Development of Burn-up Calculation System for Fusion-Fission Hybrid Reactor

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A fusion-fission hybrid reactor which is a fusion reactor with a blanket region containing nuclear fuel. In our group, a calculation system for analysis of fusion-fission hybrid reactor has been developed and transport and burnup calculations were carried out for various hybrid energy systems with three-dimensional ITER model. The burnup calculation system consists of a general-purpose Monte Carlo code MCNP-4B, a point burnup code ORIGEN2 and some other postprocessing codes developed by us, which process the calculated results. In order to evaluate collapsed cross section for burnup as precise as possible, making of collapsed cross section process was postprocessed using not tally function but neutron track length data of the MCNP calculation directly. We are using a modified version of MCNP-4B so as to output all neutron track length data in the blanket region. JENDL-3.3 pointwise data and JENDL Activation Cross Section File 96 were used as base cross section libraries. From the calculation results for a reactor type employing a water cooled uranium cycle, fusion-fission hybrid reactor is surely feasible from the standpoint of neutronics design. Further calculation and analysis are planned to improve hybrid reactor performance of LLFP transmutation and so on.

1 Introduction

A fusion-fission (FF) hybrid reactor system is a concept of combining nuclear fuel with fusion reactor. Neutrons can be well multiplied by fission in the nuclear fuel loaded in the fusion reactor blanket even for a relatively lower plasma condition. Tritium is thus bred so as to attain its self-sufficiency. Then enough energy multiplication is expected and moreover it is possible to transmute or to burn the nuclear waste such as long-lived fission products (LLFP) and minor actinides (MA) by surplus high energy neutrons.

In our group, a calculation system for the analysis had been developed to investigate the performance of FF hybrid reactor that is feasible at present. Such feasible FF hybrid reactor is expected to have the following characteristics, i.e., low plasma condition, subcriticality and tritium self-sufficiency. Target parameters considered are thus tritium breeding ratio (TBR), k_{eff}, power density, energy multiplication factor in accordance with the changes of material composition due to burnup. Transmutation ability of long-lived fission products and minor actinides is also evaluated. The author's group has performed a lot of neutronics analysis with the calculation system for various types of FF hybrid reactors[1) . Through a long experience to use the system, it was found that the system used previously had a problem in the making process of collapsed cross sections for burnup[2) . The objectives of this work is to modify the burnup calculation system to make the cross section set for burnup very strictly with a special postprocess procedure. Also, as an example of the hybrid reactor analysis, the calculation results for a reactor type employing a water cooled uranium cycle is briefly described.

2 Calculation System

2.1 Development of Calculation System

Burnup calculations for FF hybrid reactor had been performed in the author's group so far with a calculation system combining general-purpose Monte Carlo code MCNP-4B[3] with point burnup code ORIGEN2[4] . However, it was found from the experiences previously that the calculation system had a problem on making one-group cross sections for burnup calculation. The one-group cross section set was made by the tally function of MCNP. Tally function is known to be a useful function for users to estimate neutron flux, reaction rate, and so on. However the number of tallies available for users is too small to compute collapsed cross sections for all the nuclei included. Thus the calculation system was planned to be modified to improve making procedure of the one-group cross sections for burnup.

There were three options to improve the calculation system as follows. (1)Postprocessing to make one-group cross section using the calculated neutron spectrum by the MCNP tally. This option was very easy to apply and faster computation can be expected. But it would spoil continuous treatment of neutron energy. (2)Modification of MCNP to make one-group cross section directly or to remove the limit of the number of tallies available. But drastic modification of MCNP is actually unreal, and increase of the number of tallies will expand the computation time extremely. (3)Extracting neutron transport data and postprocessing of it. This is the most strict procedure and was finally adapted in the present study. This option required a little modification of MCNP with a slight increase of the computation time. But it takes a longer time required to make collapsed cross sections in the postprocessing.

Figure 1 shows the current burnup calculation procedure. One burnup cycle consists of the following four steps: 1) Criticality calculation by KCODE of MCNP, 2) Neutron transport calculation by modified MCNP, 3) Making of ORIGEN library from neutron track length data and evaluated nuclear data libraries, and 4) Burnup calculation by ORIGEN for each blanket cells. The whole burnup calculation of the FF hybrid system is completed by repeating this burnup cycle.

