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

THIRD YEAR PROGRESS REPORT

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
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2008

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Bandung, March 10, 2008

CHIEF SCIENTIFIC INVESTIGATOR ***CHERCHEUR PRINCIPAL:***	**HEAD OF INSTITUTE** ***LE DIRECTEUR DE L'INSTITUT:***
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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

In the third year, based on the results from the first and second years we propose to focus on solving FP treatment group constant with the following approach:

1. Adopting general strategy for determining fission product yield data by adopting RBEC type reactor which as reference core. This RBEC reactor actually has been adopted as the reference core in the current CRP project.
2. In determining fission product yield data, we adopt raw data from ENDF/B VI or ENDF/B VII or JENDL-3 data and by using actual spectrum of RBEC reactor we determine detail fission product yield data for the most important nuclides (has been selected in the second year CRP).
3. Investigating several alternatives fission product treatment in available cell homogenization code suitable for RBEC benchmark calculation.
4. Modifying/developing FP treatment procedure for the available cell calculation code.

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 and also the mathematical model description and the methodology of solution.

Chapter III : Discuss about RBEC Benchmark calculation results performed by ITB team

Chapter IV : Fission yield calculation results and discussion

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 AND MATHEMATICAL MODEL

Accurate FP treatment need several complicated processes and reliable data to achieve that. Among the important data are accurate fission product yield data for the relevant nuclear reactor core to be investigated. Basic FP yield data are available in ENDF/B or JENDL libraries but the data depend on the energy spectrum and they include more than 1200 nuclides of FP.

II.1 RESEARCH METHODOLOGY

From the first and second years results, we conclude that we can take onl 118 or even 50 most important nuclides and treat them accurately to get reasonably accurate FP group constants for neutronic calculations. These nuclides are shown in the following table.

Table 1 118 Most Important FP Nuclides

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

The rigorous processes for the FP treatment are very complicated and need some iteration processes. As an illustration, we need neutron energy spectrum in order to determine the practical FP yield data but to get accurate neutron energy spectrum we need accurate FP data. Therefore we adopt the following approximation processes for the FP treatment in this study.

1. The available FP treatment methods available in SRAC code system are investigated. Then based on the first and second year results we select the most appropriate method to be used in the neutron spectrum generation.
2. The RBEC nuclear reactor core is adopted as a reference core and then neutronic analysis (multi group diffusion calculation and burn-up calculation) are performed. From the RBEC neutronic calculation then the overall neutron energy spectrum is generated.
3. By selecting 118 most important FP nuclides and using basic FP data from JENDL-3.3 libraries then the independent and cumulative fission product yield for each nuclide and each energy group can be generated. Next by using neutron energy spectrum from the RBEC reactor then the independent and cumulative fission product yield for each nuclide can be calculated.
4. The data of fission product yield and the relevant decay chain should be used to modify or creating new sub library in the cell calculation code. But this step is postpone for the following year's research due to relatively large volume of works during implementation.

II.2 MATHEMATICAL MODEL

Detail mathematical model of the calculations can be breakdown into several part as follows.

II.2.1 Steady State Multigroup Diffusion Calculation

Mathematical formulation of steady state multi group diffusion calculation can be written as follows

$$-\vec{\nabla}\bullet D_g\vec{\nabla}\Psi_g(\vec{r},t)+\Sigma_{rg}\,\Psi_g(\vec{r},t)=\frac{x_g}{k_{eff}}\sum_{g'=1}^{G}\Sigma_{fg'}\,\Psi_{g'}(\vec{r},t)+\sum_{g'=1}^{G}\Sigma_{sg'\to g}\,\Psi_{g'}(\vec{r},t) \qquad (1)$$

Where:

g : energy group

D : diffusion constant

Σ_r : macroscopic cross section of removal

Σ_s : macroscopic cross section of scattering

Σ_f : macroscopic cross section of fission

v : average neutron number produced in fission

ϕ : neutron flux

χ_g : fission spectrum of energy group g

k_{eff} : Effective multiplication factor

II.2.2 Effective FP yield data calculation

The effective FP yield data can be calculated using linear interpolation of energy dependency as follows:

$$y_{i,g}^{n,m} = \frac{(u_{g,av} - u_{high})y_{i,g,high}^{n,m} + (u_{low} - u_{g,av})y_{i,g,low}^{n,m}}{u_{low} - u_{high}}$$

Where

$y_{i,g}^{n,m}$: independent fission yield of nuclide n for energy group g from fissile nuclide m

$y_{i,g,high}^{n,m}$: independent fission yield for nuclide n from fissile nuclide m at nearest upper energy of

energy group g (from ENDF/B or JENDL library)

$y_{i,g,low}^{n,m}$: independent fission yield for nuclide n from fissile nuclide m at nearest lower energy of

energy group g (from ENDF/B or JENDL library)

$u_{g,av}$: average lethargy for energy group g

$u_{g,high}$: lethargy of nearest higher energy of energy group g

$u_{g,low}$: lethargy of nearest lower energy of energy group g

Similarly, for cumulative fission yield we have

$$y_{c,g}^{n,m} = \frac{(u_{g,av} - u_{high})y_{c,g,high}^{n,m} + (u_{low} - u_{g,av})y_{c,g,low}^{n,m}}{u_{low} - u_{high}}$$

Where

$y_{c,g}^{n}$: cumulative fission yield of nuclide n for energy group g

$y_{c,g,high}^{n}$: cumulative fission yield for nuclide n at nearest upper energy of energy group g (from

ENDF/B or JENDL library)

$y_{c,g,low}^{n}$: cumulative fission yield for nuclide n at nearest lower energy of energy group g (from

ENDF/B or JENDL library)

The total FP yield data for nuclide n is given as

$$y_i^n = \frac{\sum_{g=1}^{G}\sum_{m} y_{i,g}^{n,m} N_m \sigma_{f,g,m} \phi_g}{\sum_{g=1}^{G}\sum_{m} N_m \sigma_{f,g,m} \phi_g} \qquad \text{and}$$

$$y_c^n = \frac{\sum_{g=1}^{G}\sum_{m} y_{c,g}^{n,m} N_m \sigma_{f,g,m} \phi_g}{\sum_{g=1}^{G}\sum_{m} N_m \sigma_{f,g,m} \phi_g}$$

Where

N_m : atomic density of fissile nuclide m

$\sigma_{f,g,m}$: microscopic fission cross section of nuclide m for energy g

φ_g : neutron flux at energy g

CHAPTER III

RBEC BENCHMARK CALCULATION RESULTS AND DISCUSSION

In the RBEC Benchmark calculation we adopt the data of RBEC from the reference 3 released by Kurchatov Research Institute. We use SRAC Code system combined with FI-ITB-CH1 code system to perform multigroup diffusion and burn-up calculation. The calculation were performed mainly in 25 energy groups though several 8 group and 33 energy group calculation were also performed for comparison.

The K eff (effective multiplication factor results is shown in the Table III.1 as follows.
It seems that effective multiplication is within the range of RBEC Benchmark calculation results of all participans. However if we compare with the the results other participans we can conclude that the burnup results tend to give higher value and it may caused by under estimate of FP grup constants.

