INTERNATIONAL ATOMIC ENERGY AGENCY
WAGRAMER STRASSE 5, A-1400 VIENNA, AUSTRIA
TELEPHONE: 43 1-2600, FACSIMILE: 43 1-26007
E-MAIL: official.mail@iaea.org

IAEA Co-ordinated Research Project
Small Reactors without On-Site Refuelling

FIRST YEAR FINAL REPORT

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Performance of benchmark analysis for Pb-Bi/Pb Cooled long-life cores of Small Reactors without On-Site Refuelling and optimization of their inherent/passive safety performance.

CHIEF SCIENTIFIC INVESTIGATOR
DR. ZAKI SU'UD

Bandung Institute of Technology
Indonesia

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Bandung, November 10, 2005

CHIEF SCIENTIFIC INVESTIGATOR ***CHERCHEUR PRINCIPAL:***	**HEAD OF INSTITUTE** ***LE DIRECTEUR DE L'INSTITUT:***
DR. ZAKI SU'UD	**Prof. Dr. Emmy Suparka** **Head of Research and Society Empowering Board, Bandung Institute of Technology (ITB)**
Signature / ***Signature*** **Date /** ***Date***	**Signature /** ***Signature*** **Date /** ***Date***

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CHAPTER I
INTRODUCTION

I.1 Background

Small and very small nuclear power plant with moderate economical aspect is an important candidate for electric power generation in many part of the third world countries including outside Java-Bali area in Indonesia. The nuclear energy system with the range of 5-50 Mwe match with the necessity and planning of many cities and provinces outside Java-Bali islands. In addition to electricity, desalination plant or cogeneration plant is a good candidate for nuclear energy application. Madura Island is a place where Indonesian government has planned to install desalination plant for clean water source. Due to the difference of the load between afternoon and night the use of fast reactors is a better choice due to capability to follow the load. Lead and lead bismuth cooled nuclear power reactors is now considered as potential candidate of next generation nuclear power reactors in the 21th centuries. Various versions of lead cooled nuclear power reactors have been analyzed and safety analysis also have been applied to them. The results are generally satisfactory[1-3].

One of important feature of lead/lead-bismuth cooled fast reactors is the zero burnup core capability which can eliminate possible super prompt critical accident and make possible of inherent safety feature based on reactivity feedback mechanism. The new design and safety approach however need high quality of system analysis as well as nuclear and material data to reduce calculation error so that its influence to the key design and safety parameters can be negligible. In the present research benchmarking will be performed using various calculation system and some experimental results. It is expected that the results can contribute to the achievement of the above goal.

I. 2. Objectives

The present research have the following objectives :

- **Benchmarking of long life core calculational method using various computer code**:
  Long life core without refueling may include some large core composition change during its life time. Similarly small and very small core also may need special treatment in neutronic calculation. Here we will investigate the static neutronic calculation including burnup calculation using various code to see how far existing codes and existing nuclear data give discrepancies on the main important characteristics of long live core. The calculation include criticality calculation, burnup calculation, power distribution, convertion ratio, Doppler coefficient calculation, coolant void coefficient calculation, axial fuel expansion coefficient, and radial core expansion coefficient.

- **Comparation with experimental results:** Such comparisons are important to see how much are the discrepancies between calculated results and their experimental values. The parameters to be compared includes criticality, reactivity swing during burnup, Doppler reactivity feedback, coolant void coefficient, fuel axial expansion coefficient, and core radial expansion coefficient.

- **Benchamarking of the treatment of lump FP cross section**: It is important to do benchmark calculation for lumped FP cross section data generation and investigate its impact on important static and dynamic parameters of long life fast reactors without on site refueling. This research aims to investigate existing lumped FP data especially the possibility to support the expected accuracy of non refueling long life fast reactors burnup analysis. This research expected to give recommendation on acceptable FP treatment for long life fast reactors without on site refueling.

- **Benchmarking of safety analysis :**
  Deterministic Safety analysis of lead-bismuth cooled long life fast reactors without on site refueling need coupling neutronic-thermal hydraulic analysis to simulate various accident condition from DBA to hypothetical severe accidents.

The analysis is very complex and includes many modeling process of large number of system components. Therefore such analysis usually related to uncertainties of the model, uncertainties of material data, nuclear data, and other possible errors. In order to get reliable safety analysis it is important to perform benchmark analysis among accident simulation code, and among simulation code with experimental results. For long-life fast reactors without on site refueling variation on spatial neutronic characteristic during the operation time may have important influence on the safety performance and it need special care.

3. Systematics of the report

The systematic of the present report is as follows:

Chapter I : Introduction, discuss about background and objective of the this CRP research.

Chapter II : Research metodology, discuss about the strategy and implementation of the this CRP research

Chapter III : Static neutronic parametric survey results and analysis, discuss about the parametric survey results to investigate the most dominant parameters which influent long life neutronic analysis results.

Chapter IV : Safety analysis parametric survey results and analysis, discuss about the parametric survey results to investigate the most dominant parameters which influent important safety analysis results.

Chapter V : FP Cross section treatment for long life burnup analysis, discuss about the alternatives of better FP treatment for long life burnup analysis.

Conclusion and Recommendation for future work : discuss about the most important results obtained during the present research and recommendation for further research.

CHAPTER II

RESEARCH METHODOLOGY

Current research focus on the performance of benchmark analysis for Pb-Bi/Pb Cooled long-life cores of Small Reactors without On-Site Refuelling and optimization of their inherent/passive safety performance. In general we devide the following research into several main targets as follows:

1. Identify the main important parameters to the long life core static neutronic analysis and inherent safety performance during long life operation without on site refueling.
2. Investigating better and resonable method to reduce the uncertainty error due to parameters' uncertainty.
3. Intensively compare analysis results using many comparable system codes after adopting best methods from the above target number 2.

The first target is performed in this year. Part of second target is also to be achieved in the first year. Some of second target and the third target is just part of the next year target.

II.1 Development of Program for Better study of long period burnup characteristics

We have developed a program using Delphi to control execution the burnup program which was developed by fortran so that parametric survey can be run conveniently. Using this program the graph of parametric survey results can be obtained easily, and the parametric survey process can be run more convenient. The main part of the program can be identified as follows:

- Read the input parameter including type of survey, type of nuclide for which cross section to be perturbed, number of survey, the level of perturbation. Then type of graph to be draw, name of file to save the graph, etc.
- Modify group constants according to the survey specification

- Run the burnup calculation according to the survey specification
- Accumulate the burnup calculation results for various surveys and draw the necessary graph
- Save the graph to the file

The appearance of this program is shown as follows.

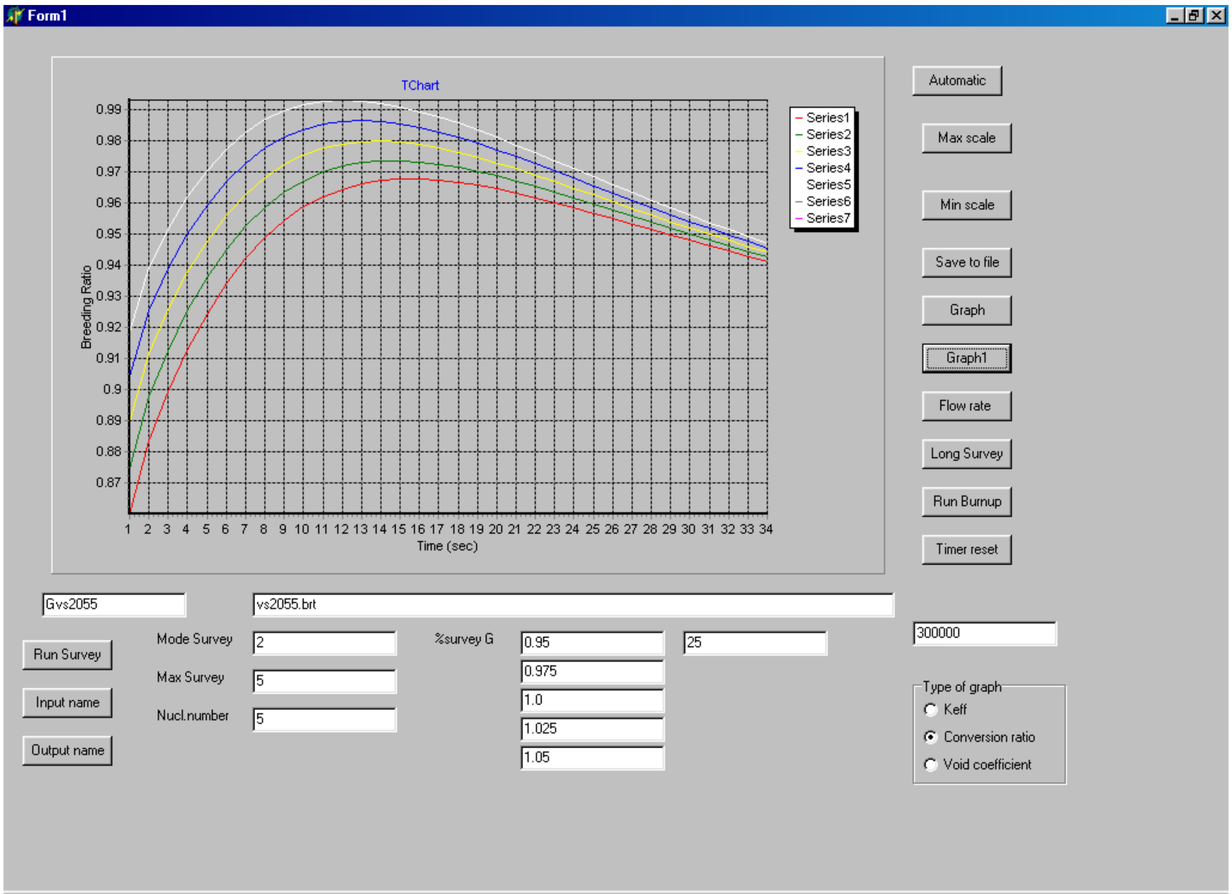

Fig. 1 The program appearance for convertion ratio parametric survey

The above graph is the results of the effect of variation in U-238's capture cross section to the conversion ratio change during burnup. We can input the type of cross section perturbation (Mode survey), number of perturbation (max. survey), nuclide number (Nucl. Number) and the magnitude of perturbation (% survey). The above results show that uncertainty in the U-238 capture cross section will significantly influence the convertion ratio during burnup.

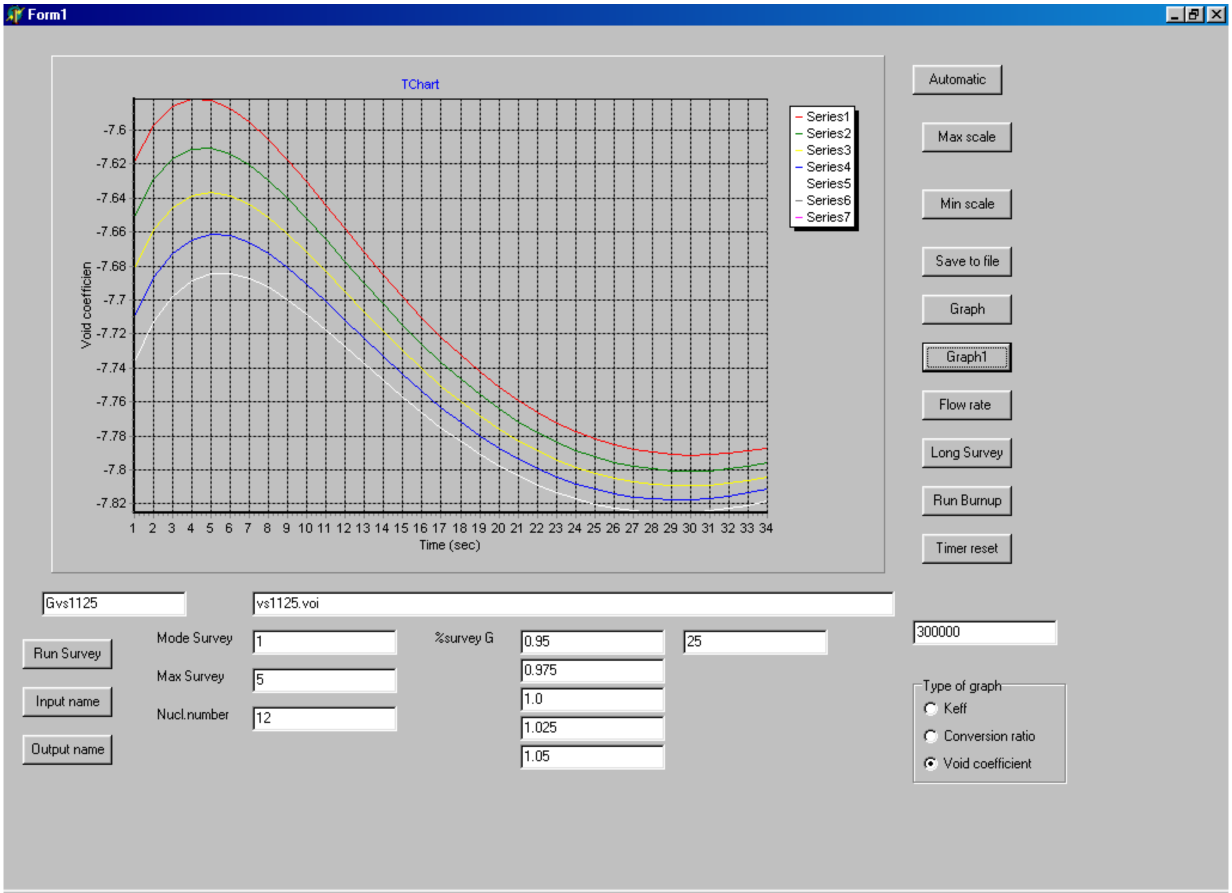

Fig. 2 The program appearance for void coefficient parametric survey

As shown in the above figure the uncertainty in the fission cross section of Pu-239 will significantly change the coolant void coefficient pattern during burnup. This influent islarger for beginning of life (BOL) condition campared to the end of life (EOL) condition.

