Neutronics Investigation of Advanced Self-Cooled Liquid Blanket Systems in Helical Reactor

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Abstract. Neutronics performances of advanced self-cooled liquid blanket systems have been investigated in design activity of the helical-type reactor FFHR2. In the present study, a new three-dimensional (3-D) neutronics calculation system has been developed for the helical-type reactor to enhance quick feedback between neutronics evaluation and design modification. Using this new calculation system, advanced Flibe-cooled and Li-cooled liquid blanket systems proposed for FFHR2 have been evaluated to make clear design issues to enhance neutronics performance. Based on calculated results, modification of the blanket dimensions and configuration have been attempted to achieve the adequate tritium breeding ability and neutron shielding performance in the helical reactor. The total tritium breeding ratios (TBRs) obtained after modifying the blanket dimensions indicated that all the advanced blanket systems proposed for FFHR2 would achieve adequate tritium self-sufficiency by dimension adjustment and optimization of structures in the breeder layers. Issues in neutron shielding performance have been investigated quantitatively using 3-D geometry of the helical blanket system, support structures, poroidal coils etc. Shielding performance of the helical coils against direct neutrons from core plasma would achieve design target by further optimization of shielding materials. However, suppression of the neutron streaming and reflection through the divertor pumping areas in the original design is important issue to protect the poroidal coils and helical coils, respectively. Investigation of the neutron wall loading indicated that the peaking factor of the neutron wall load distribution would be moderated by the toroidal and helical effect of the plasma distribution in the helical reactor.

1. Introduction

The collaborative design activity of the helical-type power reactor FFHR2 (Force Free Helical Reactor) has been conducted by NIFS (National Institute for Fusion Science) and universities [1]. In the activity, the neutronics performance has been studied as one of important issues for deciding the reactor parameters and improving the performance by design modification. Four types of advanced self-cooled liquid blanket systems with Flibe or Li coolant have been studied for FFHR2 to achieve the high-performance power generation. The fundamental neutronics potentials for the proposed blanket concepts have been studied by neutron transport calculations with 1-D [1, 2] or simple torus geometry [3,4]. After optimization of material compositions and layer thickness to achieve the compatibility between tritium self-sufficiency and radiation shielding, neutronics performance of the blanket systems has been investigated in the geometry of FFHR2. For the progress in the reactor design, evaluation of neutronics performance with 3-D geometry data has been required to simulate the neutron transport in complicated helical blanket components. The effort to import 3-D geometry data from CAD system into neutronics calculations has been conducted by each neutronics group in Japan, US, EU and China for high accuracy evaluation [5-8]. However, the approach with CAD data has been considered unsuitable for the conceptual design activity of FFHR2, since it would require time for preparation, correction and modification of data. In the present study, development of new 3-D neutron transport calculation system has been started for further understanding of neutronics characteristics in helical reactors. The system is focusing on quick generation of 3-D geometry data of helical structures and feedback to the design modification. Distribution of the fundamental neutronics parameters such as tritium production, neutron wall load, neutron flux etc. has been evaluated for the FFHR2 design. The neutroincs characteristics and issues in the helical reactor have been investigated and

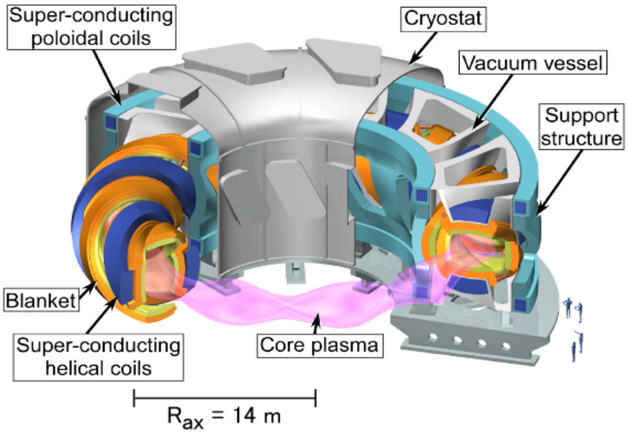

FIG. 1. Helical-type power reactor FFHR2m1.

