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Fission Yield Calculation Method and its Effect in Nuclear Fuel Cell Homogenization Calculation

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Abstract

Zero burn-up core capability can eliminate possible super prompt critical accident and make possible of inherent safety feature based on reactivity feedback mechanism. In this concept the maximum excess reactivity is limited below 1 $ of reactivity so that possibility of super prompt accident such as in Chernobyl accident case can be eleminated. This is however need high quality of system analysis, and in this study the effect of fission yield calculation method on the nuclear fuel cell homogenization process is investigated and discussed. This study use SRAC code system to investigate the efect of reactor dependent fission yield distribution calculation. Calculation restults show that this process will has important improvement effect for long life high burnup core.

Keywords : *Zero burn-up, Inherent safety, Fission yield, Neutron energy spectrum, Nuclear fuel cell homogenization*

1. Introduction

After chernobyl accident there are a strong shift in the nuclear power plant safety paradigm which shifts toward inherent/safety paradigm. Pb/Pb-Bi cooled fast reactors are among new emerging next generation nuclear power plant with many advantages such as inherent safet capability against unprotected rod run out transient over power (UTOP) accident as well as unprotected loss of flow (ULOF) accident; ability to breed uranium 238 to be utilized efficiently, ability to burn nuclear waste, and economically competitive[1-8)].

One of important feature of lead/lead-bismuth cooled fast reactors is the zero burn-up core capability which can eliminate possible super prompt critical accident and make possible of inherent safety feature based on reactivity feedback mechanism. In this concept the maximum excess reactivity is limited below 1 $ of reactivity so that possibility of super prompt accident such as in Chernobyl accident case can be eleminated. This is 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 this study the effect of fission yield calculation method on the nuclear fuel cell homogenization process is investigated and discussed. This development is important to support implementation process of zero burn-up concept in inherently safe fast reactors.

2. Mathematical Model and Calculational Method

In most of the nuclear fuel cell homogenization codes the fission yield distribution is calculated for several standard energy spectrum and during cell calculation process the user should select one of the nearest spectrum. In this study the fission yield calculation is recalculated according to the real reactor energy spectrum and the restults are compared to the previous common model.Detail mathematical model of the calculations can be divided into two parts as follows[9-12)].

2.1 Steady State Multigroup Diffusion Calculation

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

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

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

2.2 Effective FP yield data calculation

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

$$y_{i,g}^{n,m} = \frac{(u_{g,av} - u_{high}) y_{i,g,high}^{n,m} + (u_{low} - u_{g,av}) y_{i,g,low}^{n,m}}{u_{low} - u_{high}} \quad (2)$$

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

$$y_{c,g}^{n,m} = \frac{(u_{g,av} - u_{high})y_{c,g,high}^{n,m} + (u_{low} - u_{g,av})y_{c,g,low}^{n,m}}{u_{low} - u_{high}} \quad (3)$$

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

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

and

$$y_c^n = \frac{\sum_{g=1}^{G}\sum_{m} y_{c,g}^{n,m} N_m \sigma_{f,g,m}\phi_g}{\sum_{g=1}^{G}\sum_{m} N_m \sigma_{f,g,m}\phi_g} \quad (4)$$

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

3. Calculation Results and Discussion

In this study Pb-Bi cooled RBEC fast reactor is taken as a calculation model. We then investigated the contribution of each fissile and fissionable material to the total fission but we got that the Pu-239 was dominant so we focus to process the fission yield data based on Pu-239 fission. Next we calculate the fission yield produced by fission in each energy group for each important nuclide. The example of the results for Ru-101 is given in Table 1.

Then we generated the total (summed over energy groups) fission yield data for each important nuclide[12] which results are shown in Table 2.

Table 1. Fisssion yield data for Ru-101 from each group fission

Energy Group	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

Table 2. 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

Table 3. the effect of reactor core dependent fision yield calculation

Burnup Iteration (100 days)	Original	Modif first it.	Modif second it.
1	1.157302	1.157302	1.157302
2	1.157548	1.157578	1.157578
3	1.158775	1.158837	1.158837
4	1.159779	1.15987	1.159872
5	1.160584	1.160707	1.160707
6	1.161201	1.161357	1.161357
7	1.161612	1.161797	1.161797
8	1.161821	1.162039	1.162038
9	1.161843	1.162092	1.162092
10	1.161683	1.161962	1.161962
11	1.161348	1.161657	1.161656
12	1.160843	1.161183	1.161183
13	1.16018	1.160551	1.160551
14	1.159375	1.159774	1.159774
15	1.158419	1.158845	1.158846
16	1.157327	1.157782	1.157782
17	1.156122	1.156607	1.156607
18	1.154782	1.155295	1.155296
19	1.153327	1.15387	1.15387
20	1.151761	1.152329	1.152329

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. We then insert these fission yield data to modify UCM66 burn-up-chain model in SRAC[9] and then we recalculate the cell calculation. The results are shown in table 3. The first results(original) is just the calculation results using original UCM66 data. We then use the resulted spectrum to generate new fission yield distribution, modify the library and get modified first iteration result. The above process are repeated and the modified second iteration data.

From the above results we can see that the difference of Kinf calculation in Core 1 RBEC reactor after 2000 days is about $6x10^{-2}$ %dk/k or roughly 0.2 $ of reactivity for burnup level of 10%HM. His value is not significant in current conventional fast reactors, but for zero burnup core with higher burnup level the discrepancy may reach 0.5$ and it clearly significant for nuclear reactor safety consideration especially at the end of life cycle. From the above results also we can see that the effect of further iteration is not important. The effect becomes more important in the nuclear reactors which adopt fuel recycling system[13)].

4. Conclusion

The effect of fission yield calculation method on the nuclear fuel cell homogenization process has been investigated and discussed. This development is important to support implementation process of zero burn-up concept in inherently safe fast reactors. From the calculation results we can conclude that the difference of Kinf calculation in Core 1 RBEC reactor after 2000 days is about $6x10^{-2}$ %dk/k or roughly 0.2 $ of reactivity for burnup level of slightly less than%HM. This value is not significant in current conventional fast reactors, but for zero burnup core with higher burnup level the discrepancy may reach 0.5$ or more and it clearly significant for nuclear reactor safety consideration especially at the end of life cycle. The effect becomes more important in the nuclear reactors which adopt fuel recycling system.

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