Table.III.1 Evolution of keff on 1800-day campaign at thermal power 900 MW

Participant	ANL		BARC	Gidropress		RRC KI	TokyoTech	ITB
Time, day	DIF3D	TWODANT	ERANOS2.0	DIFRZ	KINRZ	MCNP5	Original	SRAC-FI-ITB-CH1
0	0.99721	0.99777	0.99498	1.0076	1.0084	1.00383	1.00271	1.00181
100	1.00011	1.00063	0.99544	1.0077	1.0085		1.00261	
200	1.00284	1.00333	0.99671	1.0086	1.0094		1.00322	
300	1.00539	1.00586	0.99785	1.0093	1.0102		1.00374	1.00567
400	1.00778	1.00822	0.99887	1.0100	1.0109		1.00418	
500	1.00999	1.01042	0.99977	1.0106	1.0114		1.00453	
600	1.01203	1.01246	1.00054	1.0110	1.0120		1.00479	1.00936
700	1.01392	1.01435	1.00121	1.0114	1.0124		1.00497	
800	1.01565	1.01608	1.00176	1.0117	1.0127		1.00505	
900	1.01724	1.01767	1.00221	1.0119	1.0128	1.01224	1.00505	1.01198
1000	1.01934	1.01912	1.00254	1.0120	1.0131		1.00497	
1200	1.02183	1.02162	1.00292	1.0120	1.0132		1.00457	1.01355
1400	1.02382	1.02363	1.00292	1.0117	1.0128		1.00387	
1600	1.02535	1.02519	1.00258	1.0111	1.0122		1.00289	1.01417*
1800	1.02647	1.02635	1.00192	1.0102	1.0113	1.01024	1.00166	1.01393

* at 1500 days

Next the energy spectrum is given in Fif III.1. The energy spectrum is relatively hard.

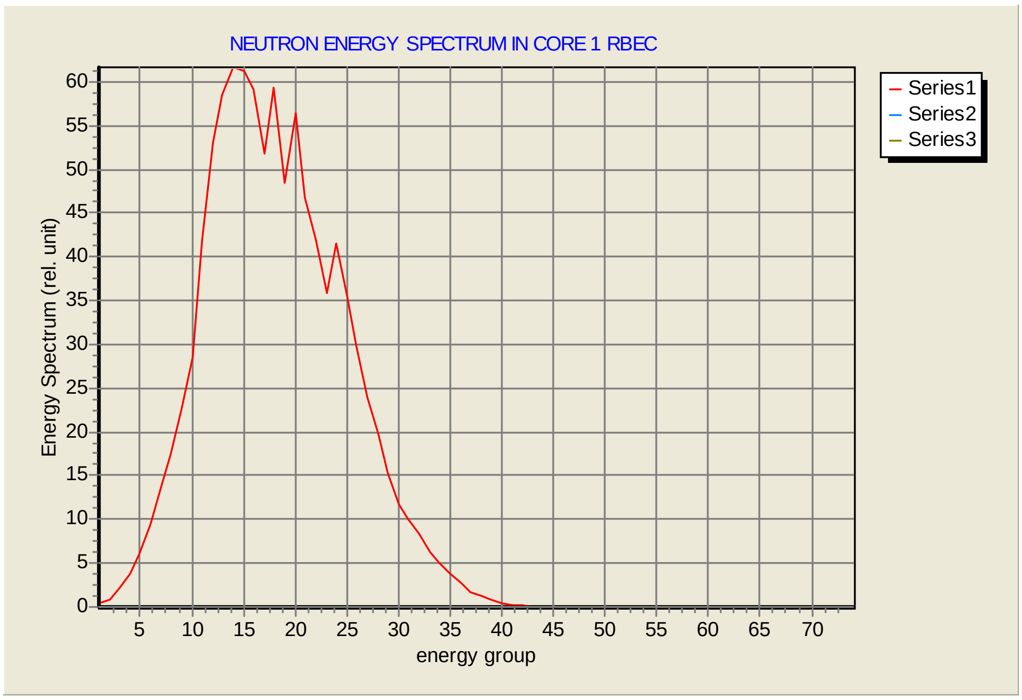

Fig III.1 Energy spectrum of Core 1 in RBEC Core

Fig. III.2 and Fig. III.3 show the infinite multiplication constant and conversion ratio in core 1 of RBEC core. It is shown that the infinite multiplication factor is increases up to nearly 800 days of operation and after that start to decrease. Fig III.3 shows the conversion ratio in Core I which continuously decrease during burn-up process.