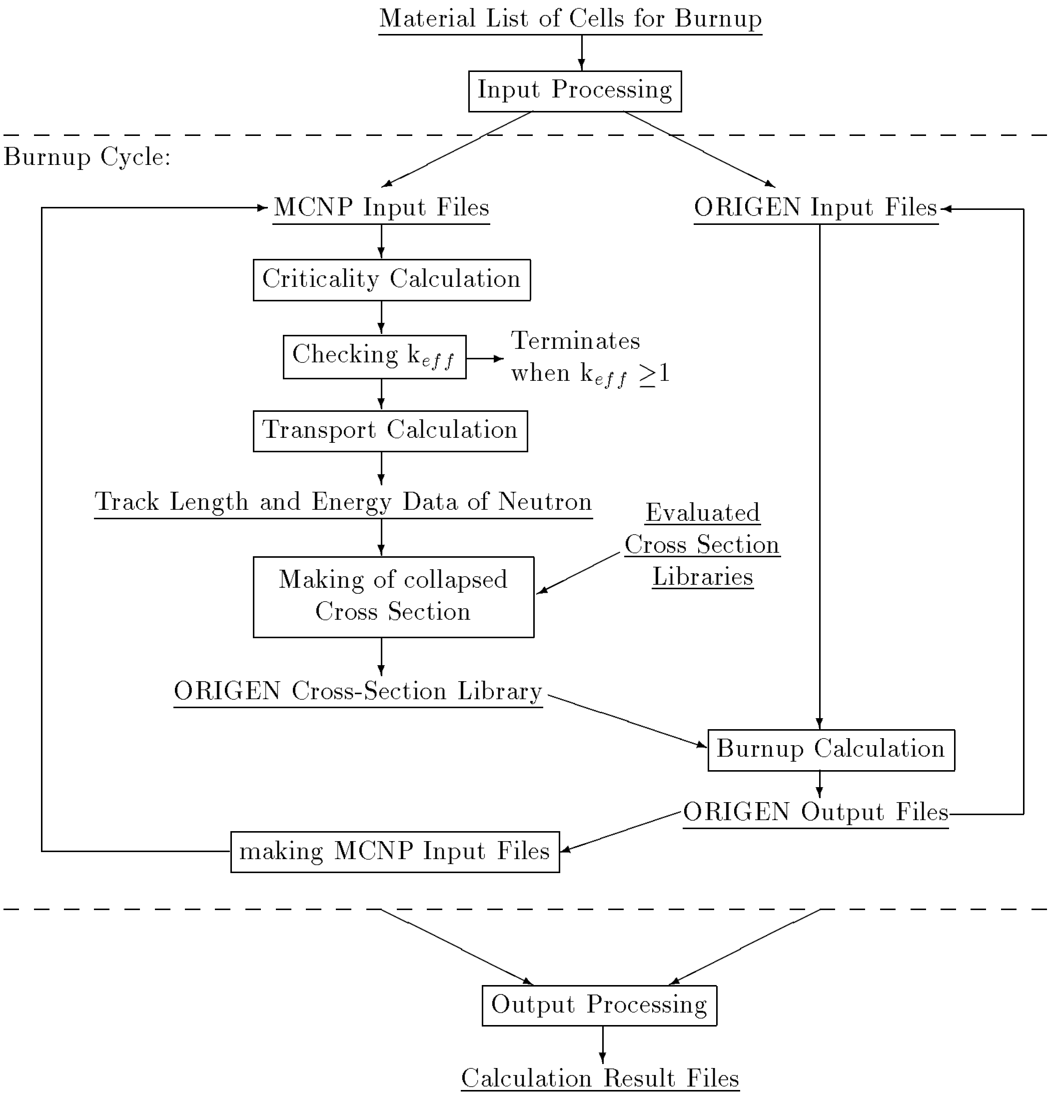

Figure 1: Flow Chart of Burnup Calculation

2.2 Making of Collapsed Cross Sections

In the ORIGEN code, cross section libraries for several types of nuclear reactor are prepared. However, those are not applicable to the present calculation, because FF hybrid reactor will be operated under subcritical condition and hence the energy spectrum varies depending on each configuration of nuclear fuel. It is thus needed to replace cross section values of the libraries attached in ORIGEN with the ones suitable for the present subcritical system. The ORIGEN code requires collapsed (one-group) cross-sections of (n,g), (n,2n), (n,α), (n,p), (n,gx) and (n,2nx) for activation products (including stable nuclei) and Fission products, and requires those of (n,g), (n,2n), (n,3n), (n,f), (n,gx) and (n,2nx) for actinides. (n,gx) and (n,2nx) mean cross sections of (n,g) and (n,2n) producing isomers.