Here we adopt 20MWe SPINNOR reactor as a sdandard case of our parametric study. The parameters is as follows:

Table 1. SPINNOR A standard case main parameters

Parameter	Parameter Value/description
	SPINNOR A
Installed capacity	55 MWth / 20 MWe
Operation life time (without refueling and fuel shuffling)	15 years
Mode of operation	Basic/load follow (selectable)
Load factor	Beyond 95% *
Summary of major design characteristics - type of fuel - fuel enrichment - type of coolant/moderator - type of structural material	UN-PuN** 10 – 12.5% Pb-Bi eutectic Stainless

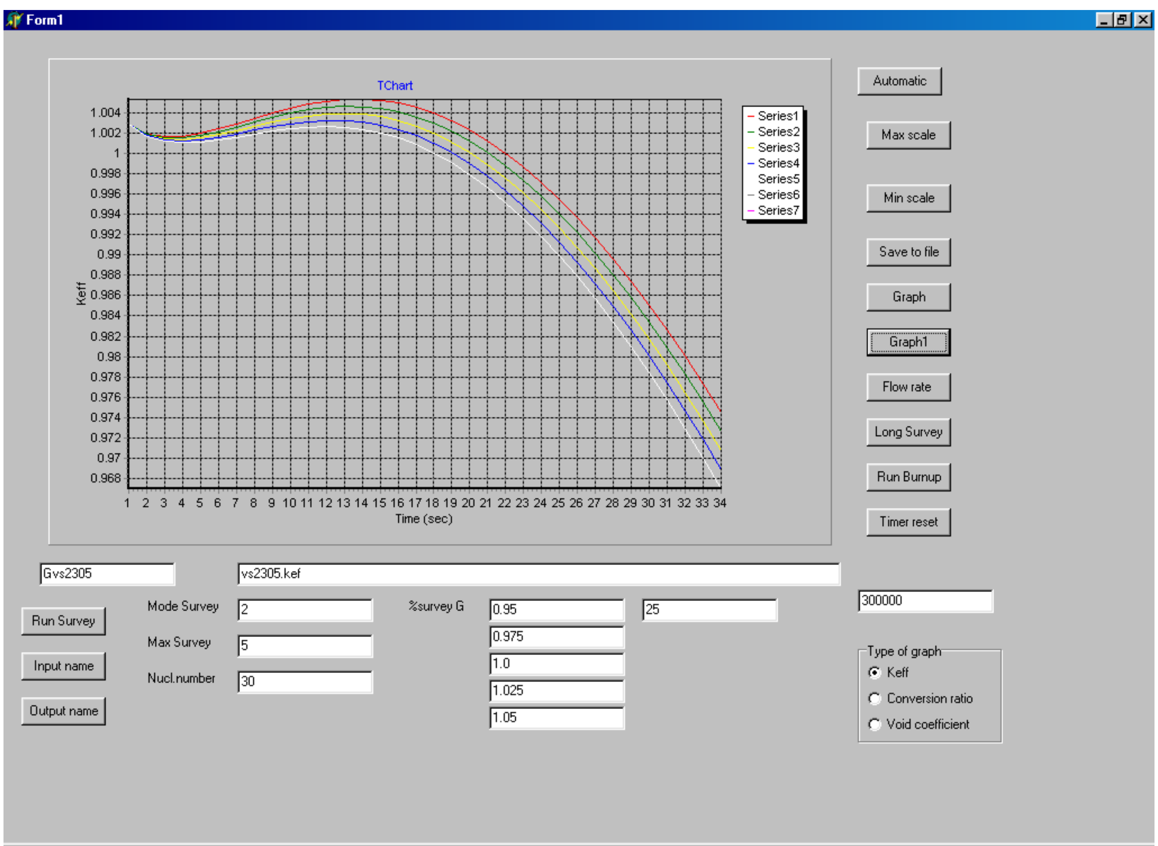

Fig. 3 The program appearance for effective multiplication factor parametric survey

As shown in the above figure the uncertainty in the capture cross section of fission product (FP) will significantly change the coolant void coefficient pattern during burnup. This influent is larger for end of life (BOL) condition compared to the beginning of life (EOL) condition due to large accumulation of FP at the EOL.

CHAPTER III

STATIC NEUTRONIC PARAMETRIC SURVEY RESULTS AND ANALYSIS

III.1 Performing Parametric Survey to Identify Important Parameter During Long Period Burnup Calculation

Here we perform wide spectrum parametric survey to identify what tipe of multigroup cross sections and what tipe of nuclei have has significant impacts on the long period burnup calculation. In general we identify for type of cross sections:

- fission
- capture
- Removal and scattering
- Transport/diffusion coefficient

And for the type of nuclei, we performed parametric survey for 46 included nuclei but we analyze the results for most important nuclei. The results and analysis of the parametric surveys are shown as follows.

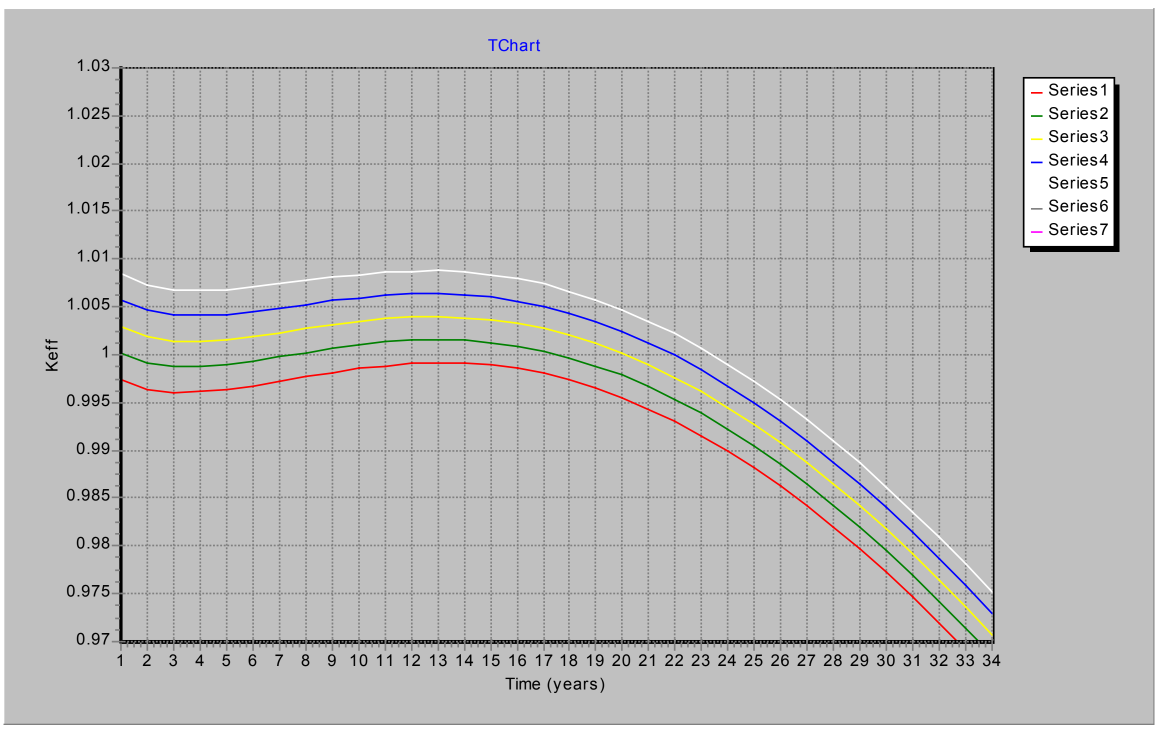

Fig. 4 The effect of fission cross section of U-238 to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

Fig. 4 shows that fission cross section of U-238 has significant influence to the absolute value of effective multiplication factor during burnup. However the basic pattern of effective multiplication factor change during burnup does not change with the variation of this fission cross section. Variation of about 5% of fission cross section of U-238 will account for about 0.6% of change in effective multiplication factors.

The next is the influence of fission cross section of Pu-239 (see. Fig. 5). The variation of Pu-239 fission cross section of about 5% will account for effective multiplication factor change during burnup for about 2% dk/k at the BOL but its influence is reduced at the EOL. This influence is much higher than the effect of U-238 fission cross section because Pu-239 is main fissile isotop in this design.

Fig. 6 shows that fission cross section of Pu-241 has significant influent to effective multiplication factors

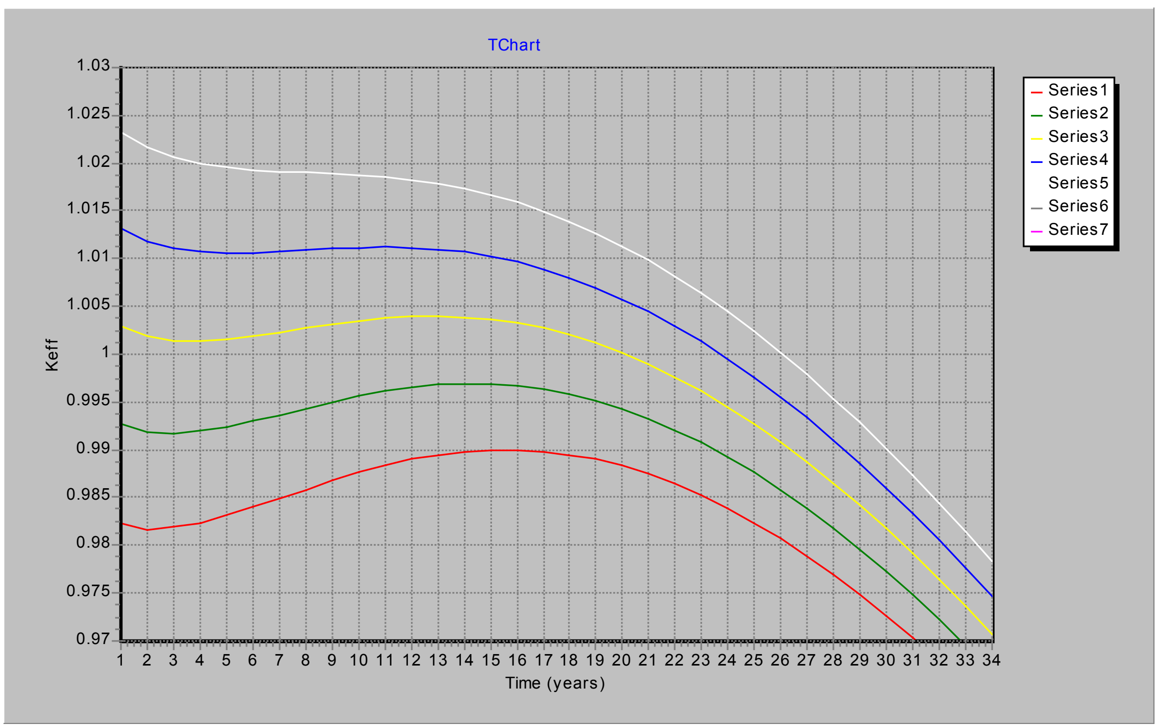

Fig. 5 The effect of fission cross section of Pu-239 to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

