Fusion Engineering and Design 137 (2018) 283–288

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Deterministic method for calculating fusion neutron emission probability distribution in tokamak through mesh mapping

Haibing Guo[a,*], Hongli Chen[b], Hongwen Huang[a], Shaohua Wang[a]

[a] Institute of Nuclear Physics and Chemistry, China Academy of Engineering Physics, China
[b] School of Nuclear Science and Technology, University of Science and Technology of China, China

ARTICLE INFO

Keywords:
Tokamak
Fusion neutron source
Deterministic
Mesh mapping
Finite element

ABSTRACT

The high-fidelity 3D neutronics analysis of fusion systems has been enabled by advanced computer-aided design (CAD)-based simulation tools, and is being driven by an increasing desire to couple the analysis results with other physical field analyses. With increase in the level of output detail, the effects of the fusion neutron source spatial distribution have become increasingly important. The current fine-resolution neutron source model was established based on the Monte Carlo (MC) method, which is a time-consuming approach to reduce the deviations at low-intensity positions. This work presents a deterministic method to calculate the fusion neutron emission probability (FNEP) distribution in a tokamak machine with high resolution. In this deterministic method, the mesh-mapping technique is utilized to transfer neutron intensity data from an irregular (a, φ) mesh to a structured or unstructured (R, Z) mesh. The intensity data on (R, Z) mesh can be used directly by a deterministic neutronics code, or further be integrated and normalized in every grid to obtain the FNEP for an MC neutronics code. This deterministic scheme was implemented in the code EDITON, and the neutron source map can be generated subsequently by this code. EDITON exhibited high efficiency, and its accuracy was verified by comparing the obtained FNEP distribution and neutron wall loading (NWL) with the corresponding data from the TRANSGEN code, which is a tokamak fusion neutron spatial distribution calculation code based on the MC method.

1. Introduction

The high-fidelity three-dimensional (3D) neutronics analysis of fusion systems has been enabled by the development of advanced computer-aided design (CAD)-based simulation tools such as SuperMC [1] and Attila [2], and some other codes such as MCNPX [3] and TRIPOLI-4 [4] also have this capability by coupling with CAD pre-processing engines CGM and MCAM, respectively. These new methodologies can facilitate a more detailed understanding of the behavior of neutrons (and the photons they generate) in fusion systems. High-fidelity 3D neutronics analysis is also being driven by an increasing desire to couple the results of such studies with other physical field analyses. For example, fine-resolution nuclear heating is required in thermal-hydraulic (T−H) analysis with computational fluid dynamics (CFD) methods, and the time-dependent neutron spectrum and flux are required in material irradiation performance analysis. Furthermore, the mechanical analysis depends on the temperature distribution calculated via T−H analysis.

However, as the level of output detail increases, the effects of the fusion neutron source spatial distribution become more important. For example, the neutron wall loading (NWL) is more strongly affected by the source distribution than other design parameters [5]. The NWL is an important measure that reflects the effect of the source distribution. Therefore, high-fidelity analysis requires accurate fine-resolution descriptions of the neutron source spatial distribution.

For toroidally symmetric machines, e.g., a tokamak, the (R, θ, Z) coordinate system is useful for neutronics analysis. In general, the physical field in the θ direction is uniform. Thus, the neutron source description can be reduced to the (R, Z) plane. Further, the 3D effect can be reflected by establishing a rotational symmetric geometry model to take into account the length difference along the θ direction at different radial locations.

The first high-fidelity plasma source model was the multi-tori-layer model [6]. It described a series of elliptical toroidal surfaces, and assumed that the source neutrons were uniformly distributed in each inter-surface layer. However, the elliptical toroidal surfaces did not fit the actual equi-emission surfaces well, and the resulting accuracy was low. A direct Monte Carlo (MC) sampling subroutine based on

* Corresponding author.
E-mail address: guohaib@caep.cn (H. Guo).

https://doi.org/10.1016/j.fusengdes.2018.09.017
Received 7 August 2017; Received in revised form 30 June 2018; Accepted 30 September 2018

coordinate transformation was previously implemented in the MCNP code, but the efficiency was limited when it was applied to large-scale problems. A deterministic method has been proposed by Dr. R. N. Slaybaugh based on spline interpolation and previously obtained poloidal magnetic flux data which were defined on an (*R*, *Z*) mesh and expressed in the community-standard “geqdsk” format [7]. Dr. R. N. Slaybaugh also calculated the 3D neutron source density in the stellarator ARIES-CS by finite element integration method in hexahedral meshes, but it must be noticed that the hexahedral meshes were constructed according to the magnetic flux surfaces which were defined in an idealized toroidal coordinate system. These hexahedral meshes can hardly be described directly by SDEF card of MCNP; actually the CGM tool was used [3]. It is an efficient strategy for a stellarator machine to define every source grid to be a geometry cell, and assign the neutron source densities to the corresponding cells.

In the previously released ITER A-lite and B-lite MCNP5 models, the vertical cross-section of the neutron source zone was partitioned into 40 × 40 identical quadrilateral (*R*, *Z*) meshes and the fusion neutron emission probability (FNEP) in each grid was provided [8–10]. This method sufficiently improved the accuracy and efficiency in source sampling. The FNEP values in the ITER A-lite and B-lite models were calculated by means of a separate MC subroutine; it was time-consuming to reduce the deviations at low-intensity positions although the variance reduction technique was utilized.