As described above, in the present study, modified version of MCNP-4B was used to get all neutron transport data in the blanket cells. The data exported from a transport calculation and stored in a file are track-length (DLS), particle weight (WGT) and neutron energy (ERG). DLS, WGT and ERG are variable names of MCNP. Track-length is neutron flying distance between events, that is, reaching cell boundary and occurring of collision, as shown in Fig. 2.

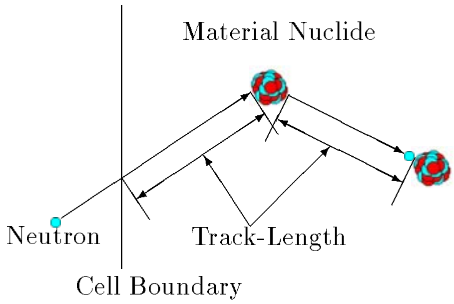

Figure 2: Neutron Track Length

The collapsed cross section is evaluated by the following equations.

$$\int \phi(E)\mathrm{d}E = \frac{\sum_i DLS_i \cdot WGT_i}{NPS \cdot VOL}$$

$$\sigma^{coll} = \frac{\int \sigma(E)\cdot\phi(E)\mathrm{d}E}{\int \phi(E)\mathrm{d}E} = \frac{\sum_i DLS_i \cdot WGT_i \cdot \sigma(ERG_i)}{\sum_i DLS_i \cdot WGT_i}$$

where:

σ^{coll}	Collapsed Cross-Section
E	Neutron Energy
$\sigma(E)$	Energy Dependent Cross-Section
$\phi(E)$	Energy Dependent Neutron Flux
DLS_i	Track-Length of Neutron Track i
WGT_i	Weight of Neutron Track i
ERG_i	Energy of Neutron Track i
NPS	History Number
VOL	Cell Volume

The cross sections of ORIGEN library are replaced by σ^{coll}. Point-wise data of $\sigma(E)$ is directly used from evaluated nuclear data libraries of JENDL-3.3 and JENDL Activation Cross Section File 96[5)]. The library for LMFBR contained in ORIGEN was also referred for minor nuclei as not contained in both of JENDL-3.3 and JENDL Activation Cross Section File 96.

3 Calculation Results

With the developed calculation system, various calculations have been carried out so far[6, 7]. In the present paper, as a typical example, brief results of a water cooled uranium cycle are described.

3.1 Calculation Model and Condition

Figure 3 shows cross sections of calculation geometry and tables 1 and 2 show calculation conditions. The 3-dimensional ITER[8] model (18 degree sector model) and physics parameters derived from the plasma conditions already achieved at JT-60 of JAEA[9] were employed.

The blanket was divided into 5 sections radially and each section was divided into 6 layers (30 cells total). The material composition of fuel cell and breeder cell are shown in table 3. Li_2ZrO_3 was selected as breeding material, and loaded into breeder cell with beryllium. ^{6}Li enrichment was 30 %. Fuel cell consists of structural material SS316, natural UO_2 and reprocessed PuO_2 as nuclear fuel and water as coolant.

There were two transmutation cells in second layer. 5 long-lived fission products (LLFP) of ^{93}Zr, ^{99}Tc, ^{107}Pd, ^{129}I and ^{135}Cs were loaded into transmutation cells with moderator.

In the present calculation, the period of burnup is 5 years and the plant factor is 70 %. Each year includes five burnup cycles and a cooling cycle of 100 days. Each burnup cycle lasts 53 days.