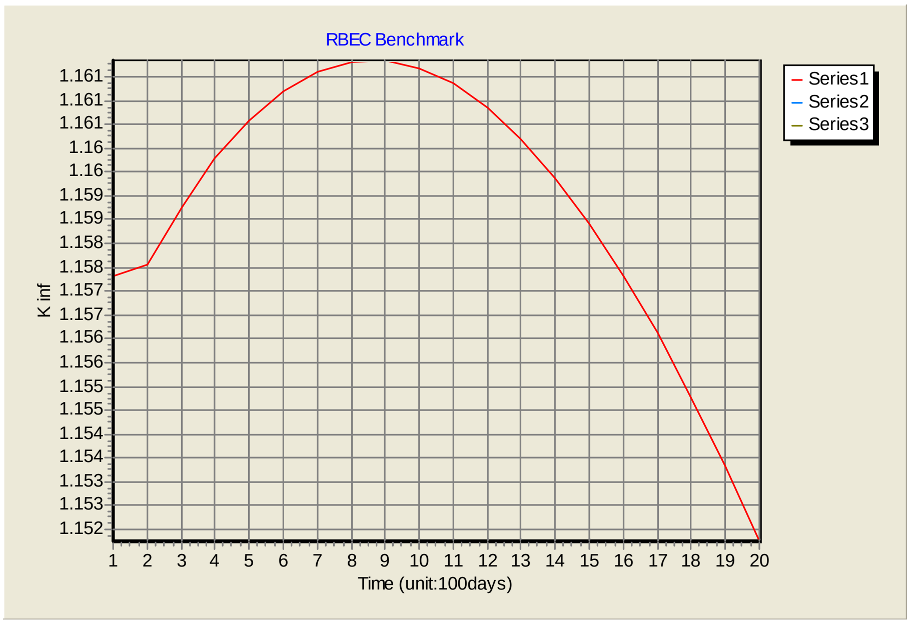

Fig. III.2 K inf change during burnup of Core I in RBEC Reactor

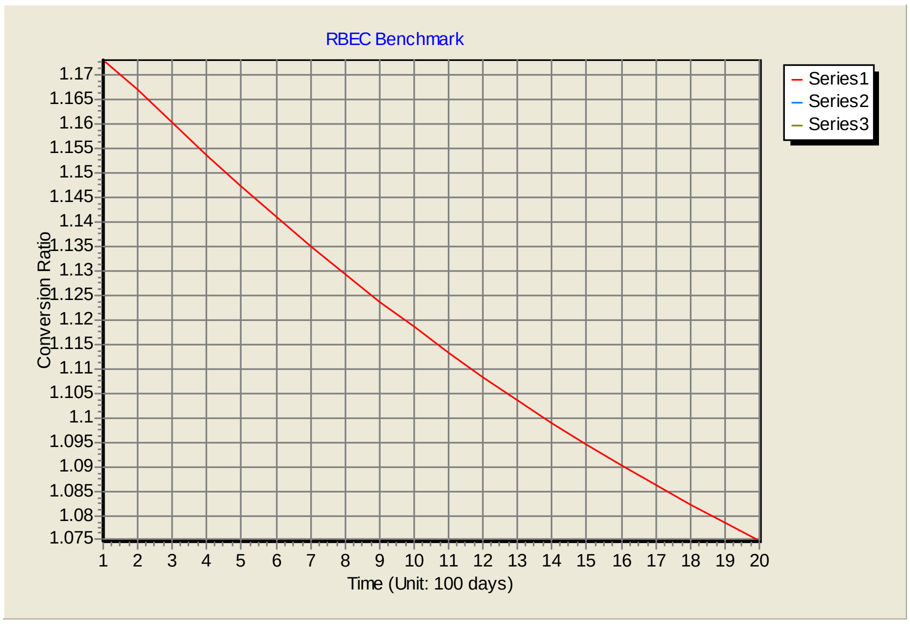

Fig. III.3 Conversion ratio change during burnup of Core I in RBEC Reactor

Next Figs. III. 4 to III.8 show the atomic density change during burnup of U-235, U-238, Pu-239, Pu-240 and Pu-241.