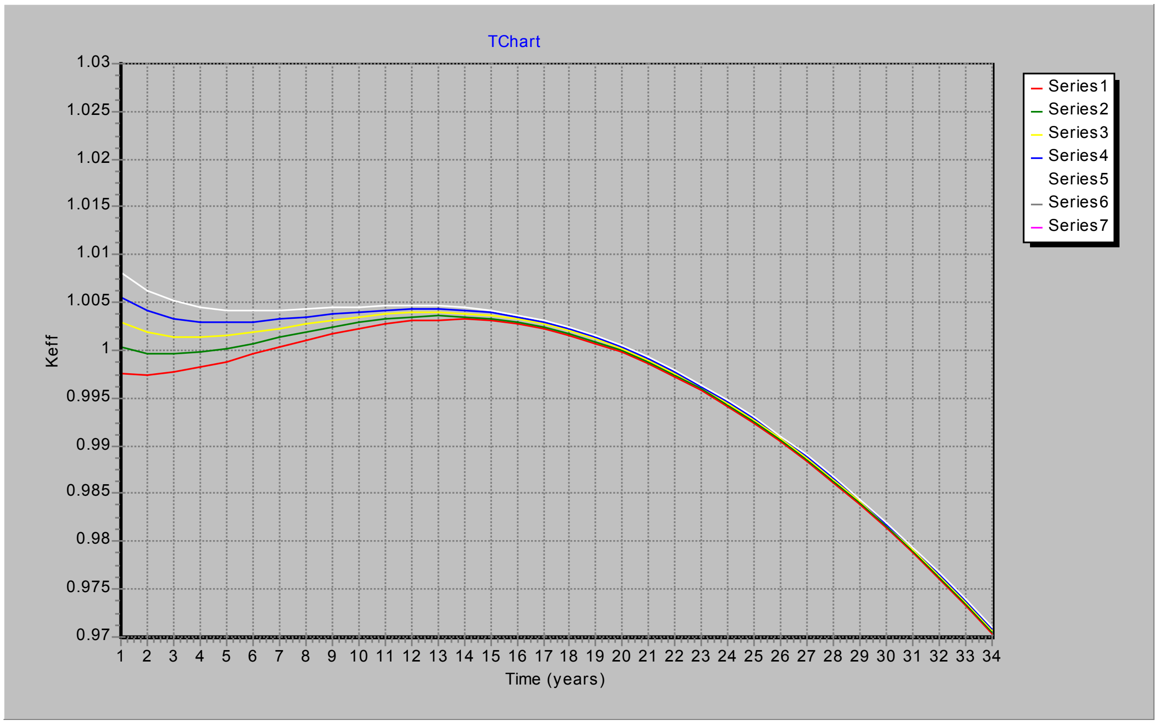

Fig. 6 The effect of fission cross section of Pu-241 to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

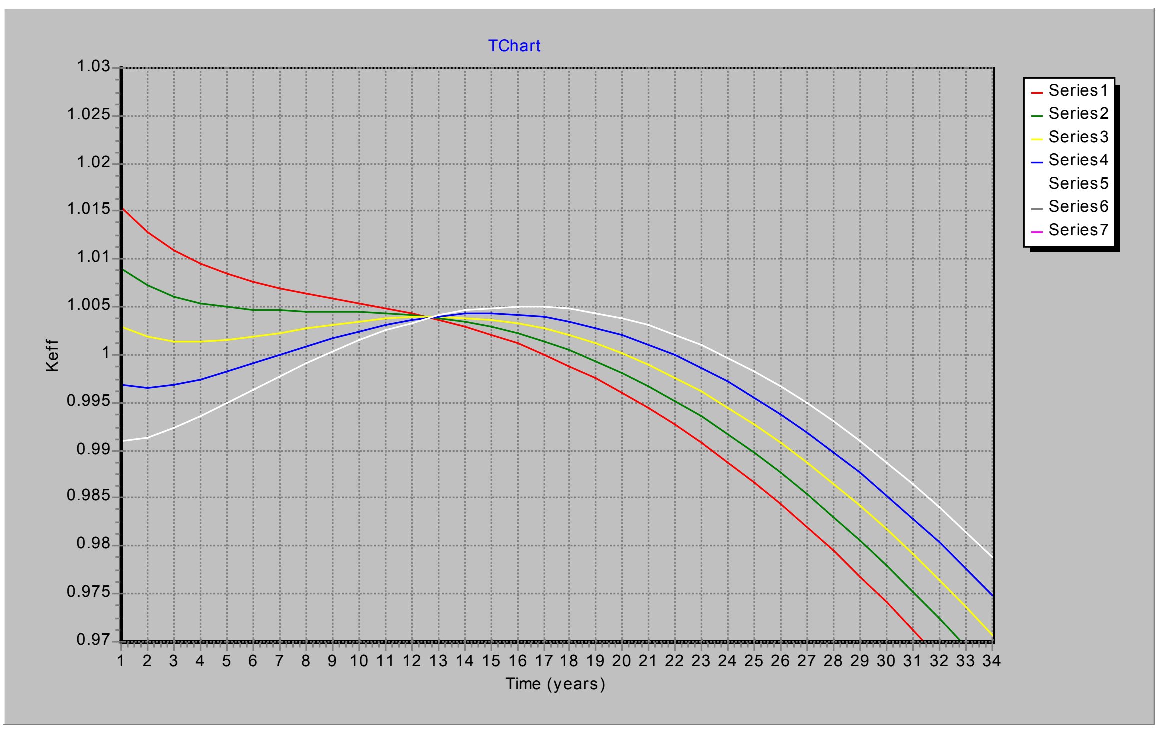

Fig. 7 The effect of capture cross section of U-238 to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

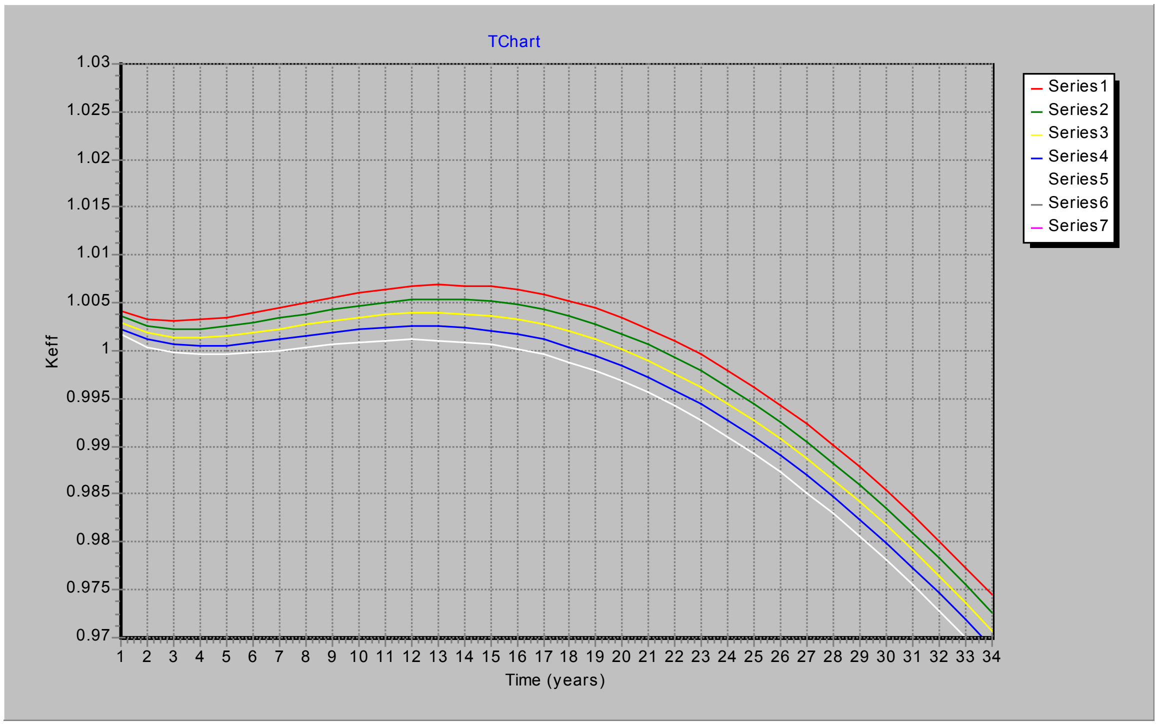

Fig. 8 The effect of capture cross section of Pu-239 to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

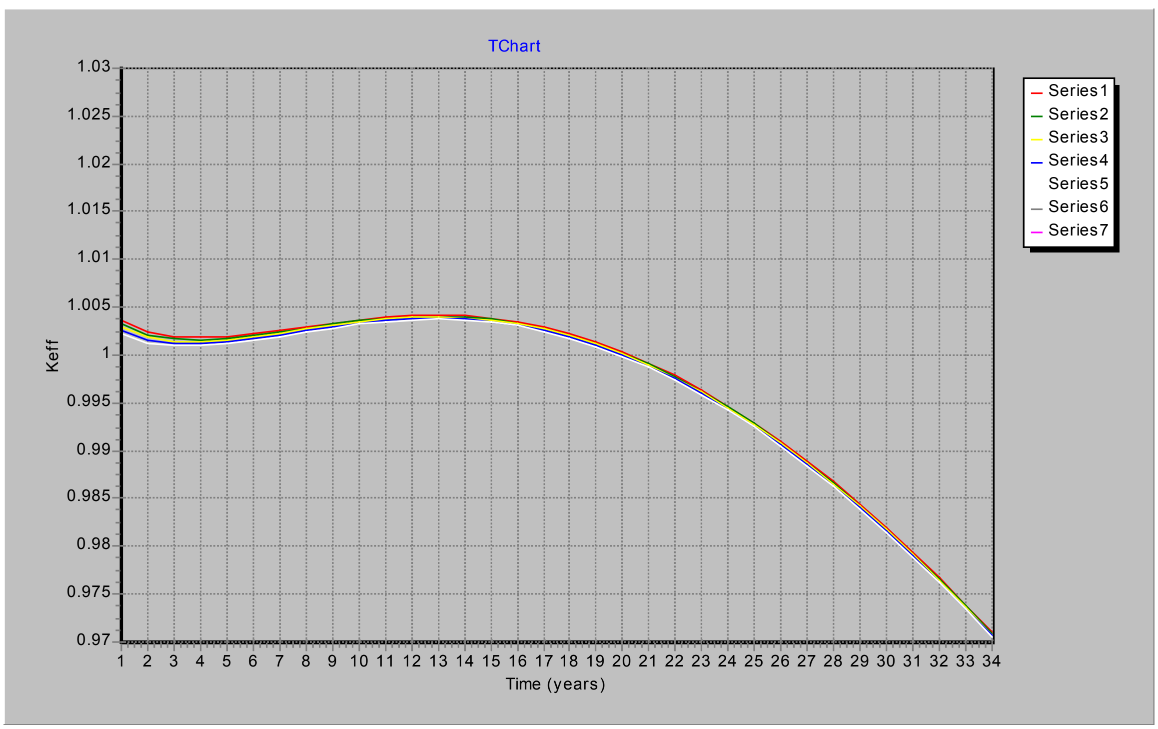

Fig. 9 The effect of capture cross section of Pu-240 to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

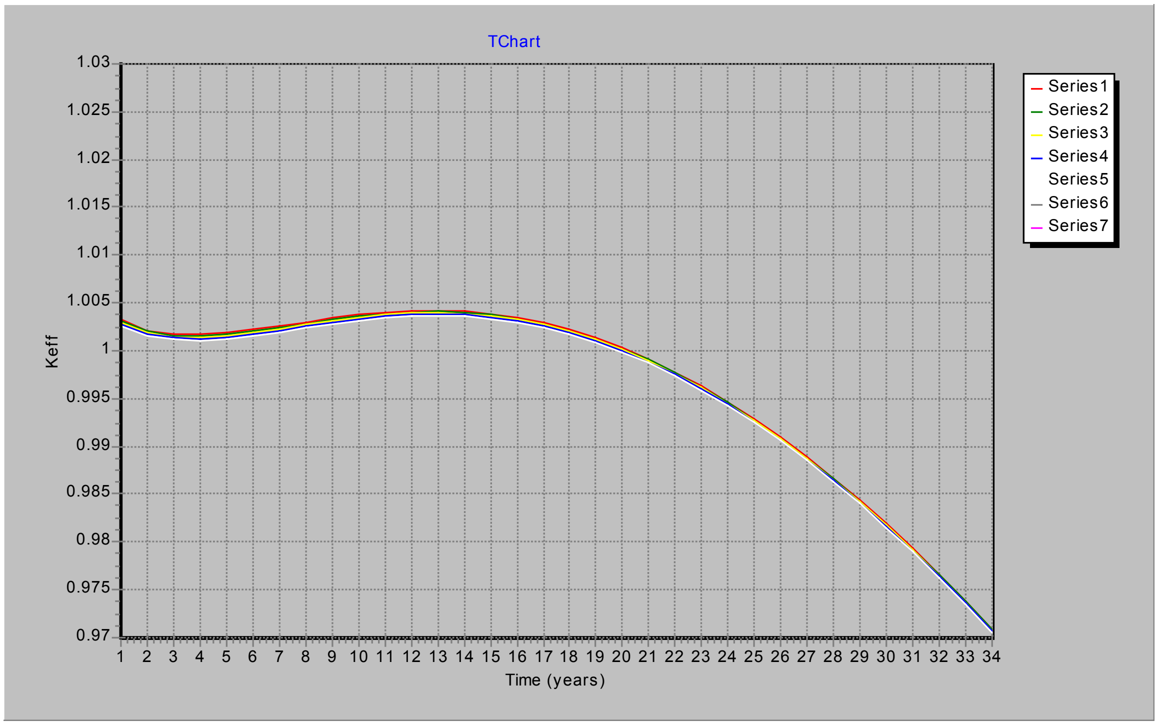

Fig. 10 The effect of capture cross section of Pu-241 to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow: 100%: standard case, green: 97.5%, red : 95%)

