TECHNICAL REPORT

Monte Carlo Calculation of First Wall Neutron Flux in Tokamak Fusion Reactor

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The poloidal distribution of the first wall 14 MeV neutron flux and the tritium breeding ratio in a Tokamak fusion reactor were calculated using Monte Carlo method. The poloidal distribution of the 14 MeV neutron flux in the first wall was found to be quite different from that of the primary incident flux. The tritium breeding ratio calculated by the Monte Carlo method became about 5% larger than the value obtained from $S_N$ transport calculations.

**KEYWORD:** Monte Carlo method, neutron flux, first wall, tritium breeding ratio, Tokamak fusion reactors, track length estimator

I. INTRODUCTION

The poloidal peaking of the first wall 14 MeV neutron flux and the tritium breeding ratio in a Tokamak fusion reactor(1) are calculated using a newly modified Monte Carlo code(2).

The poloidal peaking of 14 MeV neutron flux in the first wall of a toroidal fusion reactor will be one of the factors limiting the lifetime of the first wall. The primary 14 MeV neutron flux incident on the first wall from a toroidal plasma has been calculated by several authors(3)~(7), using the ray-tracing method which derives the primary flux of a point on the first wall by numerically integrating the source density distribution observable from that point. With this method, only the primary 14 MeV neutron flux can be obtained but not the actual** 14 MeV neutron flux. The neutron attenuation by the coating material on the first wall and by the first wall itself and the neutron reflection by the blanket material behind the first wall must be taken into account for the calculation of the actual flux. The actual flux is not always proportional to the primary flux. In order to obtain an accurate distribution of the actual flux, either a discrete ordinates radiation transport code which can treat toroidal geometry(8) or a Monte Carlo code with the same capability(2)(9) is required. A conventional two-dimensional discrete ordinates code may also be employed(2)(10) but the mitigation of the ray effect and the large number of required spatial mesh points to model a torus make the employment difficult.

Two Monte Carlo radiation transport codes with the capability of treating torus geometry have been recently developed and applied to solve neutronics problem of toroidal fusion reactors(2)(9). The Monte Carlo code MORSE-GG(11) has been modified in Japan Atomic Energy Research Institute (JAERI) to treat torus geometry and to use

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** Here and throughout this report the adjective “actual” is used in the meaning of “accurately simulated”. 
several kinds of scoring estimators\(^{(6)}\). The newly evolved code is called MORSE-I. In this report the code is applied to obtain the first wall neutron flux and tritium breeding in the JAERI Experimental Fusion Reactor (JXFR)\(^{(1)}\).

II. CALCULATIONAL MODEL AND METHOD

The model of the plasma and blankets of JXFR used in the present calculation is shown schematically in Fig. 1. The inboard and outboard blankets consist of different materials as described in the attached table in Fig. 1. The thin black region with the media number 4 is the first wall made of stainless steel. The application of Monte Carlo method is especially suited to such a complex model. The 42-energy group macroscopic neutron cross sections for each medium in Fig. 1 were compiled from the 42-group neutron cross section library GICX40\(^{(12)}\). Track length estimator was used for the calculations of both 14 MeV neutron flux and the tritium breeding ratio.

In order to obtain the poloidal distribution of neutron flux, the first wall was divided into 19 regions in the poloidal direction.

In order to test the code MORSE-I\(^{(2)}\), the primary 14 MeV neutron flux was calculated for a tokamak reactor with a circular cross section plasma filled with a uniform isotropic neutron source. The result of MORSE-I was compared to a previously studied case by Dänner\(^{(4)}\). A configuration with the aspect ratio of the plasma \(A=4\) and the ratio of the plasma to wall radius \(r_p/r_w=0.5\), was chosen. The primary 14 MeV neutron flux was calculated by making the first wall thickness infinitesimal (0.1 mm), and by terminating each history once a neutron crossed the first wall.

