平面角	ラジアン	rad
立体角	ステラジアン	sr

表2　SIと併用される単位

名称	記号
分，時，日	min, h, d
度，分，秒	°, ′, ″
リットル	l, L
トン	t
電子ボルト	eV
原子質量単位	u

1 eV=1.60218×10^{-19}J

1 u=1.66054×10^{-27}kg

表5　SI接頭語

倍数	接頭語	記号
10^{18}	エクサ	E
10^{15}	ペタ	P
10^{12}	テラ	T
10^{9}	ギガ	G
10^{6}	メガ	M
10^{3}	キロ	k
10^{2}	ヘクト	h
10^{1}	デカ	da
10^{-1}	デシ	d
10^{-2}	センチ	c
10^{-3}	ミリ	m
10^{-6}	マイクロ	μ
10^{-9}	ナノ	n
10^{-12}	ピコ	p
10^{-15}	フェムト	f
10^{-18}	アト	a

表3　固有の名称をもつSI組立単位

量	名称	記号	他のSI単位による表現
周波数	ヘルツ	Hz	s^{-1}
力	ニュートン	N	m·kg/s^2
圧力，応力	パスカル	Pa	N/m^2
エネルギー,仕事,熱量	ジュール	J	N·m
工率，放射束	ワット	W	J/s
電気量，電荷	クーロン	C	A·s
電位，電圧，起電力	ボルト	V	W/A
静電容量	ファラド	F	C/V
電気抵抗	オーム	Ω	V/A
コンダクタンス	ジーメンス	S	A/V
磁束	ウェーバ	Wb	V·s
磁束密度	テスラ	T	Wb/m^2
インダクタンス	ヘンリー	H	Wb/A
セルシウス温度	セルシウス度	℃	
光束	ルーメン	lm	cd·sr
照度	ルクス	lx	lm/m^2
放射能	ベクレル	Bq	s^{-1}
吸収線量	グレイ	Gy	J/kg
線量等量	シーベルト	Sv	J/kg

表4　SIと共に暫定的に維持される単位

名称	記号
オングストローム	Å
バーン	b
バール	bar
ガル	Gal
キュリー	Ci
レントゲン	R
ラド	rad
レム	rem

1 Å=0.1nm=10^{-10}m

1 b=100fm^2=10^{-28}m^2

1 bar=0.1MPa=10^5Pa

1 Gal=1cm/s^2=10^{-2}m/s^2

1 Ci=3.7×10^{10}Bq

1 R=2.58×10^{-4}C/kg

1 rad=1cGy=10^{-2}Gy

1 rem=1cSv=10^{-2}Sv

(注)

1．表1－5は「国際単位系」第5版，国際度量衡局1985年刊行による。ただし，1 eVおよび1 uの値はCODATAの1986年推奨値によった。

2．表4には海里，ノット，アール，ヘクタールも含まれているが日常の単位なのでここでは省略した。

3．barは，JISでは流体の圧力を表わす場合に限り表2のカテゴリーに分類されている。

4．EC関係理事会指令ではbar，barnおよび「血圧の単位」mmHgを表2のカテゴリーに入れている。

換　算　表

力	N(=10^5dyn)	kgf	lbf
	1	0.101972	0.224809
	9.80665	1	2.20462
	4.44822	0.453592	1

粘度　1 Pa·s(N·s/m^2)=10 P(ポアズ)(g/(cm·s))

動粘度　1m^2/s=10^4St(ストークス)(cm^2/s)

圧力	MPa(=10bar)	kgf/cm^2	atm	mmHg(Torr)	lbf/in^2(psi)
	1	10.1972	9.86923	7.50062×10^3	145.038
	0.0980665	1	0.967841	735.559	14.2233
	0.101325	1.03323	1	760	14.6959
	1.33322×10^{-4}	1.35951×10^{-3}	1.31579×10^{-3}	1	1.93368×10^{-2}
	6.89476×10^{-3}	7.03070×10^{-2}	6.80460×10^{-2}	51.7149	1

エネルギー・仕事・熱量	J(=10^7 erg)	kgf·m	kW·h	cal(計量法)	Btu	ft·lbf	eV
	1	0.101972	2.77778×10^{-7}	0.238889	9.47813×10^{-4}	0.737562	6.24150×10^{18}
	9.80665	1	2.72407×10^{-6}	2.34270	9.29487×10^{-3}	7.23301	6.12082×10^{19}
	3.6×10^6	3.67098×10^5	1	8.59999×10^5	3412.13	2.65522×10^6	2.24694×10^{25}
	4.18605	0.426858	1.16279×10^{-6}	1	3.96759×10^{-3}	3.08747	2.61272×10^{19}
	1055.06	107.586	2.93072×10^{-4}	252.042	1	778.172	6.58515×10^{21}
	1.35582	0.138255	3.76616×10^{-7}	0.323890	1.28506×10^{-3}	1	8.46233×10^{18}
	1.60218×10^{-19}	1.63377×10^{-20}	4.45050×10^{-26}	3.82743×10^{-20}	1.51857×10^{-22}	1.18171×10^{-19}	1

1 cal= 4.18605J（計量法）

= 4.184J（熱化学）

= 4.1855J（15℃）

= 4.1868J（国際蒸気表）

仕事率　1 PS(仏馬力)

= 75 kgf·m/s

= 735.499W

放射能	Bq	Ci
	1	2.70270×10^{-11}
	3.7×10^{10}	1

吸収線量	Gy	rad
	1	100
	0.01	1

照射線量	C/kg	R
	1	3876
	2.58×10^{-4}	1

線量当量	Sv	rem
	1	100
	0.01	1

(86年12月26日現在)

OECD/NEA Burnup Credit Criticality Benchmarks Phase IIIA: Criticality Calculations of BWR Spent Fuel Assemblies in Storage and Transport