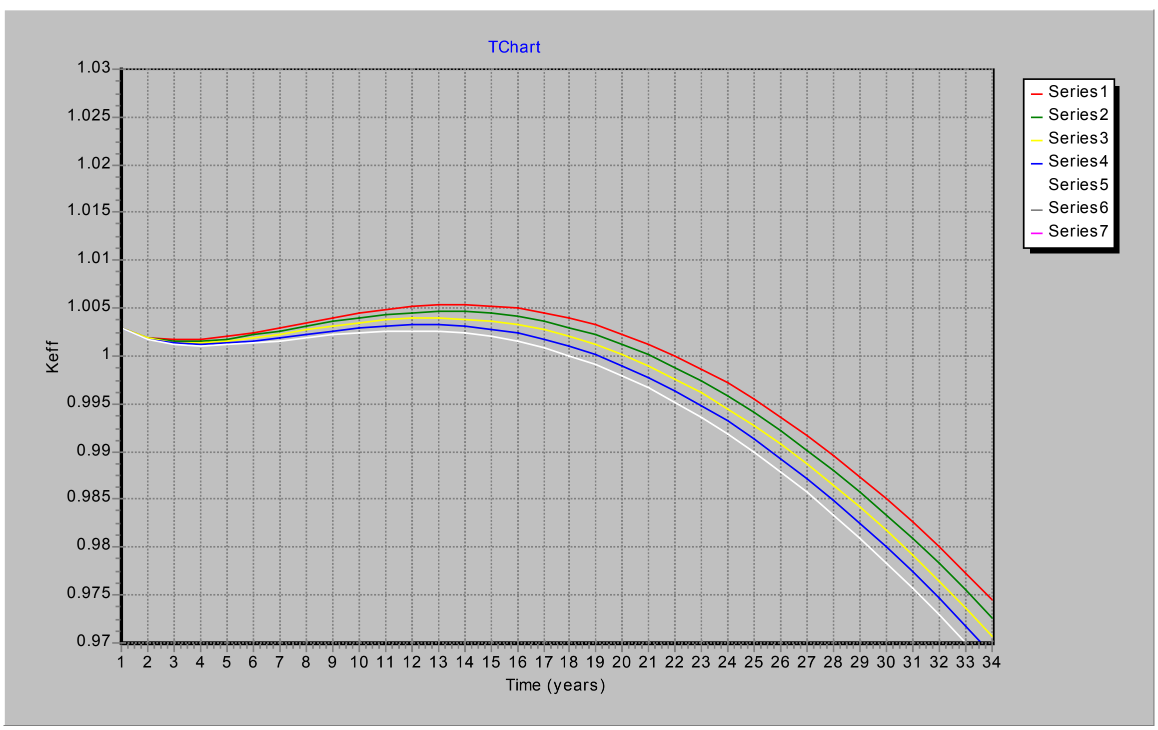

Fig. 11 The effect of capture cross section of Fission product to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

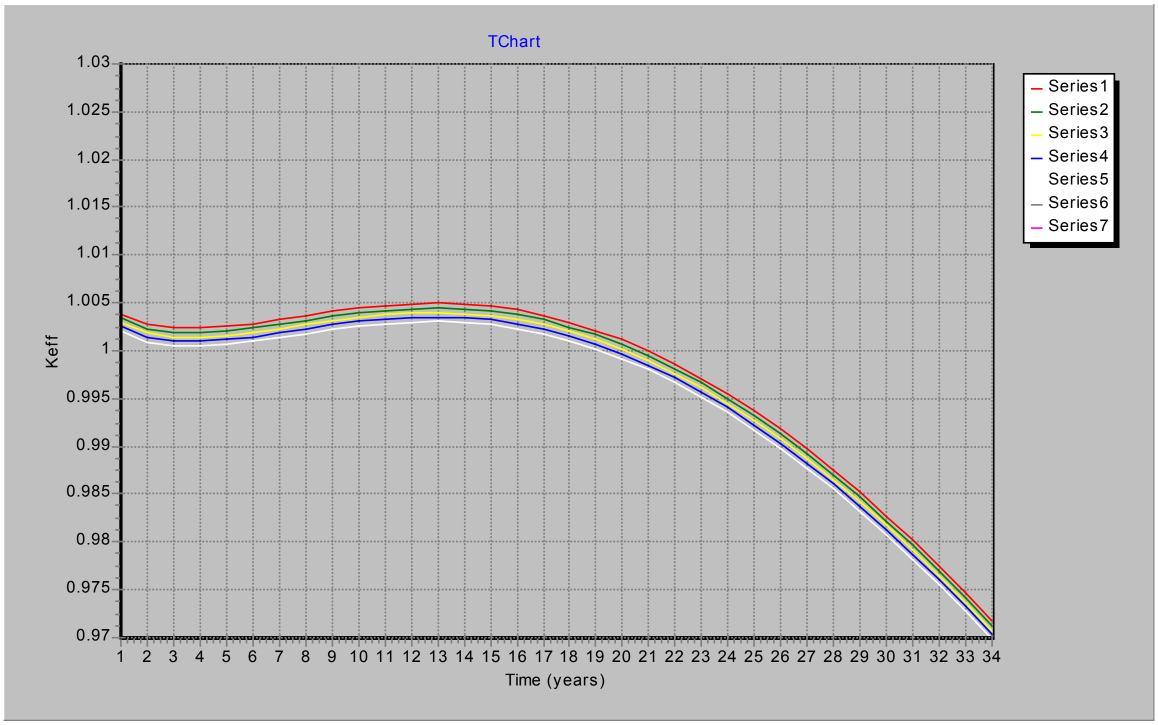

Fig. 12 The effect of capture cross section of lead (Pb) to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%: standard case, green: 97.5%, red : 95%)

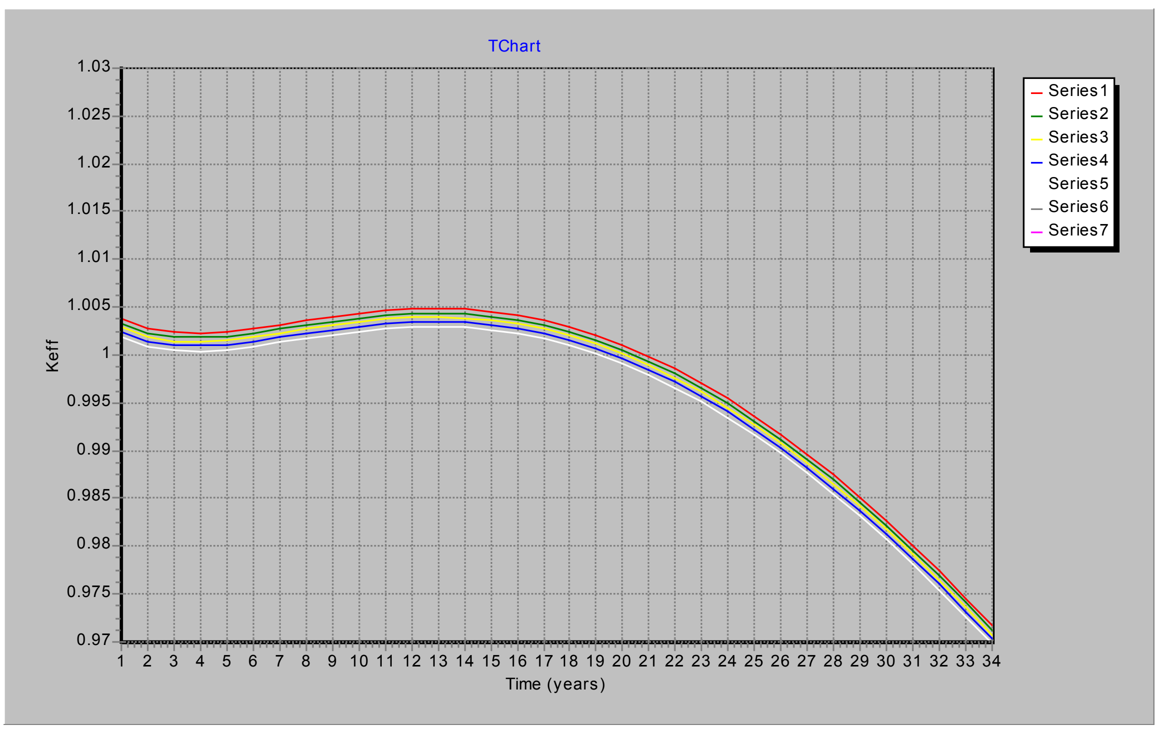

Fig. 13 The effect of capture cross section of Bismuth (Bi) to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%: standard case, green: 97.5%, red : 95%)

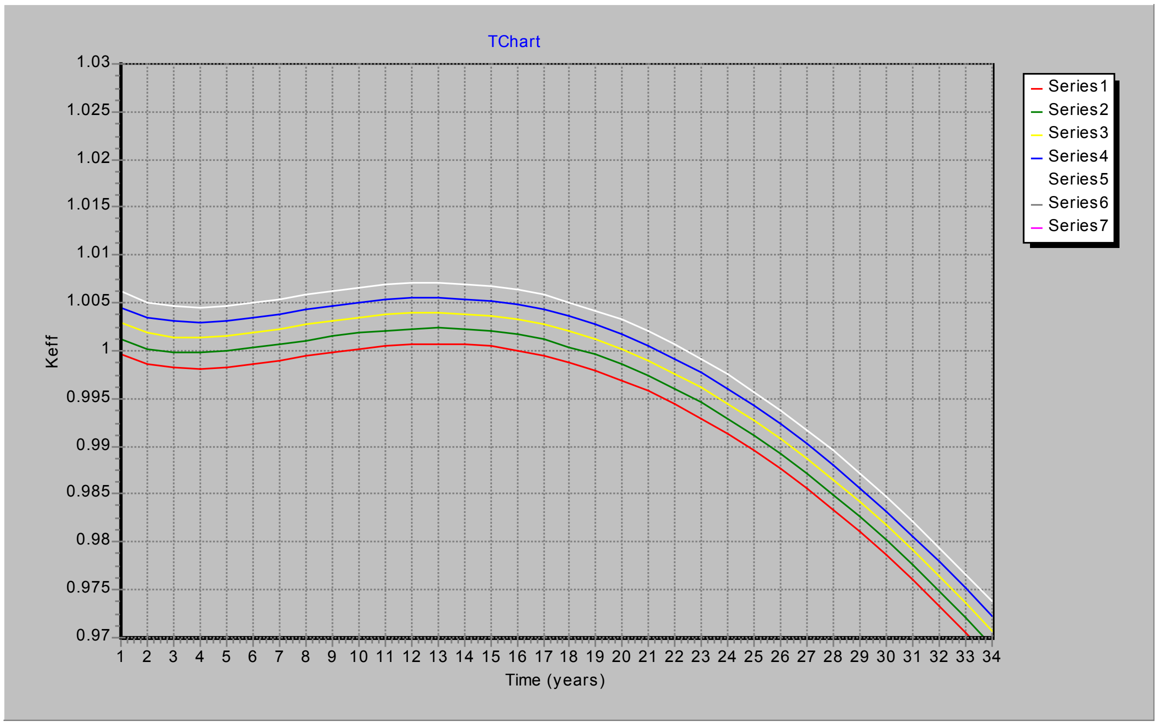

Fig. 14 The effect of transport cross section of lead (Pb) to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%: standard case, green: 97.5%, red : 95%)