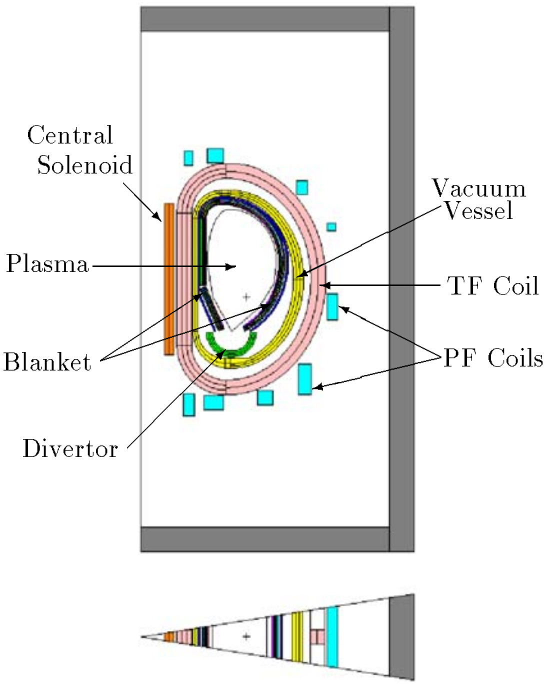

Figure 3: Cross Section of the Calculation Model

Table 1: Main parameters for calculations based on JT-60 and ITER

Plasma parameters	
Major radius (m)	6.2
Minor radius (m)	2.1
Plasma volume (m^3)	884
Plasma temperature (keV)	19
Confinement time (s)	1.1
Electron density (m^{-3})	4.8×10^{19}
Fusion power (MW)	646
neutron yield (n/s)	2.2×10^{20}
Neutron wall load (MW/m^2)	0.40 (average)
Blanket parameters	
Total thickness (m)	0.38
Total volume (m^3)	265

Table 2: Material loading arrangement

First Wall side	
1st layer	Breeder
2nd layer	Breeder*
3rd layer	Fuel
4th layer	Fuel
5th layer	Breeder
6th layer	Breeder
Vacuum Vessel side	

*two of five regions are replaced to FP transmutation cell

Table 3: Material composition

Composition of fuel (%)	
Reprocessed Pu	6.2
Natural UO_2	40.6
Water	36.7
SS316	16.5

Composition of breeder (%)	
Li_2ZrO_3	10
Be	90
^{6}Li enrichment (%)	30

3.2 Validation of Postprocessing

Table 4 shows validity confirmation result of the present making procedure of collapsed cross section. These 10 nuclei are especially significant when estimating the performance of FF hybrid reactor. The value in the table is the ratio of one-group cross section evaluated by the present procedure to FM option in MCNP. As shown in the table, present making procedure can be sufficiently trusted.

Table 4: Validity Confirmation Result of the Present Collapsed Cross-Section Making Procedure

Ratio of Calculated Cross-Section (this method* / MCNP Tally**)

	(n,g)	(n,2n)	(n,p)	(n,α)
^{93}Zr	9.998E−01	1.000E+00	9.998E−01	1.000E+00
^{99}Tc	1.000E+00	1.000E+00	1.000E+00	1.000E+00
^{107}Pd	9.999E−01	9.997E−01	1.000E+00	1.000E+00
^{129}I	9.998E−01	1.000E+00	1.000E+00	1.000E+00
^{135}Cs	9.998E−01	1.000E+00	1.000E+00	1.000E+00
	(n,g)	(n,2n)	(n,3n)	(n,f)
^{232}Th	9.999E−01	9.998E−01	9.999E−01	1.000E+00
^{233}U	9.977E−01	1.000E+00	1.000E+00	9.993E−01
^{235}U	1.000E+00	9.999E−01	9.998E−01	9.997E−01
^{238}U	1.000E+00	1.000E+00	1.000E+00	9.996E−01
^{239}Pu	9.998E−01	1.000E+00	9.999E−01	1.000E+00

*One-group cross-section estimated by the present procedure
*One-group cross-section estimated by dividing the calculated reaction rate by the calculated total flux

3.3 Burnup Calculation Results

The present burnup calculation system c an estimate parameters such as effective multiplication factor (k_{eff}), power density in fuel cell, neutron flux and changes of nuclide number density. In the present paper, three most significant parameters on discussing feasibility of FF hybrid system are selected.