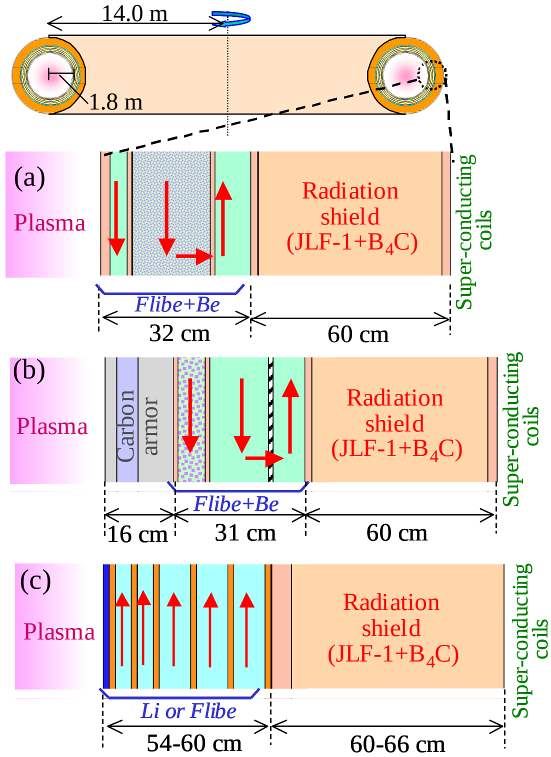

FIG. 2. Structures of advanced liquid blanket systems proposed for FFHR2m. (a)Flibe+Be/JLF-1 (b)Flibe cooled STB (Spectral-shifter and Tritium breeding Blanket) (c) Li/V-alloy, Flibe/V-alloy blanket systems

discussed.

2. Advanced Liquid-Cooled Blanket Systems for FFHR

Figure 1 shows the drawing of FFHR2m1, which is the present version of the FFHR2 design [9]. The major radius of the torus is 14.0 m and the plasma radius is 1.73 m. Structures of blanket systems proposed for the FFHR2 are shown in Fig.2. The design activity has originally been conducted with the self-cooled Flibe+Be/JLF-1 (Reduced Activation Ferritic/Martensitic Steel) blanket system (Fig.2 (a)) for the attractive merits on safety aspects, low MHD resistance etc. Recently, the Flibe cooled STB (Spectral-shifter and Tritium breeding Blanket) concept employing thick carbon armor (Fig.2 (b)) has been proposed for the FFHR2m1 design to avoid the critical neutron damage on the first wall [9,10]. In this concept, the replacement-free blanket would be achieved during the reactor lifetime of ~30 years by reducing the irradiation damage on the first wall of ferritic steel. In addition to the original Flibe cooled systems, Li/V-alloy (Vanadium alloy) and Flibe/V-alloy blanket concepts without solid Be multiplier (Fig.2 (c)) also have been studied as alternatives for high-temperature operation [3,4]. Neutronics properties of the candidate blanket systems have been investigated by using the Monte-Carlo neutron transport code MCNP-4C [11] and nuclear library JENDL-3.2 [12] for simple torus geometry as shown in Fig. 2. The blanket space of 1.2 m and the neutron wall load of 1.5 MW/m^2 (~5 x 10^{14} n/cm^2/s) are parameters related to the neutronics investigation. After optimization of material composition and thickness for the breeder and shielding layers, both requirements on the local tritium breeding ratio (local TBR) of 1.2-1.3 and the neutron shielding performance (<1.0x10^{10} n/cm^2/s for fast neutrons (>0.1 MeV) at superconducting magnet system) are satisfied in all of the blanket systems with combination of a breeder layer of 30-60 cm and a radiation shield of ~60 cm.

3. 3-D Neutronics Calculation System

Since the original FFHR2 design has been adapting the helical divertor system, the DT core plasma is covered with four separated blanket layers running helically in the toroidal direction as shown in Fig. 3 (a). The openings between the separated blanket layers are the divertor pumping areas. For neutronics evaluation in the helical reactor, neutron transport in the helical blanket components, streaming through the divertor pumping areas and reflection from the components such as support structures, vacuum vessels etc. are important factors to be