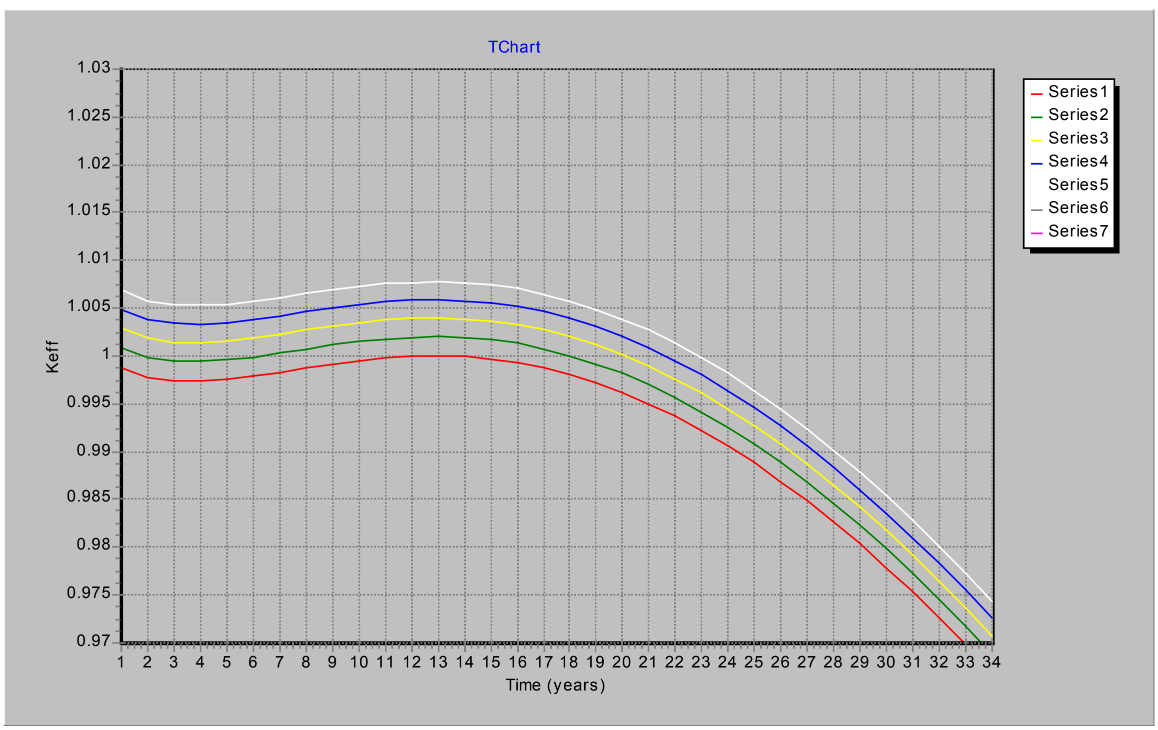

Fig. 15 The effect of transport cross section of bismuth (Bi) to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%: standard case, green: 97.5%, red : 95%)

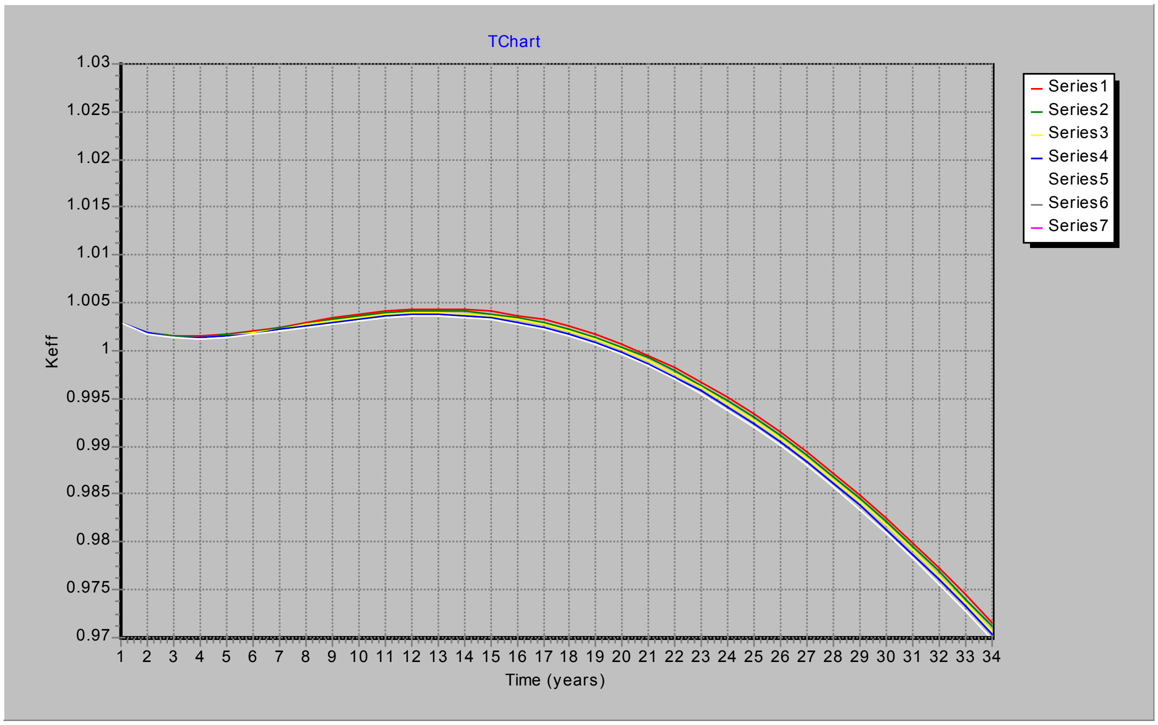

Fig. 16 The effect of scattering cross section of fission product (FP) to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%: standard case, green: 97.5%, red : 95%)

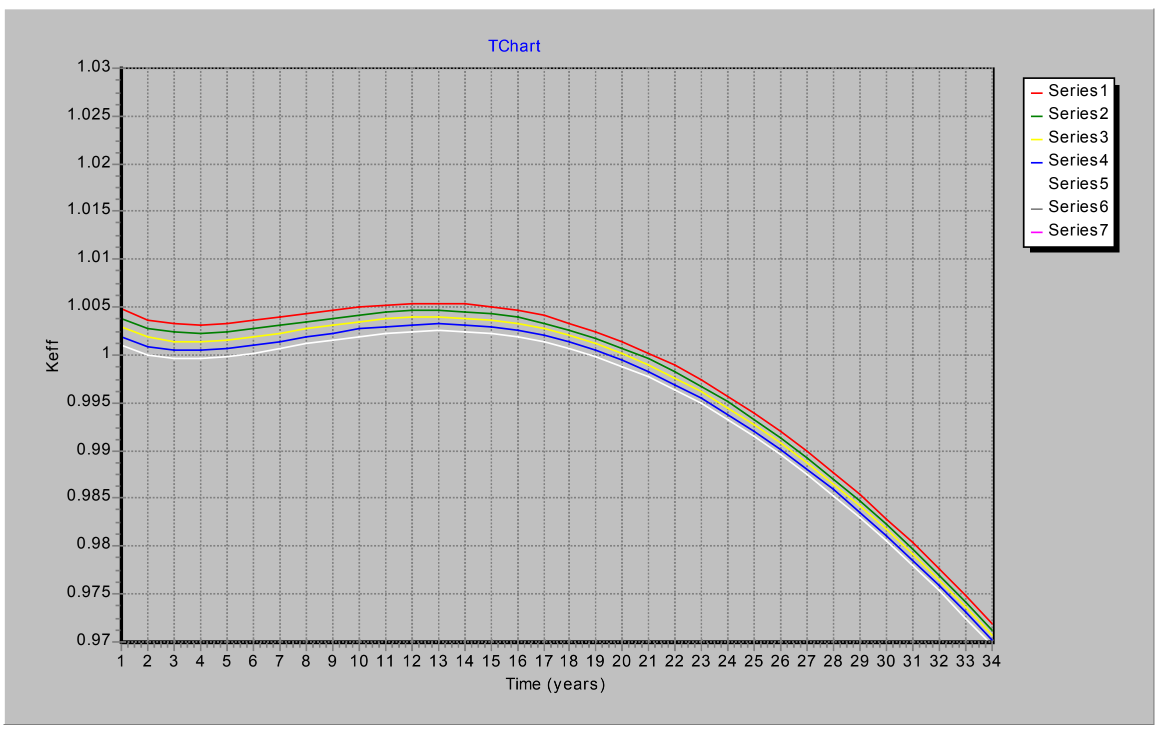

Fig. 17 The effect of scattering cross section of lead (Pb) to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%: standard case, green: 97.5%, red : 95%)

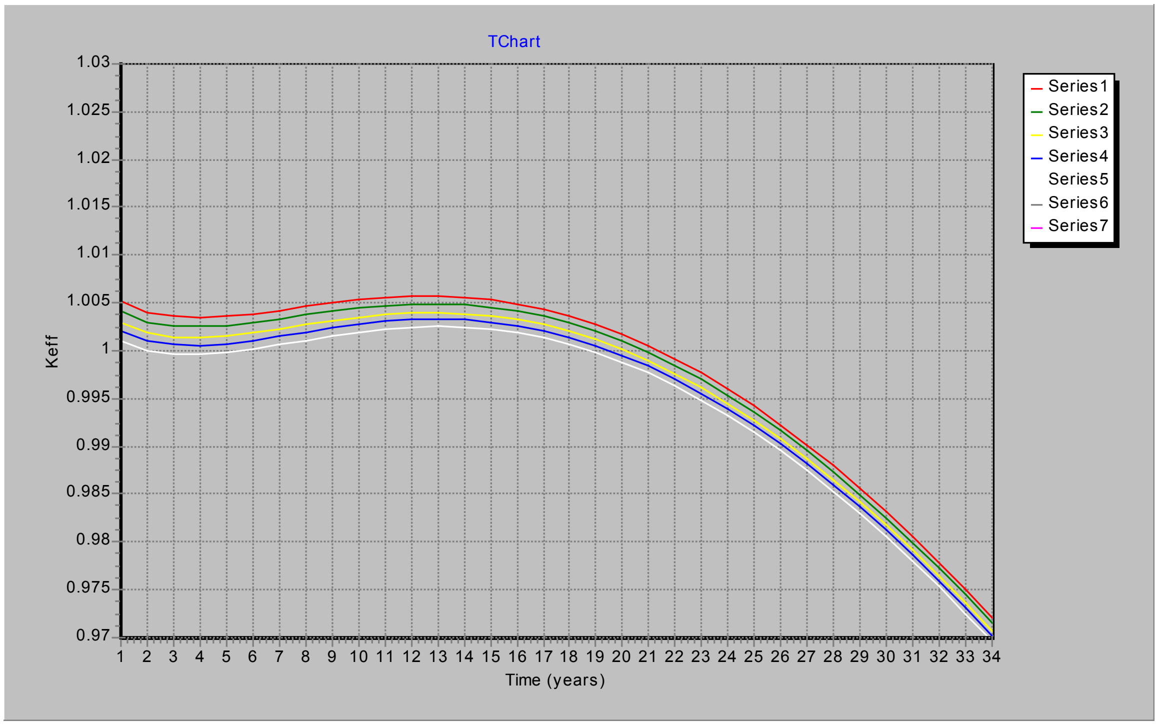

Fig. 18 The effect of scattering cross section of Bismuth (Bi) to the effective multiplication factor change during burnup (white: 105%, blue: 102.5%, yellow:100%: standard case, green: 97.5%, red : 95%)

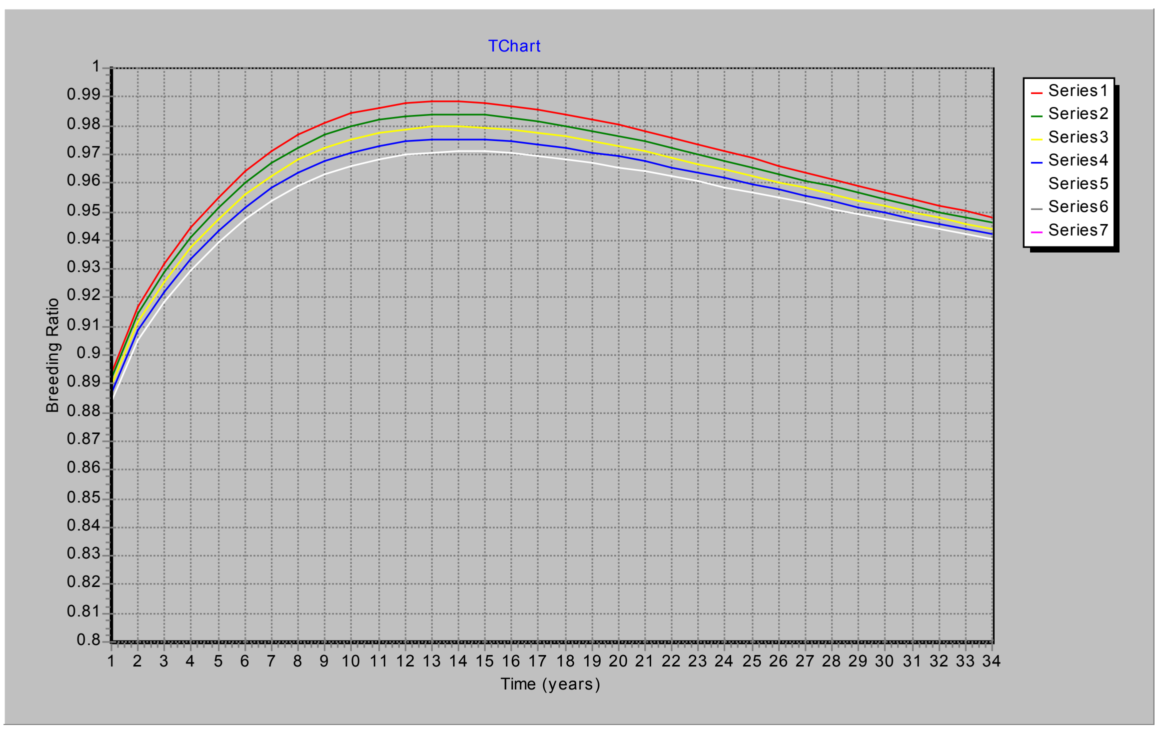

Fig. 19 The effect of fission cross section of Pu-239 to the convertion ratio change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