Against these backdrops, in this work, we develop a deterministic method to calculate the FNEP distribution in a tokamak with high resolution. The mesh-mapping technique is utilized to transfer fusion neutron intensity (FNI) data from an irregular (a, φ) mesh to the destination structured or unstructured (*R*, *Z*) mesh. The intensity data on the (*R*, *Z*) mesh can be used directly by a deterministic neutronics code, or further be integrated and normalized in every grid to obtain the FNEP for an MC neutronics code. This scheme was implemented in the code EDITON (DeterminIstic TOkamak Neutron source map generator, the letter ‘E’ is the first letter of the Chinese name for Institute of Nuclear Physics and Chemistry), and the neutron source map can be generated subsequently by this code. EDITON exhibited high efficiency, and its accuracy was verified by comparing the obtained FNEP distribution and NWL results with the corresponding data from TRANSGEN code [11], which is a fusion neutron spatial distribution calculation code based on the MC method.

2. Theoretical distribution of fusion neutron source

For a realistic tokamak neutron source model, the spatial distribution of the neutrons can be characterized by a series of D-shaped equi-emissivity surfaces [5,11], which are defined as

$$R = R_0 + a\cos(\varphi + \varepsilon\sin\varphi) + \delta(1 - a^2/A^2),$$
$$Z = ka\sin\varphi + \sigma. \tag{1}$$

Here, R and Z represent the horizontal and vertical positions in cylindrical coordinates, respectively; R_0 and A the major and minor radii of the last closed magnetic surface (LCMS), respectively; a the reduced plasma horizontal minor radius; φ the poloidal angle; ε the triangularity; δ and σ the maximum plasma radial and vertical shifts, respectively; and k the plasma elongation.

Parameters R_0, A, ε, δ, σ, and k were determined for a specific tokamak machine, and their typical values for European HCLL-DEMO-2007 (EURO-DEMO) [12] and the Chinese Fusion Engineering Test Reactor (CFETR) [13] are listed in Table 1.

Via discretizing the reduced minor radius a ($0 \le a \le A$) and the poloidal angle φ ($0 \le \varphi \le 2\pi$), the spatial boundaries of a series of equi-emissivity surfaces can be shaped. Meanwhile, the spatial distribution of the FNI can be derived from the plasma physics formula according to the confinement mode. In the low-confinement mode (*L*-mode), the FNI formula is expressed as

Table 1
Plasma parameters of EURO-DEMO and CFETR.

Parameters	EURO-DEMO	CFETR
Major radius R_0 / m	7.5	5.7
Minor radius A / m	2.5	1.6
Plasma elongation k	2.1	1.8—2.0
Plasma triangularity ε	0.7	0.4—0.8
Maximum radial shift / m	0.0	0.20
Maximum vertical shift / m	0.0	0.44

$$S(a) = S_0\left[1 - \left(\frac{a}{A}\right)^2\right]^P, \quad P = 2\alpha_n + \gamma\alpha_T, \tag{2}$$

where S_0 indicates the FNI at the central magnetic surface ($a = 0$), i.e., at ($R_0 + \delta$, σ); α_n and α_T denote parameters from plasma physics calculations related to the ion density and ion temperature, respectively; γ denotes a parameter used to simplify the Maxwellian reactivity expression, being equal to 2.0 and 3.5 for average ion temperatures of 15 keV and 5 keV, respectively, for deuterium-tritium (D–T) reactions; and P denotes the power peak factor, which is user-calculated. For EURO-DEMO and CFETR, P is 1.3 and 1.4, respectively [12,13].

Upon calculating the value and curvature of $S(a)/S_0$ for variable a, we found that the $S(a)/S_0$ curve is curly in a narrow range near the LCMS, although the value is rather small. The case for the EURO-DEMO is shown in Fig. 1, from which we note that the curvature is rather large and increases abruptly near the LCMS. Therefore, in a numerical discretization process, a high-order shape function or very fine mesh are needed near the LCMS to accurately fit the original value. Otherwise, on a coarse mesh with linear shape function, the integration will be overestimated.

For a given point (*R*, *Z*) in cylindrical coordinates, the corresponding (a, φ) value cannot be determined analytically or by any straightforward numerical method from Eq. (1). A nonlinear iterative numerical process is needed to solve the Eq. (1), but this algorithm may be tedious and time-consuming when applying it to all the nodes on the (*R*, *Z*) mesh. As a result, the corresponding FNI value for point (*R*, *Z*) cannot be calculated analytically or by any straightforward numerical method. In this work, we calculated the FNI values on the (a, φ) mesh first according to Eqs. (1) and (2), and then transfer the FNI data to the destination (*R*, *Z*) mesh map mathematically through the mesh-mapping technique.

3. Mesh mapping and data transfer scheme

We have to first establish the (a, φ) mesh in order to calculate the FNI and to apply the mesh-mapping scheme. The main concern is the

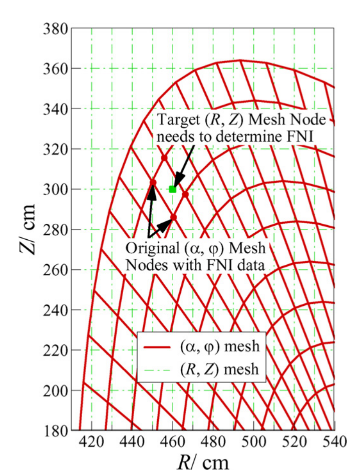

Fig. 1. Variation in relative intensity and its curvature versus reduced minor radius.