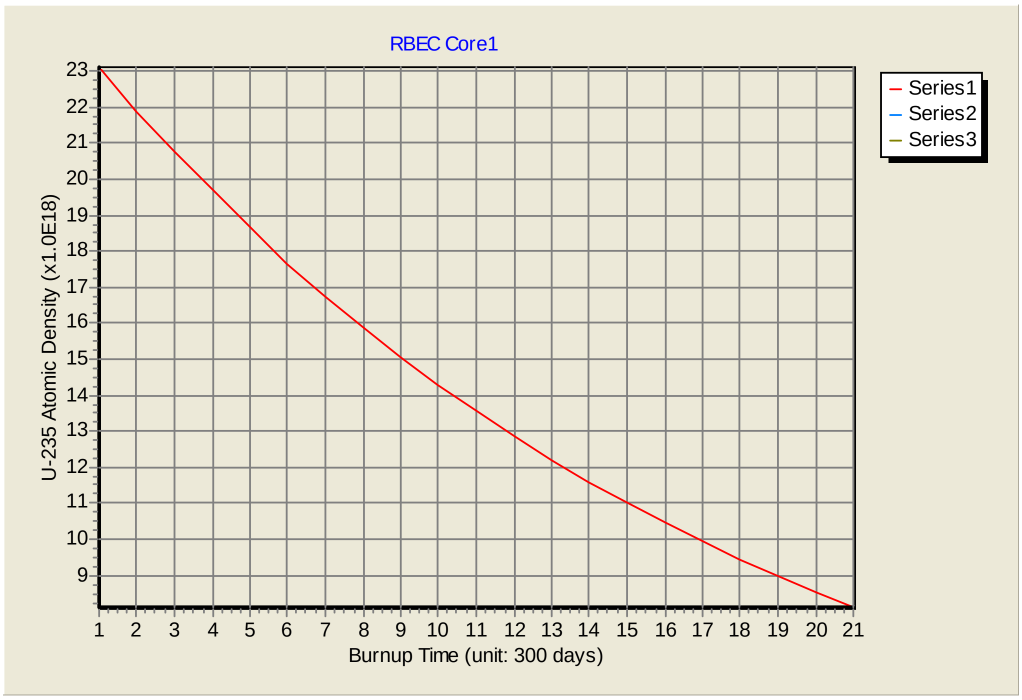

Fig. III.4 Atomic density change during burn-up of U-235

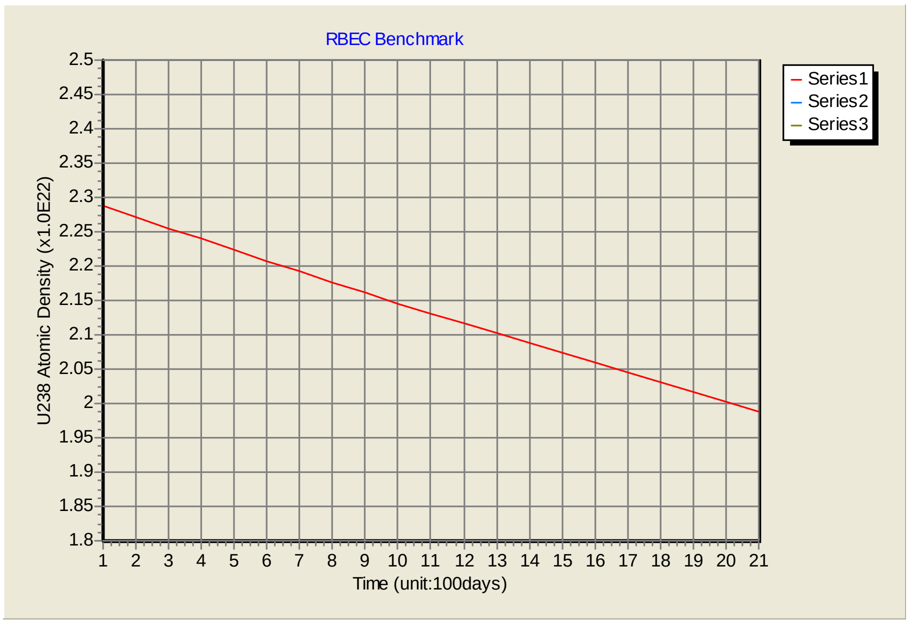

Fig. III.5 Atomic density change during burn-up of U-238

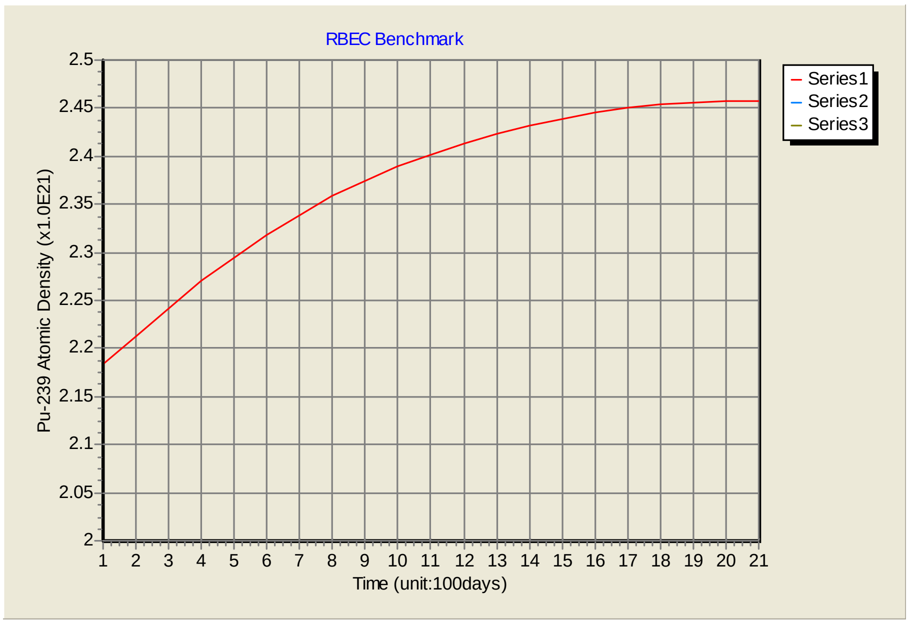

Fig. III.6 Atomic density change during burn-up of Pu-239

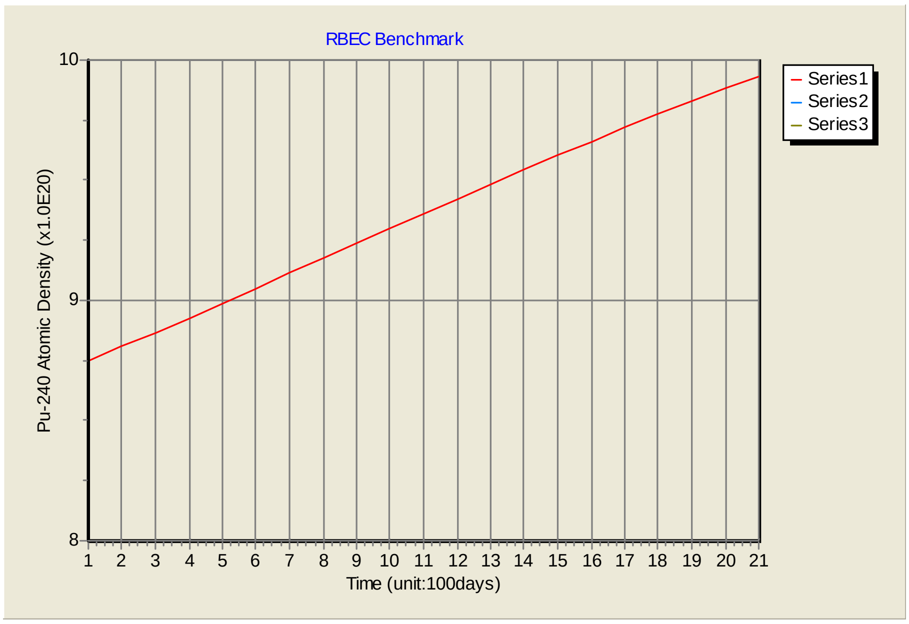

Fig. III.7 Atomic density change during burn-up of Pu-240

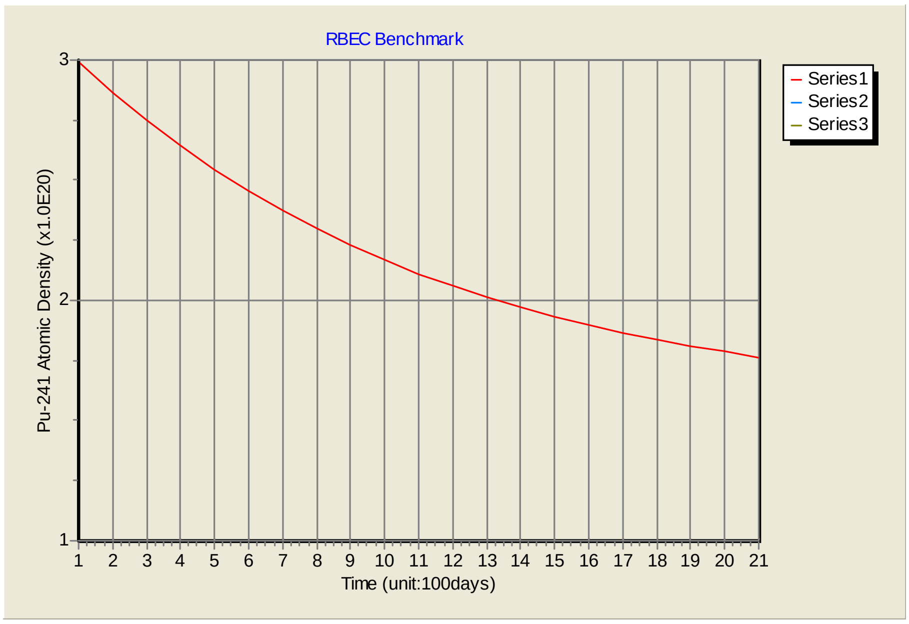

Fig. III.8 Atomic density change during burn-up of Pu-241

It is shown that the atomic density of U-235, U-238 and Pu-241 decreases continuously during burn-up process while for Pu-239 and Pu-240 increases continuously.

Next Figs. III.9 to III.12 show the radial and axial power distribution at BOL and EOL. Comparing the BOL and EOL data, we found that the peak power density is slightly decreases from BOL to EOL.