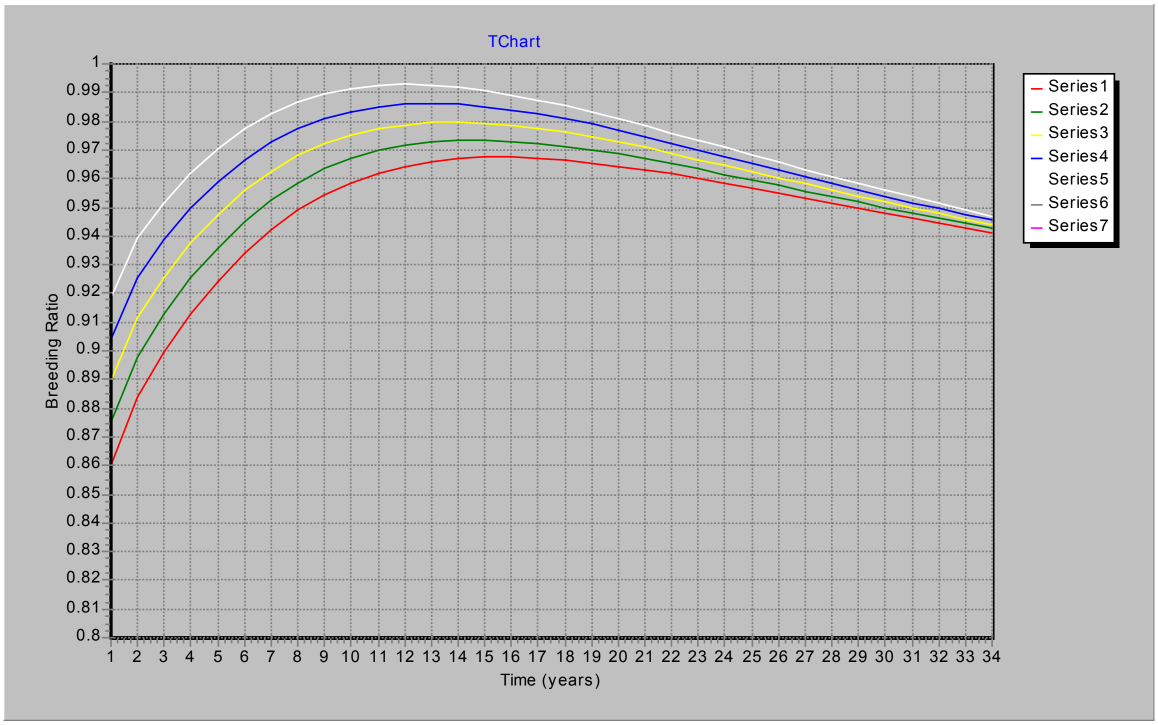

Fig. 20 The effect of capture cross section of U-238 to the convertion ratio change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

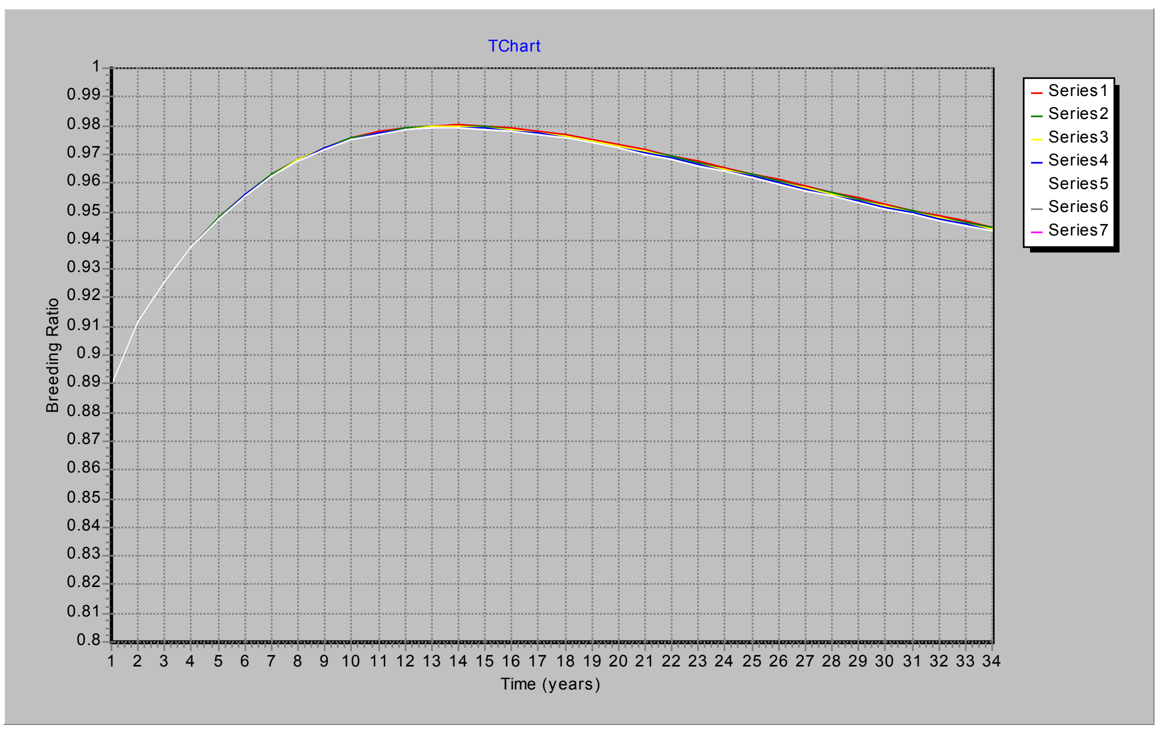

Fig. 21 The effect of capture cross section of fission product (FP) to the convertion ratio change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

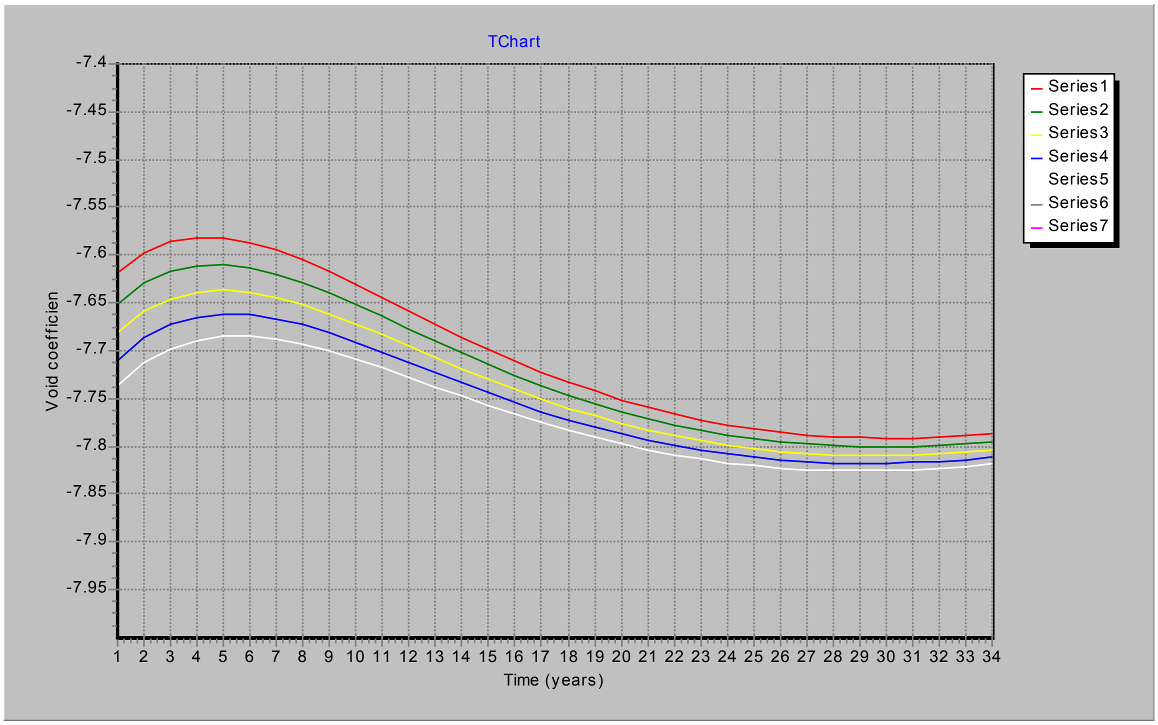

Fig. 22 The effect of fission cross section of Pu-239 to the coolant void coefficient change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

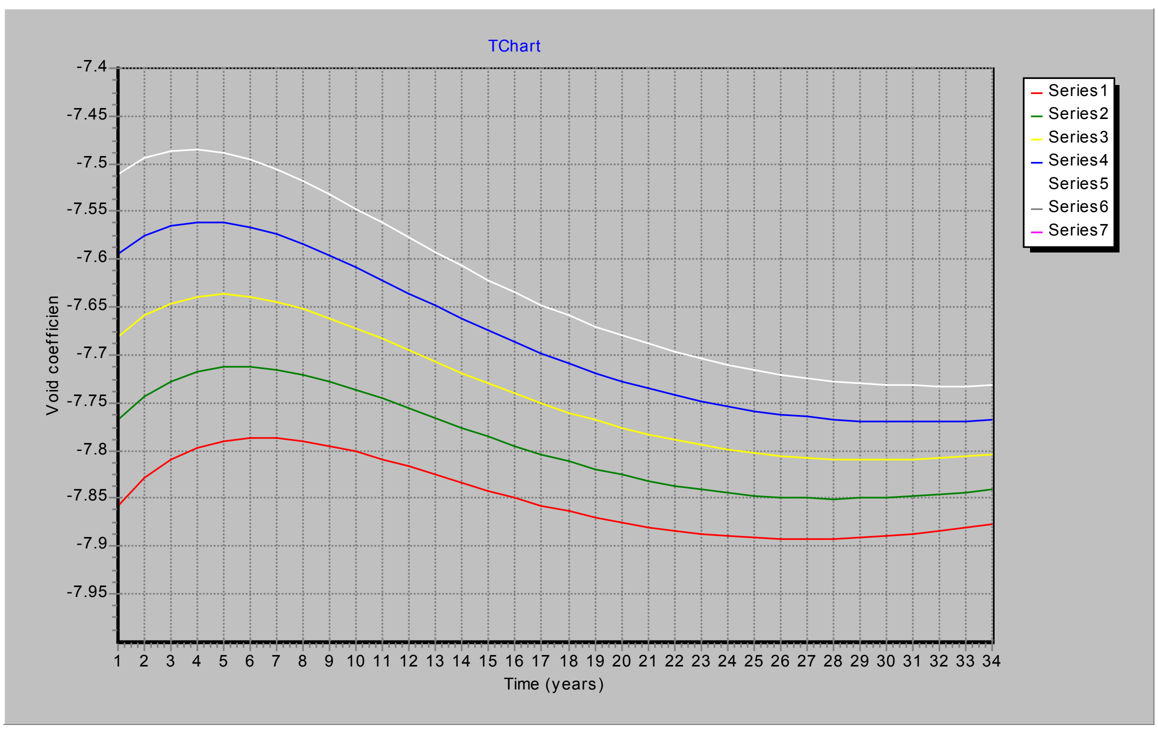

Fig. 23 The effect of capture cross section of U-238 to the coolant void coefficient change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

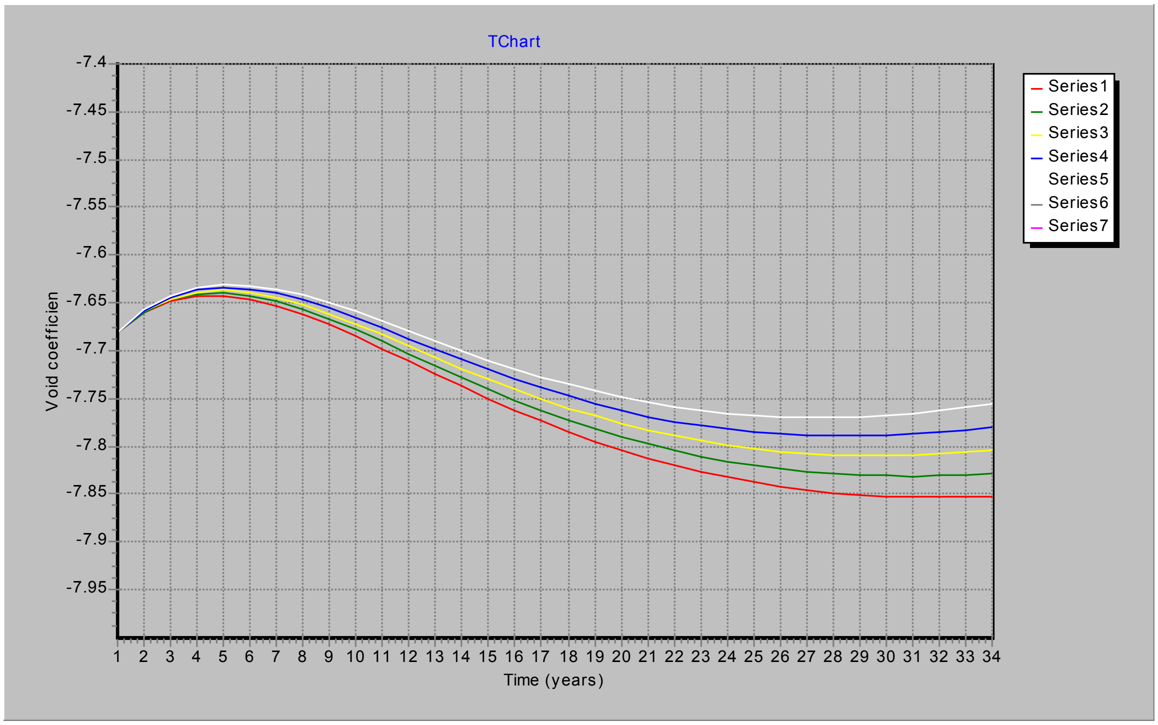

Fig. 24 The effect of capture cross section of fission product (FP) to the coolant void coefficient change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