Figure 4 shows tritium breeding ratio (TBR) and energy multiplication factor and table 5 indicates transmutation performance of 5 LLFPs over the whole burnup period. The most important restriction is to ensure the TBR to be over 1.05 during the operation of fusion reactor. Figure 4 shows that the TBR is always over 1.2 throughout the burnup period of 5 years. The energy multiplication factor is about 10, however decreasing monotonously to around 8. These results indicates that this type of FF hybrid system is feasible. But the results for transmutation performance is not so excellent. Some of them show even a negative result, meaning increase of LLFP. Practically, transmutation ratios of ^{99}Tc, ^{107}Pd and ^{129}I are acceptable for performance in the transmutation cell and in the whole blanket as well. However, the amount of ^{93}Zr and ^{135}Cs are inversely increased compared to their initial inventory. This result suggests further investigation especially for LLFP transmutation performance.

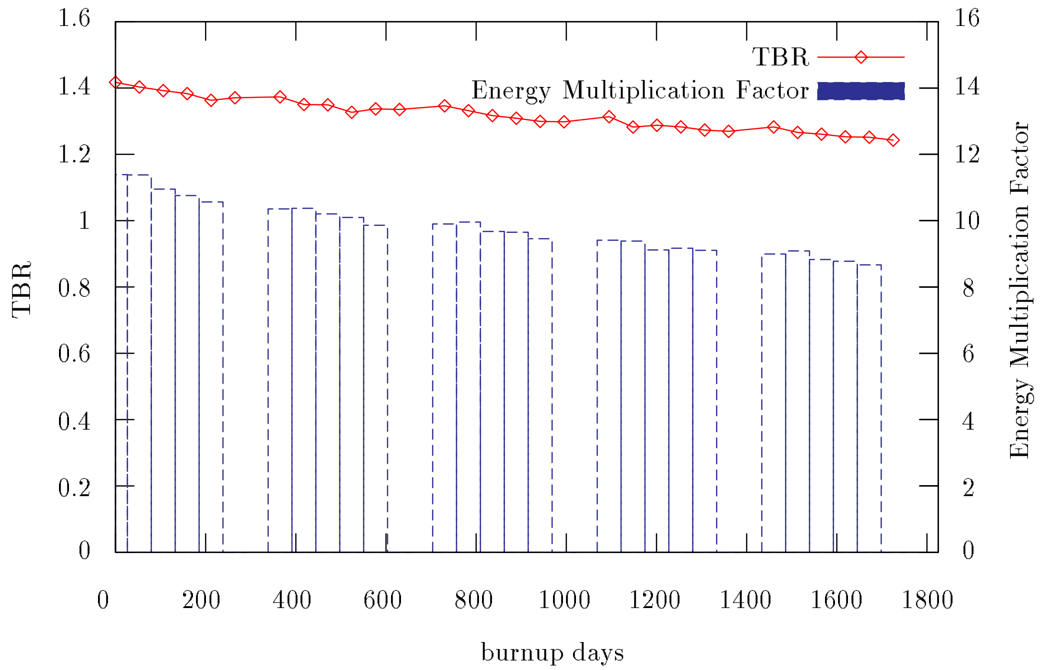

Figure 4: Burnup calculation results for TBR and Energy generation

Table 5: Transmutation Performance for LLFP (%)

(decreased percentage per initial load)

	^{93}Zr	^{99}Tc	^{107}Pd	^{129}I	^{135}Cs
transmutation cell	0.9	20	15	8	6
blanket total	-2	11	7	2	-19

4 Summary

A general-purpose transport and burnup code system for precise analysis of the fusion fission hybrid reactor has been developed. One group cross sections for burnup are updated in each burnup cycle, which is the most strict method to collapse cross sections. The present calculation system enables to perform precise neutronics analysis of subcritical systems such as a fusion fission hybrid reactor. However, at the moment, one difficult problem is still left, i.e., it takes a lot of computation time to calculate one-group cross section set.

From the present calculation results carried out with the newly developed calculation system, it is suggested that FF hybrid reactor is surely feasible from the standpoint of neutronics design. Further calculation and analysis are nevertheless needed to improve hybrid reactor performance including nuclear waste transmutation performance.

References

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