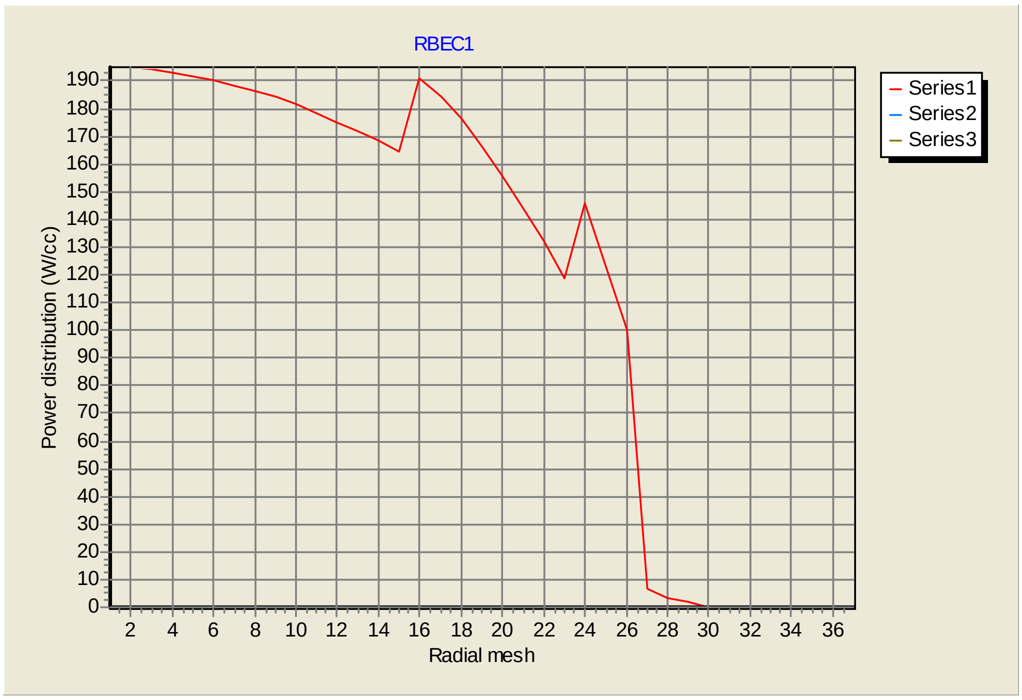

Fig. III.9 Radial power distributon at the BOL

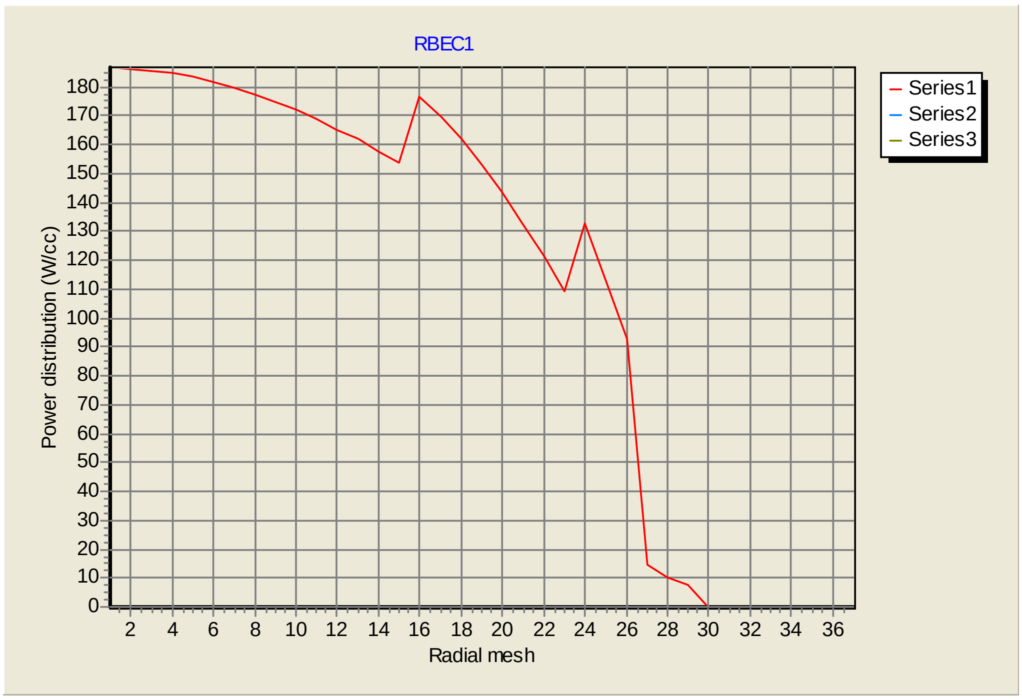

Fig. III.10 Radial power distributon at the EOL

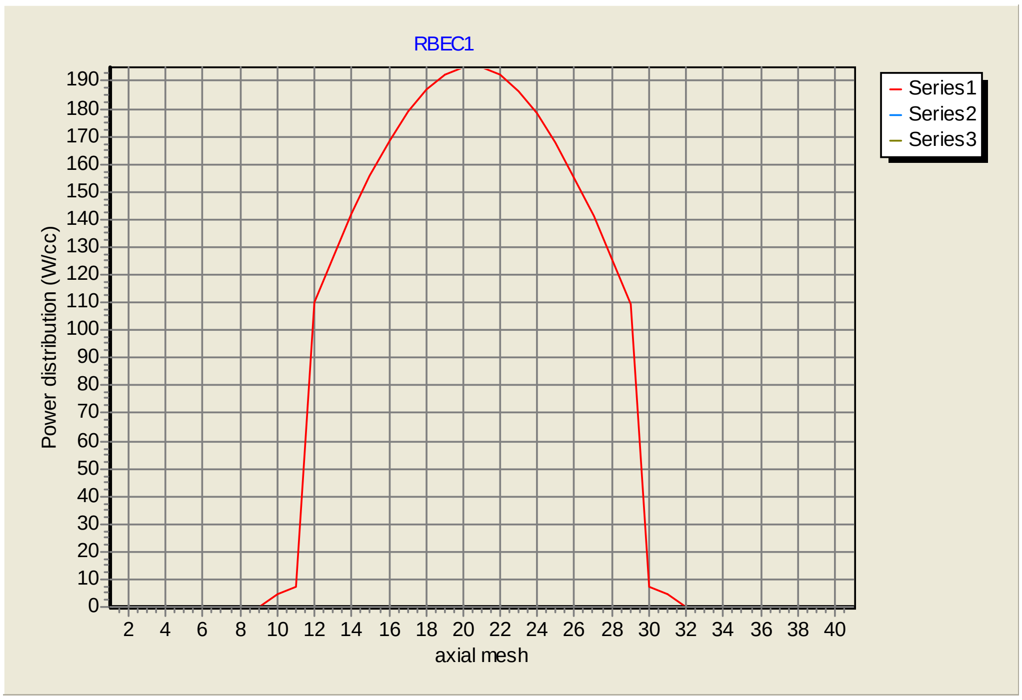

Fig. III.11 Axial power distributon at the BOL

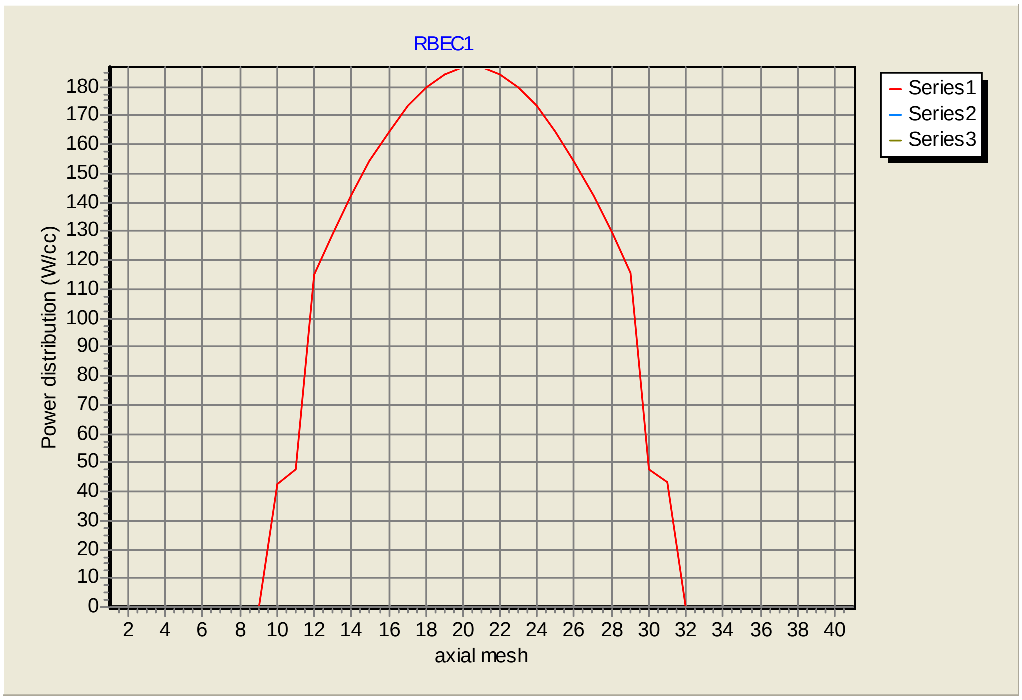

Fig. III.12 Axial power distributon at the EOL

CHAPTER IV

FISSION YIELD CALCULATION RESULTS AND DISCUSSION

To get accurate FP group constant data we first need to get accurate fission yield data which actually depend on the neutron spectrum and composition of the core
We adopt the neutron spectrum from the RBEC Benchmark calculation results and we generate also 74 groups microscopic cross section. The data is given partially as follows (for Pu-239 and U-238)