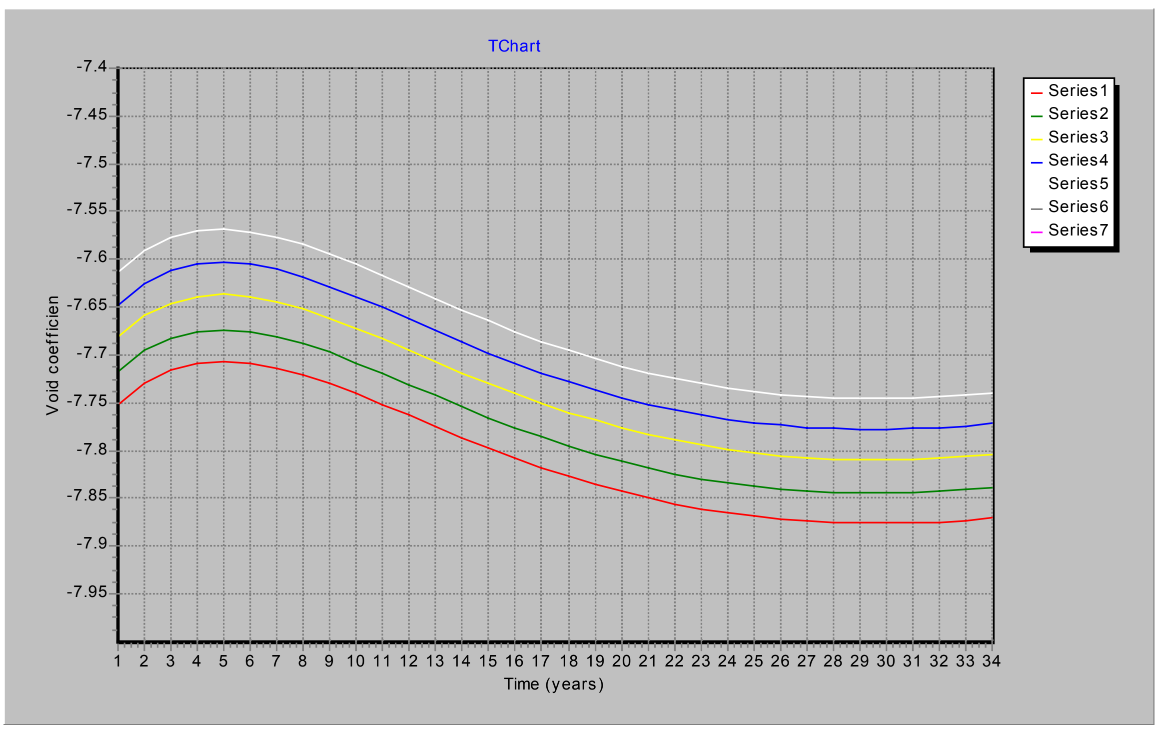

Fig. 25 The effect of scattering cross section of lead (Pb) to the coolant void coefficient change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

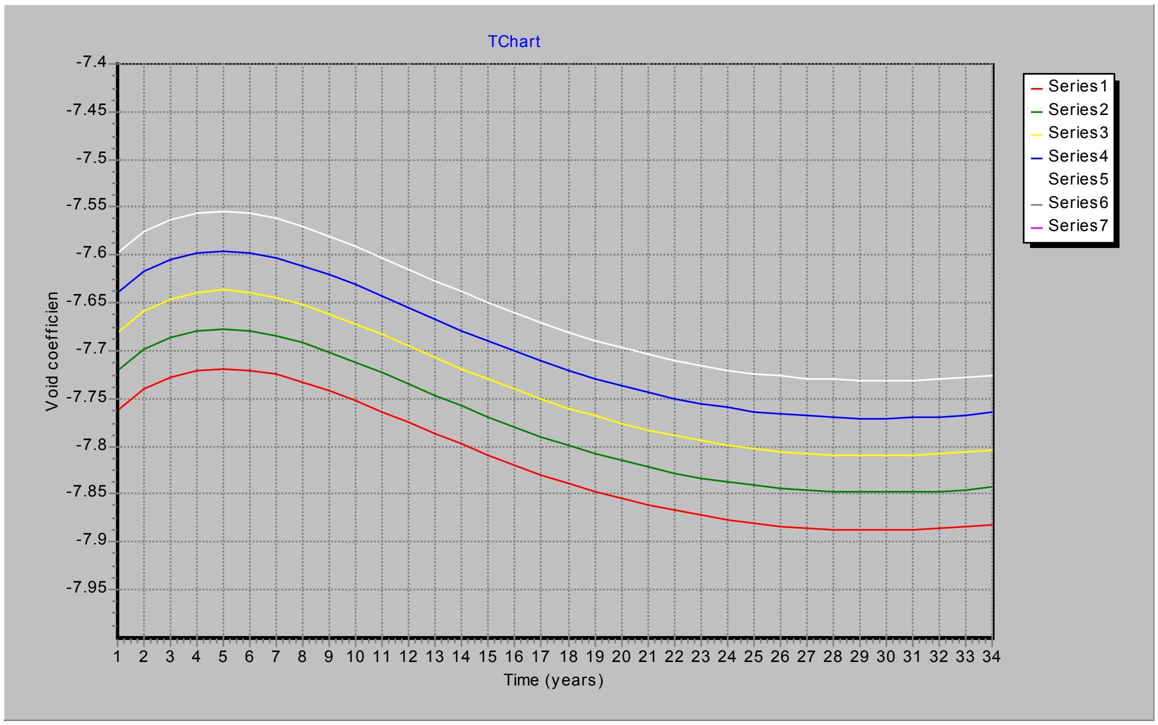

Fig. 26 The effect of scattering cross section of bismuth (Bi) to the coolant void coefficient change during burnup (white: 105%, blue: 102.5%, yellow:100%:standard case, green: 97.5%, red : 95%)

CHAPTER IV

SAFETY ANALYSIS RESULTS AND DISCUSSION

For the safety analysis we perform parametric survey to investigate the effect of parameters to the safety analysis results. Here we perform 5 parametric survey for radial expansion reactivity coefficient, Doppler effect reactivity coefficient, fuel axial expansion reactivity coefficient, coolant density reactivity coefficient, and pump coastdown time.

The results are shown in Table 2 through Table 5. Table 2 shows the results for radial expansion reactivity coefficient variation effects. The table shows that the increase of radial expansion coefficient results in higher maximum total feedback, faster approaching asymptotic level, and lower hot spot coolant, cladding and fuel temperatures.

Table 2 : Radial Expansion Parametric Survey

Parameters	Case +50%	Standard	Case -50%
Power at 90s	37.1%	38.8%	41.3%
Core flowrate at 90s (kg/s)	722	730	742
Max. total feedback	-6.60E-4	-6.02E-4	-5.28E-4
Time of max. feedback(s)	20.15	21.15	23
Primary SG flowrate at 90s	723	732	743
Max. coolant temperature (C)	581.7	600	602
Time of max. coolant temp(s)	18.87	20.4	21.5
Max. cladding temper.(C)	595	604	616
Time of max. cladding temp (s)	18.5	20.5	21.6
Max. fuel temperature (^{o}C)	628	637.8	651
Time of max.fuel temper (s)	18.2	19.1	20

Table 3 shows the results of Doppler coefficient parametric survey results. It is shown that larger doppler coefficient resulted in higher maximum total feedback, faster approaching asymptotic level, and lower hot spot coolant, cladding and fuel temperatures. But the effect is relatively smaller than the radial expansion.

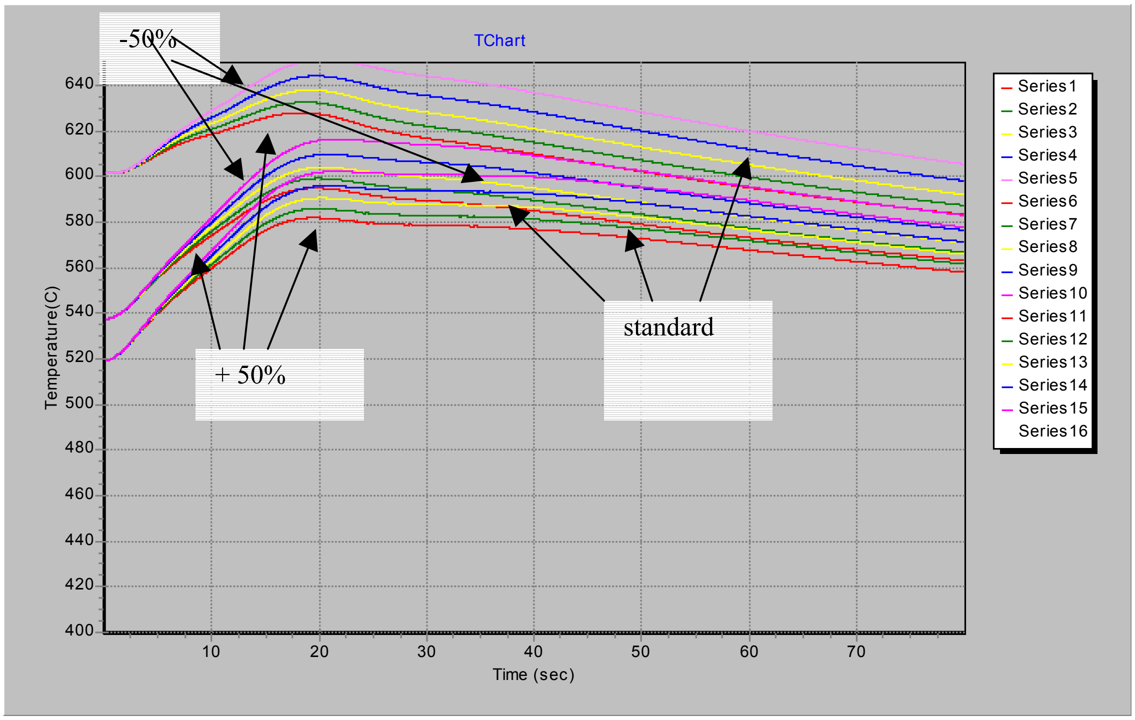

Fig. 27 the effect of radial expansion reactivity coefficient variation to the hot spot temperature.(red: +50%, green: +25%, yellow: standard, blue:-25%, maroon: -50%)

Fig. 27 shows the change of fuel, cladding and coolant temperature hot spot with the time as the results of radial expantion coefficient variation. It is clear that the effect is significant.

Table 3 : Doppler Parametric Survey

Parameters	Case +50%	Standard	Case -50%
Power at 90s	38.3%	38.8%	39.5%
Core flowrate at 90s (kg/s)	727	730	734
Max. total feedback	-6.3E-4	-6.02E-4	-5.63E-4
Time of max. feedback(s)	20.4	21.15	22.1
Primary SG flowrate at 90s	730	732	735
Max. coolant temperature (C)	585	600	596
Time of max. coolant temp(s)	21.5	20.4	20.5
Max. cladding temper.(C)	600	604	610
Time of max. cladding temp (s)	19.4	20.5	21
Max. fuel temperature (^{o}C)	632	637.8	644
Time of max.fuel temper(s)	18.5	19.1	19.7

Table 4 shows the results of axial expansion parametric survey studies. It is shown that the effect of axial fuel expansion variation to the safety performance is small. The higher fuel axial expansion coefficient only lower few degrees in maximum coolant, cladding and fuel temperatures. This effects are much smaller that those of Doppler coefficient and radial expansion coefficient.

Table 5 shows the results of coolant density reactivity coefficient parametric survey studies. It is shown that the effect of coolant density coefficient variation to the safety performance is also small. The higher coolant density coefficient only lower few degrees in maximum coolant, cladding and fuel temperatures. This effects are much smaller that those of Doppler coefficient and radial expansion coefficient bur slightly higher than those of fuel axial expansion coefficient.

