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 eliminated. 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 effect of reactor dependent fission yield distribution calculation. Calculation results 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 safety 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 competitive1-8).

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 eliminated. 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 results are compared to the previous common model. Detail mathematical model of the calculations can be divided into two parts as follows9-12).

2.1 Steady State Multigroup Diffusion Calculation

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

\[
\frac{\nabla \cdot D_g \nabla \psi_g (\vec{r}, t)}{k_{\text{eff}}} + \sum_{g'} \psi_{g'} (\vec{r}, t) = \sum_{g'=1}^{G} \sum_{s'=1}^{S} \Sigma_{g' \rightarrow g} \psi_{g'} (\vec{r}, t) + \sum_{g'=1}^{G} \Sigma_{g' \rightarrow g} \psi_{g'} (\vec{r}, t)
\]

(1)

Where:
- \( g \) : energy group
- \( D \) : diffusion constant
- \( \Sigma_{v} \) : macroscopic cross section of removal
- \( \Sigma_{v} \) : macroscopic cross section of scattering
- \( \Sigma_{f} \) : macroscopic cross section of fission
- \( v \) : average neutron number produced in fission
- \( \phi \) : neutron flux
- \( \chi_g \) : fission spectrum of energy group \( g \)
- \( k_{\text{eff}} \) : 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:

\[
y_{g,m} = \frac{(u_{g,m} - u_{\text{high}}) y_{g,m,\text{high}} + (u_{\text{low}} - u_{g,m}) y_{g,m,\text{low}}}{u_{\text{low}} - u_{\text{high}}}
\]

(2)

where \( y_{g,m} \) independent fission yield of nuclide \( n \) for energy group \( g \) from fissile nuclide \( m \).
\[ Y_{i,g,\text{high}}^{n,m} \] independent fission yield for nuclide n from fissile nuclide m at nearest upper energy of energy group g (from ENDF/B or JENDL library);

\[ Y_{i,g,\text{low}}^{n,m} \] independent fission yield for nuclide n from fissile nuclide m at nearest lower energy of energy group g (from ENDF/B or JENDL library);

\[ u_{g,\text{av}} \] : average lethargy for energy group g

\[ u_{g,\text{high}} \] : lethargy of nearest higher energy of energy group g

\[ u_{g,\text{low}} \] : lethargy of nearest lower energy of energy group g

Similarly, for cumulative fission yield we have

\[
Y_{c,g}^{n,m} = \frac{(u_{g,\text{av}} - u_{g,\text{high}})Y_{c,g,\text{high}}^{n,m} + (u_{g,\text{low}} - u_{g,\text{high}})}{u_{g,\text{low}} - u_{g,\text{high}}}
\]  

(3)

where:

\[ Y_{c,g}^{n,m} \] : cumulative fission yield of nuclide n for energy group g

\[ Y_{c,g,\text{high}}^{n,m} \] : cumulative fission yield for nuclide n at nearest upper energy of energy group g (from ENDF/B or JENDL library)

\[ Y_{c,g,\text{low}}^{n,m} \] : 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

\[ Y_{f}^{n} = \sum_{g=1}^{G} \sum_{m} \sum_{\sigma_{f,g,m}} N_{m} \sigma_{f,g,m} \phi_{g} \]

and

\[
Y_{c}^{n} = \sum_{g=1}^{G} \sum_{m} \sum_{\sigma_{f,g,m}} N_{m} \sigma_{f,g,m} \phi_{g}
\]

(4)

where

\[ N_{m} \] : atomic density of fissile nuclide m

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

\[ \phi_{g} \] : 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 which results are shown in Table 2.

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

<table>
<thead>
<tr>
<th>Energy Group</th>
<th>Independent</th>
<th>Cumulative</th>
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<td>6</td>
<td>2.451964E-08</td>
<td>5.7351E-02</td>
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<td>7</td>
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<td>5.6333E-02</td>
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<tr>
<td>8</td>
<td>2.893010E-08</td>
<td>5.5306E-02</td>
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<tr>
<td>9</td>
<td>3.113534E-08</td>
<td>5.4279E-02</td>
</tr>
<tr>
<td>10</td>
<td>3.328823E-08</td>
<td>5.3252E-02</td>
</tr>
</tbody>
</table>

Table 2. Fission yield data for Ru-101 from each group fission
Table 2. Fission Yield Data for Important Nuclides

<table>
<thead>
<tr>
<th>Iteration</th>
<th>Original</th>
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<tr>
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<td>4</td>
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<td>1.161674</td>
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<tr>
<td>5</td>
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<td>1.161657</td>
<td>1.161656</td>
</tr>
<tr>
<td>6</td>
<td>1.160843</td>
<td>1.161183</td>
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<td>1.161657</td>
<td>1.161656</td>
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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 \(K_{\text{inf}}\) calculation in Core 1 RBEC reactor after 2000 days is about \(6 \times 10^{-2}\) %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 \(K_{\text{inf}}\) calculation in Core 1 RBEC reactor after 2000 days is about \(6 \times 10^{-2}\) %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.

References