Table IV.1 Plutonium 239 microscopic cross section Data

Production (from group 1 to group 74)

```
  9.34858E+00  7.89452E+00  6.15262E+00  6.16446E+00  6.19863E+00  6.25174E+00
  6.27072E+00  6.03524E+00  5.56326E+00  5.12750E+00  4.84830E+00  4.64733E+00
  4.51778E+00  4.44010E+00  4.37884E+00  4.23212E+00  4.20770E+00  4.32332E+00
  4.37923E+00  4.43568E+00  4.52773E+00  4.70196E+00  4.77838E+00  4.82677E+00
  5.02046E+00  5.22467E+00  5.35475E+00  5.56610E+00  6.52894E+00  6.40597E+00
  7.01525E+00  8.51804E+00  1.02752E+01  8.98960E+00  1.02354E+01  1.40376E+01
  1.92288E+01  1.65625E+01  1.20145E+01  4.39779E+01  2.17844E+01  3.83696E+01
  4.57022E+01  5.29285E+01  5.28367E+01  6.00174E+01  1.30915E+02  1.84452E+02
  1.83225E+02  4.81624E+01  1.06564E+01  5.26357E+01  1.69302E+02  3.18329E+02
  4.97308E+02  4.06099E+01  2.56340E+02  2.07256E+01  2.15301E+01  2.57463E+01
  3.21937E+01  4.12972E+01  5.04401E+01  5.83068E+01  6.80560E+01  8.04208E+01
  9.65179E+01  1.18164E+02  1.48440E+02  1.92920E+02  2.62123E+02  3.78291E+02
  5.93324E+02  1.04493E+03
```

fission(from group 1 to group 74)

```
  2.24786E+00  2.03246E+00  1.68215E+00  1.77109E+00  1.85435E+00  1.93301E+00
  1.99121E+00  1.95734E+00  1.83223E+00  1.70923E+00  1.63132E+00  1.57528E+00
  1.54018E+00  1.52055E+00  1.50484E+00  1.45842E+00  1.45313E+00  1.49557E+00
  1.50817E+00  1.52156E+00  1.55313E+00  1.61290E+00  1.63912E+00  1.65572E+00
  1.72216E+00  1.79221E+00  1.83683E+00  1.90933E+00  2.23962E+00  2.19743E+00
  2.40643E+00  2.92194E+00  3.52470E+00  3.08555E+00  3.51451E+00  4.81390E+00
  6.59224E+00  5.68041E+00  4.12309E+00  1.51196E+01  7.56994E+00  1.33916E+01
  1.58967E+01  1.83332E+01  1.82981E+01  2.09918E+01  4.55462E+01  6.39542E+01
  6.36574E+01  1.69763E+01  3.74113E+00  1.83922E+01  5.90000E+01  1.10830E+02
  1.73144E+02  1.41393E+01  8.92500E+01  7.21610E+00  7.49620E+00  8.96417E+00
  1.12090E+01  1.43786E+01  1.75619E+01  2.03009E+01  2.36954E+01  2.80005E+01
  3.36051E+01  4.11418E+01  5.16831E+01  6.71712E+01  9.13016E+01  1.31844E+02
  2.06920E+02  3.64601E+02
```

absorbtion(from group 1 to group 74)

```
  2.24786E+00  2.03250E+00  1.68262E+00  1.77332E+00  1.86120E+00  1.94777E+00
  2.01480E+00  1.98992E+00  1.87619E+00  1.76744E+00  1.70837E+00  1.67413E+00
  1.66580E+00  1.67377E+00  1.68783E+00  1.66263E+00  1.67978E+00  1.74981E+00
  1.78926E+00  1.83495E+00  1.90773E+00  2.01827E+00  2.11016E+00  2.22409E+00
  2.39773E+00  2.58996E+00  2.75977E+00  2.95446E+00  3.57279E+00  3.86765E+00
  4.40508E+00  5.13396E+00  6.25147E+00  6.57487E+00  7.12671E+00  8.47130E+00
  1.08794E+01  9.94453E+00  1.01007E+01  2.33848E+01  1.17760E+01  2.89221E+01
  3.33638E+01  2.70668E+01  3.60727E+01  4.23721E+01  7.06393E+01  9.03581E+01
  1.14581E+02  7.09028E+01  7.23883E+00  3.62580E+01  1.08308E+02  1.93823E+02
  2.59456E+02  2.05805E+01  1.60571E+02  8.77570E+00  8.91208E+00  1.06383E+01
```

```
1.33451E+01  1.72270E+01  2.11867E+01  2.46502E+01  2.90167E+01  3.46644E+01
4.21910E+01  5.25810E+01  6.75359E+01  9.01693E+01  1.26503E+02  1.89304E+02
3.08577E+02  5.64531E+02
```

Table IV. 2 Uranium 238 microscopic cross section data

production(from group 1 to group 74)

```
3.59261E+00  2.85853E+00  1.69501E+00  1.63642E+00  1.46603E+00  1.44808E+00
1.38274E+00  9.05580E-01  1.24873E-01  3.47321E-02  6.41119E-03  1.69603E-03
7.45663E-04  4.16915E-04  2.21915E-04  2.46912E-04  2.21606E-04  1.05267E-04
8.85272E-05  1.06715E-04  2.22668E-04  1.93396E-04  8.86315E-05  1.61720E-04
2.50035E-04  3.06804E-04  2.67529E-04  1.61223E-04  3.54022E-04  9.55655E-05
2.45278E-08  5.92992E-09  2.52117E-09  5.69438E-09  3.42308E-08  8.58370E-07
2.50188E-03  7.12038E-05  8.99161E-03  6.11080E-05  2.16101E-06  1.26661E-05
3.77980E-06  1.54534E-05  4.58149E-07  3.90870E-06  1.34037E-05  2.65009E-05
9.39137E-07  1.39682E-06  1.74854E-05  2.91854E-06  1.63430E-04  3.52780E-06
1.62783E-06  8.73717E-07  2.79797E-05  1.75910E-05  5.00494E-06  4.11678E-06
3.97294E-06  4.08378E-06  4.26041E-06  4.41656E-06  4.59696E-06  4.80308E-06
5.03116E-06  5.28459E-06  5.55806E-06  5.85692E-06  6.17911E-06  6.52979E-06
6.90398E-06  7.30729E-06
```