Table 4 : Axial Expansion Parametric Survey

Parameters	Case +50%	Standard	Case -50%
Power at 90s	38.6%	38.8%	39.0%
Core flowrate at 90s (kg/s)	729	730	731
Max. total feedback	-6.12E-4	-6.02E-4	-5.9E-4
Time of max. feedback(s)	21.0	21.15	21.4
Primary SG flowrate at 90s	731	732	733
Max. coolant temperature (C)	587	600	592
Time of max. coolant temp(s)	25.1	20.4	20.4
Max. cladding temper.(C)	602.4	604	605.6
Time of max. cladding temp (s)	20.3	20.5	20.6
Max. fuel temperature (^{o}C)	636	637.8	639.4
Time of max.fuel temper(s)	19.0	19.1	20.4

Table 5 : Coolant Void Parametric Survey

Parameters	Case +50%	Standard	Case -50%
Power at 90s	38.4%	38.8%	39.1
Core flowrate at 90s (kg/s)	729	730	732
Max. total feedback	-6.14E-4	-6.02E-4	-5.9E-4
Time of max. feedback(s)	21	21.15	21.4
Primary SG flowrate at 90s	731	732	734
Max. coolant temperature (C)	589	600	592
Time of max. coolant temp(s)	20.0	20.4	20.3
Max. cladding temper.(C)	602	604	606
Time of max. cladding temp (s)	20.0	20.5	20.5
Max. fuel temperature (^{o}C)	636	637.8	639.5
Time of max. fuel temper.(s)	19.0	19.1	20.8

Table 6 : Primary pump coastdown Time Parametric Survey

Parameters	Case +50%	Standard	Case -50%
Power at 90s	41%	38.8%	36.9
Core flowrate at 90s (kg/s)	767	730	702
Max. total feedback	-4.89E-4	-6.02E-4	-8.36E-4
Time of max. feedback(s)	33.1	21.15	19.1
Primary SG flowrate at 90s	770	732	704
Max. coolant temperature (C)	578	600	619
Time of max. coolant temp(s)	34.5	20.4	18.4
Max. cladding temper.(C)	590	604	630
Time of max. cladding temp (s)	34.0	20.5	18.6
Max. fuel temperature (^{o}C)	627.5	637.8	658
Time of max. fiel temper.(s)	19.5	19.1	18.3

The last parametric survey is about the effect of pump coastdown halving time. Table 6 shows this parametric survey results. From Table 6 we can learn that pump coastdown halving time is important parameter and its reults is comparable or even higher than the effect of core radial expansion coefficient. The changes in coolant, cladding and fuel temperatures are quite significant. The increase or decrease about 50% of this parameter can reulted in the change of coolant clading and fuel hot spot temperature for about few ten degrees.

CHAPTER V

PRELIMINARY STUDY ON FISSION PRODUCT TREATMENT DURING LONG LIFE CORE BURNUP

The results of burnup parametric studies show that FP cross section is important to be treated better in order that the error should be as small as possible (less than 1%) so that the neutronic analysis results becomes more reliable. The problem is that so many nuclides of FP are unstable and has varied decay conditions. Therefore we should find reasonable method with expected resonable accuracy.

To start our discussion, it is very good to refer the work done by Shiro TABUCHI and Takafumi AOYAMA of JAERI on the study of "Lumped Group Constants of FP Nuclides for Fast Reactor Shielding Calculation Based on JENDL-3.2". They ranked the FP nuclides from the poin of view of contribution to the total FP cross section under fast reactor (sodium cooled) neutron energy spectrum. From their study we find that in order to cover up to 165 nuclides to get high accuracy of FP capture cross section.

The problem to treat such number of FP cross section individually is very complicated because we need also to consider other nuclides related to those 165 nuclides. For ilustration the FP nuclide with contribution of capture cross section more than 0.01% is shown in the following table.

Table 7 Important FP Nuclides Data

No	Z	A	%X-sect	Symbol
1	44	101	8.93	Ru
2	46	105	8.93	Pd
3	43	99	7.06	Tc
4	45	103	6.02	Rh
5	55	133	5.72	Cs
6	46	107	4.65	Pd
7	42	97	4.54	Mo
8	62	149	4.39	Sm
9	61	147	3.77	Pm
10	60	145	3.37	Nd
11	55	135	2.74	Cs

12	60	143	2.64	Nd
13	54	131	2.38	Xe
14	44	102	2.21	Ru
15	62	151	2.19	Sm
16	42	95	2.15	Mo
17	42	98	1.89	Mo
18	47	109	1.80	Ag
19	44	104	1.69	Ru
20	42	100	1.58	Mo
21	63	153	1.56	Eu
22	40	93	1.27	Zr
23	44	103	1.19	Ru
24	59	141	1.03	Pr
25	53	129	0.97	I
26	40	95	0.88	Zr
27	40	96	0.75	Zr
28	60	146	0.70	Nd
29	54	132	0.69	Xe
30	46	108	0.68	Pd
31	41	95	0.67	Nb
32	58	141	0.62	Ce
33	40	91	0.61	Zr
34	40	92	0.48	Zr
35	54	134	0.48	Xe
36	44	106	0.48	Ru
37	62	152	0.48	Sm
38	60	148	0.46	Nd
39	48	111	0.44	Cd
40	37	85	0.43	Rb
41	53	127	0.42	I
42	57	139	0.42	La
43	46	106	0.41	Pd
44	63	155	0.35	Eu
45	40	94	0.32	Zr
46	62	147	0.31	Sm
47	58	142	0.29	Ce
48	60	150	0.28	Nd
49	60	147	0.26	Nd
50	55	137	0.25	Cs
51	39	91	0.20	Y
52	60	144	0.19	Nd
53	36	83	0.19	Kr
54	58	144	0.18	Ce
55	64	157	0.18	Gd
56	46	110	0.14	Pd
57	42	99	0.14	Mo
58	64	156	0.13	Gd
59	48	113	0.11	Cd
60	55	134	0.11	Cs
61	63	154	0.10	Eu
62	58	140	0.10	Ce
63	51	125	0.10	Sb
64	65	159	0.10	Tb
65	62	154	0.10	Sm
66	38	90	0.10	Sr
67	53	131	0.09	I
68	39	89	0.09	Y

69	56	138	0.08	Ba
70	59	143	0.08	Pr
71	35	81	0.08	Br
72	52	130	0.08	Te
73	49	115	0.08	In
74	52	128	0.07	Te
75	48	112	0.07	Cd
76	52	129m	0.07	Te
77	37	87	0.06	Rb
78	36	84	0.06	Kr
79	54	133	0.05	Xe
80	51	121	0.05	Sb
81	52	127m	0.05	Te
82	61	148m	0.05	Pm
83	34	79	0.05	Se
84	45	105	0.05	Rh
85	62	150	0.04	Sm
86	51	123	0.04	Sb
87	64	155	0.03	Gd
88	50	117	0.03	Sn
89	61	149	0.03	Pm
90	54	136	0.03	Xe
91	46	104	0.03	Pd
92	64	158	0.03	Gd
93	44	100	0.03	Ru
94	36	85	0.03	Kr
95	38	89	0.03	Sr
96	48	114	0.02	Cd
97	38	88	0.02	Sr
98	50	119	0.02	Sn
99	62	148	0.02	Sm
100	34	82	0.02	Se
101	56	136	0.02	Ba
102	47	110m	0.02	Ag
103	34	77	0.01	Se
104	36	86	0.01	Kr
105	63	156	0.01	Eu
106	34	80	0.01	Se
107	63	151	0.01	Eu
108	48	116	0.01	Cd
109	50	118	0.01	Sn
110	48	110	0.01	Cd
111	34	78	0.01	Se
112	54	130	0.01	Xe
113	56	137	0.01	Ba
114	64	160	0.01	Gd
115	56	140	0.01	Ba
116	50	126	0.01	Sn
117	52	125	0.01	Te
118	50	120	0.01	Sn

Here we find many alternatives method to treat this FP nuclides:

First alternative: Rigorous treatment : We cover 165 nuclides with other relevant FP nuclides in direct individual burnup calculation. This method will give rigorous results but with considerable calculation time. However this method is important to test other simpler methods.

$$\frac{dN_i}{dt} = y_i \Sigma_f \phi + \sum_{\substack{\text{all relevant} \\ \text{source}}} \lambda_j N_j + \sum_{\substack{\text{all relevant} \\ \text{source}}} \sigma_j \phi N_j - \lambda_i N_i - \sigma_i \phi N_i$$

This method will need much of computing time due to the fact that important nuclides also depend on other nuclides such as shown in the following chain

$${}_{36}K_r^{101} \rightarrow {}_{37}R_b^{101} \rightarrow {}_{38}S_r^{101} \rightarrow {}_{39}Y^{101} \rightarrow {}_{40}Z_r^{101} \rightarrow$$

$${}_{41}N_b^{101} \rightarrow {}_{42}M_o^{101} \rightarrow {}_{43}T_c^{101} \rightarrow {}_{44}R_u^{101} \leftarrow {}_{45}R_h^{101}$$

$$\uparrow {}_{45}R_{hm}^{101}$$

Second alternative: Lumped FP treatment : We just build best FP lumped cross section for many general condition and use this FP group constant in burnup calculation. This method can give accurate results if the spectrum is same or near the spectrum to build the lumped FP cross section. The process in this method mainly can be formulated as follows

$$\frac{dN_i}{dt} = \sum_{\substack{\text{all relevant} \\ \text{source}}} \lambda_j N_j + \sum_{\substack{\text{all relevant} \\ \text{source}}} \sigma_j \phi N_j - \lambda_i N_i - \sigma_i \phi N_i$$

$$\text{at certain } \phi \text{ and } \mathrm{t} = \mathrm{t}_1 \Rightarrow \mathrm{N_i}(t_1)$$

$$\sigma_{cFP}(\phi, t_1) = \frac{\sum_i \sigma_{ci} N_i(\phi, t_1)}{\sum_i N_i(\phi, t_1)}$$

Third alternatives : Combination method: We treat some most important nuclides individually and treat the rest FP using lumped FP cross section. This method seems to be good alternative for general usage.

$$\frac{dN_i}{dt} = y_i \Sigma_f \phi + \sum_{\substack{\text{all relevant}\\ \text{source}}} \lambda_j N_j + \sum_{\substack{\text{all relevant}\\ \text{source}}} \sigma_j \phi N_j - \lambda_i N_i - \sigma_i \phi N_i$$

$$\text{at certain } \phi \text{ and } \mathrm{t} = \mathrm{t}_1 \Rightarrow \mathrm{N_i}(t_1)$$

$$\sigma_{cFPTail}(\phi, t_1) = \frac{\sum_{i,\text{rest of FP}} \sigma_{ci} N_i(\phi, t_1)}{\sum_{i,\text{rest of FP}} N_i(\phi, t_1)}$$

Forth alternative : Lumped FP cross section with many interpolable parameter: We develop the concept similar to the back ground cross section in the Bondanrenko based cell calculation libraries. This will improve Lumped FP cross section results for general usage.

$$\frac{dN_i}{dt} = y_i \Sigma_f \phi + \sum_{\substack{\text{all relevant}\\ \text{source}}} \lambda_j N_j + \sum_{\substack{\text{all relevant}\\ \text{source}}} \sigma_j \phi N_j - \lambda_i N_i - \sigma_i \phi N_i$$

$$\text{at certain composition of } \mathrm{N_i} \text{ of FP nuclides and other nuclides}$$

$$\sigma_{oFPi} = \frac{\sum_{\substack{\text{All FP and}\\ \text{other nuclides}\\ \text{except i nuclide}}} \sigma_j N_j(\phi, t_1)}{N_i(\phi, t_1)}$$

$$\sigma_{cFPi} = f_i(\sigma_{oFPi})$$

Fifth alternative : We develop the few group effective FP similar to that in reactor kinetic problem. If we can get reasonable good few group effective FP then we can solve for all type of the core generally.

$$\frac{dN_i}{dt} = y_i \Sigma_f \phi + \sum_{\substack{\text{all relevant} \\ \text{source}}} \lambda_j N_j + \sum_{\substack{\text{all relevant} \\ \text{source}}} \sigma_j \phi N_j - \lambda_i N_i - \sigma_i \phi N_i$$

i is refer to certain FP effective group

CHAPTER VI
CONCLUSION AND RECOMENDATION

From the parametric survey results, we find that FP cross section is important to be considered to get reliable neutronic analysis results. Some other cross section is also critical such as U-238 capture cross section and main fissile fission cross section, and Pb and Bi transport and scattering cross section. FP cross section is important to be treated in more accurate way to get better accuracy especially at the end of life.

From the accident analysis parametric survey results we find that radial expansion, doppler and pump coastdown halving time are important and have strong influence to the safety analysis results. Radial expansion coefficient need better verification to get less uncertainty.

To treat FP group constant we propose 5 aternative methods, rigorous treatment, usual lumped model, combination of rogorous and lumped model, lumped model with many parameters, and effective FP group similar to the kinetic decay group.

Here we recommend to focus on FP group constant treatment and Radial expansion coefficient more rigorous treatment during the next year program.

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