As shown in Fig. 2, the result of MORSE-I agreed with that by the ray-tracing method within the statistical errors for most of the angles. The error bars in the MORSE-I result show the extent of one fractional standard deviation (FSD). Note the FSD’s are larger at the inside...
surface of the torus ($\beta<0^\circ$) than at the outside. This is partly due to the more frequent incidence of neutrons grazing the wall surface on the inside which results in extremely long track length among the shorter majority. It is also caused by the relative smallness of the detector region volume on the inside resulting in less samples. Tritium breeding ratio was calculated by using much larger detector regions consisting of all the regions, containing media 5 and 6 (see Fig. 1).

### III. RESULT AND DISCUSSIONS

Figure 3 shows the poloidal distribution of the primary 14 MeV neutron flux and the actual 14 MeV neutron flux in the first wall of JXFR. The actual flux is smaller than the primary flux because the attenuation of the former by the collisions in the 5 mm thick carbon coating and the first wall itself (1 cm thick) is greater than the increase of the reflected neutrons by the blanket. The reflected 14 MeV neutron flux was calculated to be only 1.4% of the primary flux. Figure 3 gives an example where the poloidal distribution of the actual flux is quite different from that of the primary flux. The actual flux is higher on the outside of the torus ($\beta>0^\circ$) while the primary flux has a single peak at $\beta\sim-20^\circ$. Therefore the extrapolation of the peaking position of the actual flux from that of the primary flux is not valid in this particular case.

One explanation for the relatively low value of the actual flux on the inside of the torus ($\beta<0^\circ$) is due to the larger reduction of the grazing components due to the attenuation of the flux by the neutron collisions in the carbon coating and the first wall. Since the reflected 14 MeV neutron flux is only 1.4% of the primary flux, the actual 14 MeV neutron flux in the first wall is relatively independent of the blanket composition.

In order to save computation (CPU) time, energy cut-off was set at 14 MeV to follow only the particle histories of 14 MeV neutrons. Nevertheless the computation time required to obtain a poloidal distribution of a primary flux or an actual 14 MeV flux was about 6 h by FACOM-230/75 system. Even with such a great amount of computation time, there still remain some inconvenient statistical errors. If only the calculation of the primary neutron flux is required, the ray-tracing method is far superior than the Monte Carlo method. For a circular cross section plasma and first wall, 2D $S_N$ method with toroidal geometry handling capability has been demonstrated to be effective for the calculation of the actual flux$^{(13)}$. However, for non-circular cross section plasma and first wall$^{(9),(14)}$, the Monte Carlo method seems to be most effective.

Tritium breeding ratio was calculated to be $0.95\pm0.02$ with 0.02 being the statistical error. This value is a little larger than the value 0.90 estimated from a series of 2D $S_N$
transport calculations employing $R-Z$ and $R-\theta$ geometry\(^{(1)}\). This large difference may be attributed to various differences between the two calculational methods, such as those in source and geometry modeling, ray effect, employed multigroup structure etc. The largest source of error in the $S_N$ calculations is estimated to be in the approximation in the modeling of the source and geometry. This Monte Carlo calculation for tritium breeding without any neutron energy cut-off took only 30 min or about 1/5 of the total for $S_N$ calculations. Monte Carlo method is effective in calculating a nuclear characteristic value integrated over a large volume but not so for obtaining a quantity integrated over a small detector region such as poloidal distribution of neutron flux in the first wall.

IV. CONCLUSIONS

The poloidal distribution of the first wall 14 MeV neutron flux and the tritium breeding ratio in JXFR were calculated using Monte Carlo method. As a result the following conclusions have been obtained:

1. The poloidal distribution of the actual 14 MeV neutron flux in the first wall was found to be quite different from that of the primary flux. This is mainly due to the consideration of the attenuation of the grazing components in the carbon coating and the first wall.

2. The tritium breeding ratio was calculated to be about 5% larger than the previously obtained value by $S_N$ calculations. Major cause for the 5% increase is estimated to be in the improvement in the modeling of the real geometry.

3. The Monte Carlo method is not recommendable for the calculation of flux distributions in a relatively simple geometry where other methods may be applicable. However it seems to be the best choice in calculating nuclear characteristics of a toroidal fusion reactor with noncircular cross section plasma and first wall.

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