fission(from group 1 to group 74)

```
9.89288E-01  8.49578E-01  5.40464E-01  5.58681E-01  5.31115E-01  5.42566E-01
5.25744E-01  3.48190E-01  4.84516E-02  1.35840E-02  2.52143E-03  6.70287E-04
2.95764E-04  1.65826E-04  8.84728E-05  9.86038E-05  8.86089E-05  4.21360E-05
3.54621E-05  4.27788E-05  8.92990E-05  7.75865E-05  3.55678E-05  6.49148E-05
1.00381E-04  1.23189E-04  1.07430E-04  6.47435E-05  1.42098E-04  3.83743E-05
9.85857E-09  2.38026E-09  1.01370E-09  2.28513E-09  1.37683E-08  3.44373E-07
1.00489E-03  2.85982E-05  3.61195E-03  2.45557E-05  8.67961E-07  5.08859E-06
1.51800E-06  6.20817E-06  1.84069E-07  1.57010E-06  5.38409E-06  1.06454E-05
3.77271E-07  5.61112E-07  7.02408E-06  1.17243E-06  6.56552E-05  1.41719E-06
6.53933E-07  3.50980E-07  1.12398E-05  7.06666E-06  2.01058E-06  1.65379E-06
1.59601E-06  1.64053E-06  1.71149E-06  1.77422E-06  1.84669E-06  1.92949E-06
2.02111E-06  2.12292E-06  2.23278E-06  2.35284E-06  2.48227E-06  2.62314E-06
2.77346E-06  2.93548E-06
```

absorbtion(from group 1 to group 74)

```
9.93108E-01  8.54941E-01  5.48016E-01  5.69321E-01  5.46595E-01  5.65828E-01
5.61308E-01  4.04509E-01  1.41483E-01  1.39137E-01  1.23189E-01  1.13153E-01
1.08547E-01  1.12391E-01  1.19378E-01  1.28180E-01  1.40555E-01  1.55619E-01
1.80945E-01  2.17729E-01  2.72410E-01  3.43239E-01  3.92940E-01  4.38121E-01
4.85724E-01  5.37096E-01  5.83438E-01  6.12357E-01  6.07322E-01  7.54261E-01
7.74494E-01  7.48628E-01  9.29823E-01  9.43291E-01  9.49739E-01  9.33879E-01
1.30843E+00  1.48940E+00  1.54707E+00  1.56580E+00  1.30923E+00  1.43097E+00
2.03598E+00  2.45197E+00  1.52528E+00  5.27991E+00  2.65210E+00  4.75081E+00
1.42402E-01  2.05019E+00  1.45377E+01  7.20227E-01  2.76584E+01  4.50570E-01
3.40654E-01  8.70034E-01  2.94078E+01  9.49631E+00  1.12848E+00  6.15580E-01
5.01802E-01  4.67792E-01  4.64516E-01  4.69767E-01  4.79177E-01  4.92164E-01
5.08295E-01  5.27538E-01  5.49249E-01  5.73803E-01  6.00910E-01  6.30980E-01
6.63501E-01  6.98930E-01
```

We then investigated the contribution of each fissile and fissionable material to the total fission but we got that the Pu-239 was dominant do we focus do process the fission yield data based on Pu-239 fissile data

Next we calculate the fission yield produced by fission in each energy group for each important nuclide. The example of the results is given in Table IV.3 for Ru-101.

Table IV.3 Fisssion yield data for Ru-101 from each group fission

G	Independent	Cumulative
1	1.349347E-08	6.282428E-02
2	1.569870E-08	6.142716E-02
3	1.790394E-08	6.003004E-02
4	2.010917E-08	5.863292E-02
5	2.231440E-08	5.723580E-02
6	2.451964E-08	5.583868E-02
7	2.672487E-08	5.444156E-02
8	2.893010E-08	5.304445E-02
9	3.113534E-08	5.164733E-02
10	1.328823E-08	5.900539E-02
11	1.323276E-08	5.909721E-02
12	1.317729E-08	5.918903E-02
13	1.312181E-08	5.928085E-02
14	1.306634E-08	5.937266E-02
15	1.301087E-08	5.946448E-02
16	1.295539E-08	5.955630E-02
17	1.289992E-08	5.964812E-02
18	1.284445E-08	5.973993E-02
19	1.278897E-08	5.983175E-02
20	1.273350E-08	5.992356E-02
21	1.267803E-08	6.001538E-02
22	1.262255E-08	6.010720E-02
23	1.256708E-08	6.019901E-02
24	1.251161E-08	6.029083E-02
25	1.245613E-08	6.038265E-02
26	1.240066E-08	6.047447E-02
27	1.234519E-08	6.056628E-02
28	1.228971E-08	6.065810E-02
29	1.223424E-08	6.074992E-02
30	1.217877E-08	6.084174E-02
31	1.212329E-08	6.093355E-02
32	1.206782E-08	6.102537E-02
33	1.201235E-08	6.111719E-02
34	1.195687E-08	6.120900E-02
35	1.190140E-08	6.130082E-02
36	1.184593E-08	6.139264E-02
37	1.179046E-08	6.148445E-02
38	1.173498E-08	6.157627E-02
39	1.167951E-08	6.166809E-02
40	1.162404E-08	6.175990E-02
41	1.156856E-08	6.185172E-02
42	1.151309E-08	6.194354E-02
43	1.145762E-08	6.203536E-02
44	1.140214E-08	6.212717E-02
45	1.134667E-08	6.221899E-02
46	1.129120E-08	6.231081E-02
47	1.123572E-08	6.240262E-02
48	1.118025E-08	6.249444E-02

49	1.112478E-08	6.258626E-02
50	1.106930E-08	6.267808E-02
51	1.101383E-08	6.276989E-02
52	1.095836E-08	6.286171E-02
53	1.090288E-08	6.295352E-02
54	1.084741E-08	6.304534E-02
55	1.079194E-08	6.313716E-02
56	1.073647E-08	6.322897E-02
57	1.068099E-08	6.332079E-02
58	1.062552E-08	6.341261E-02
59	1.057005E-08	6.350443E-02
60	1.051457E-08	6.359624E-02
61	1.045910E-08	6.368806E-02
62	1.040363E-08	6.377988E-02
63	1.036202E-08	6.384874E-02
64	1.033428E-08	6.389465E-02
65	1.030655E-08	6.394055E-02
66	1.027881E-08	6.398647E-02
67	1.025107E-08	6.403238E-02
68	1.022334E-08	6.407828E-02
69	1.019560E-08	6.412419E-02
70	1.016786E-08	6.417010E-02
71	1.014013E-08	6.421601E-02
72	1.011239E-08	6.426191E-02
73	1.008465E-08	6.430782E-02
74	1.005692E-08	6.435373E-02

Finally we generated the fission yield data for each important nuclide as shown in Table IV.4.

Table IV.4 Fission Yield Data for Important Nuclides

No.	Nuclide	Independent	Cumulative
1	44101	1.39488412E-08	0.0594559051
2	46105	7.24691418E-11	0.0528501496
3	43099	8.94911238E-08	0.0598406382
4	45103	1.10945884E-10	0.0682032108
5	55133	1.2921422E-06	0.0686191767
6	46107	5.61074742E-08	0.0328232199
7	42097	8.35450635E-07	0.0536595806
8	62149	2.97196734E-09	0.0122768953
9	61147	1.76317592E-08	0.0201525874
10	60145	9.16564176E-08	0.0293119382
11	55135	0.000143690457	0.
12	60143	5.13469302E-11	0.0428884849
13	54131	8.18919432E-07	0.0390982628
14	44102	4.00069894E-07	0.0604525246
15	62151.	1.20191066E-06	0.00770399207
16	42095	4.79276396E-10	0.0479308628
17	42098.	1.52272778E-05	0.0569780767
18	47109	1.54989905E-08	0.0192497894
19	44104	5.83728724E-05	0.0599739552
20	42100	0.00116378581	0.0667999163

21	63153	1.04220319E-07	0.00380806136
22	40093	1.12774869E-05	0.038283091
23	44103	6.54340784E-06	0.0682032108
24	59141	6.37891129E-10	0.0513297245
25	53129	9.84646067E-06	0.015332452
26	40095	0.00111476786	0.0479257517
27	40096	0.00546665257	0.049789656
28	60146	2.0546679E-06	0.0241217129
29	54132	5.22244482E-05	0.0537467673
30	46108	1.02779777E-05	0.0214385558
31	41095	4.21908271E-06	0.0479200929
32	58141	4.06300796E-06	0.0513297245
33	40091	3.29098562E-08	0.0246609803
34	40092	2.57482975E-06	0.0299894158
35	54134	0.00104259967	0.0746262074
36	44106	0.00478176028	0.0425091088
37	62152	1.12275375E-05	0.00586442789
38	60148	0.000230132951	0.0161371008
39	48111	9.53392365E-10	0.
40	37085	3.76806099E-07	0.00596903265
41	53127	6.72185649E-08	0.
42	57139	1.51741406E-05	0.
43	46106	5.20088994E-09	0.0425161459
44	63155	6.24127051E-06	0.
45	40094	0.000139707321	0.0433628
46	62147	7.33138702E-13	0.0201525874
47	58142	5.83158871E-05	0.0482551605
48	60150	0.00257304939	0.00967459101
49	60147	3.15005091E-05	0.0201525111
50	55137	0.00723981159	0.0654700324
51	39091	3.37635788E-06	0.024660904
52	60144	2.12788609E-09	0.0365167931
53	36083	3.97525184E-07	0.00309072481
54	58144	0.00190069433	0.0365113653
55	64157	5.38209179E-07	0.000826512056
56	46110	0.000235285333	0.00678964658
57	42099	0.00016201062	0.0598405376
58	64156	8.1606693E-08	0.00128967955
59	48113	8.94998422E-08	0.00153323298
60	55134	1.26644045E-05	2.84286089E-05
61	63154	5.24785719E-07	1.09989969E-06
62	58140	2.19996153E-07	0.0540958382
63	51125	9.95659284E-05	0.00240752683
64	65159	3.34624417E-08	0.000259730499
65	62154	0.000204086493	0.00275957375
66	38090	0.000506891985	0.0211175736
67	53131	0.00060731516	0.0390940532
68	39089	6.96272018E-09	0.0174264144
69	56138	0.00062428799	0.0590804704
70	59143	4.51386285E-07	0.0428884849
71	35081	7.70846214E-07	0.
72	52130	0.00385167915	0.0249443837
73	49115	4.8210044E-08	0.00102181779
74	52128	0.000155515387	0.00909668952

75	48112	7.40597983E-08	0.0022579045
76	52129	0.000610249641	0.00200529373
77	37087	6.0477927E-05	0.0101353796
78	36084	1.00259203E-05	0.00506843394
79	54133	0.000174431727	0.0686179176
80	51121	1.36808509E-07	0.0013621588
81	52127	2.26742177E-05	0.000875361147
82	61148	2.81923462E-07	2.81923462E-07
83	34079	8.83708481E-07	0.000518747373
84	45105	5.52293386E-07	0.0528501496
85	62150	8.83653115E-08	2.21177816E-05
86	51123	1.0363422E-05	0.00160690665
87	64155	8.9122576E-09	0.00179360155
88	50117	1.08045128E-09	0.00131984882
89	61149	5.79242806E-06	0.0122768888
90	54136	0.0305194221	0.066190742
91	46104	8.52468215E-13	2.35028459E-08
92	64158	2.78086827E-06	0.00048069976
93	44100	3.88423099E-10	2.12777377E-06
94	36085	8.35864266E-05	0.00133366848
95	38089	5.34944957E-05	0.0174264014
96	48114	3.77324818E-06	0.00144537922
97	38088	4.13364023E-06	0.0140417218
98	50119	1.50662146E-07	0.00122217531
99	62148	6.33070263E-11	3.86537693E-07
100	34082	0.000532393111	0.00213768682
101	56136	2.21158416E-06	0.00130924035
102	47110	7.73662862E-07	7.73662862E-07
103	34077	5.86704435E-11	9.47328372E-05
104	36086	0.000686592888	0.00788113568
105	63156	2.09438167E-05	0.00128960051
106	34080	1.54168374E-05	0.0011528678
107	63151	2.19690613E-10	0.00770399207
108	48116	8.04563824E-05	0.00124024425
109	50118	1.39797379E-07	0.00121764641
110	48110	3.5894171E-10	9.93145591E-07
111	34078	3.40994966E-08	0.000300992629
112	54130	2.62221761E-07	0.000116128431
113	56137	2.25803033E-05	0.0655588508
114	64160	2.04950757E-05	0.000141239143
115	56140	0.010802621	0.0539481267
116	50126	0.00207613781	0.0036603976
117	52125	1.0644932E-07	0.00240827538
118	50120	6.40889766E-06	0.00119782588

From the obtained data we found that to get highly accurate FP treatment in burn-up calculation we should modify the fission yield and burn-up chain in the cell burn-up calculation. In the present study we select UCM66 as burnup-chain model in SRAC due to the consideration the nearest chain to the results of our second year study. WE then use the spectrum and cross section to generate fission yield data along with JENDL library data.

In the next study we should consider iteration in which we modif burnup chain data and feedback the fission yield data in the cell burnup model to get more accurate FP treatment during fast reactor burnup calculation.

CONCLUDING REMARK

In the present study we select UCM66 as burnup-chain model in SRAC due to the consideration the nearest chain to the results of our second year study. We then use this scheme to calculate RBEC Benchmark.

Frm the main parameter comparison such as effective multiplication constant we found that our RBEC benchmark results in the range of RBEC Benchmark calculation results of all participans. However if we compare with the the results other participans in detail we can conclude that the burn-up results tend to give higher value and it may caused by under estimate of FP grup constants.

We then use the spectrum and cross section to generate fission yield data along with JENDL library data. In the next study we should consider iteration in which we modif burnup chain data and feedback the fission yield data in the cell burnup model to get more accurate FP treatment during fast reactor burnup calculation.

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