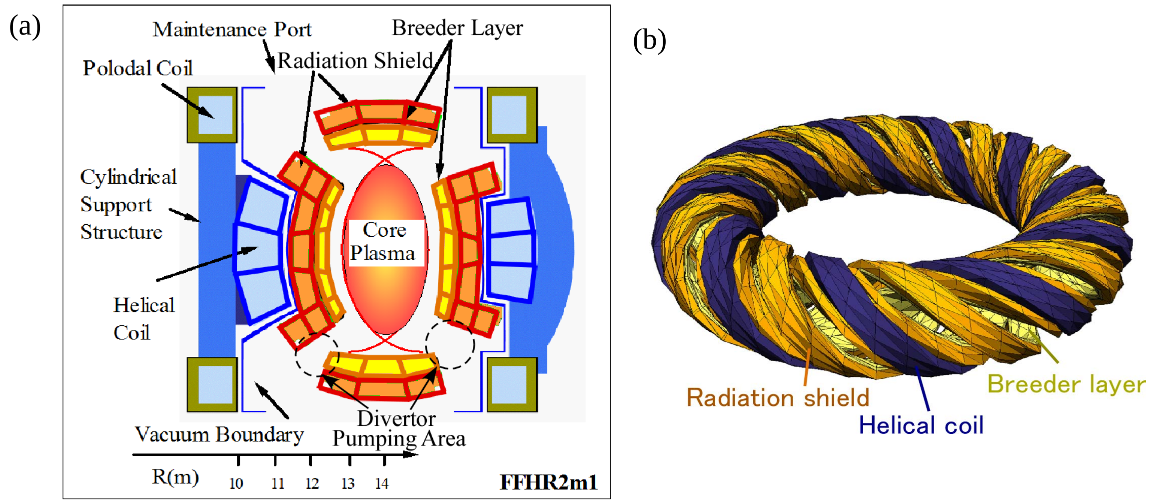

FIG. 3. Making of 3-D geometry data for neutronics calculation. (a) Division of cross-section of helical blanket components into quadrangular meshes. (b) Example of 3-D geometry of helical blanket system. Vertices are calculated according to the numerical equations defining all the helical structures of FFHR2.

simulated with 3-D geometry data. In the present study, the development of the 3-D neutronics calculation system has been started focusing on quick feedback between the neutronics evaluation and design modification. Input data for the system are prepared by dividing the cross-section of blanket components into quadrangular meshes on design drawing as shown in Fig. 3 (a). From the coordinates of each quadrangular mesh, a computer program written for the present system calculates the vertices of the 3-D helical structure according to the numerical equations defining all the helical structures of FFHR2 [13,14]. Figure 3 (b) shows an initial model of the geometry data for the original blanket configuration of FFHR2m1. The vertices are calculated at every 6 degrees along the troidal direction in the initial model. The geometry data are converted to an input file for the MCNP5 Monte-Carlo neutron transport code with the program. The full torus geometry data shown in Fig. 3 (b) are consisting of ~3,000 cells for the transport code. Calculated neutronics parameters are extracted from an output file of the MCNP code with a post-process program also developed for the present system and 3-D distribution of the results can be drawn with visualization software.

4. Neutronics Performance of Helical Blanket System and Discussions

4.1 Tritium Breeding Ability

Tritium breeding abilities in the Flibe+Be/JLF-1 and Li/V-alloy blanket systems have been evaluated using the 3-D calculation system [14]. Cross-section and compositions of the blanket systems are shown in Fig. 4 (a). The breeder layer was simplified to uniform mixture of a breeding material, neutron multiplier and structural material. From results of transport calculations using a simple torus geometry shown in Fig. 2, it was confirmed that this simplification gives lower TBR (Tritium Breeding Ratio) by ~4 % for the Flibe+Be/JLF-1 blanket and by ~0.3 % for the Li/V-alloy blanket compared with the geometry simulating the distribution of the materials in the breeder layer. Shielding layers of JLF-1 and B_4C were placed outside of the breeder layer. A uniform torus-shaped neutron source of 1.5 m in diameter was assumed in the investigation.

The total TBRs evaluated for the original FFHR2m1 geometry were 0.82 (Flibe+Be/JLF-1 blanket system) and 0.81 (Li/V-alloy blanket system), respectively. Comparison with the local TBRs calculated for the simple torus model indicated that the effective coverage of the original blanket configuration was 60-70 % due to the wide opening at the divertor pumping

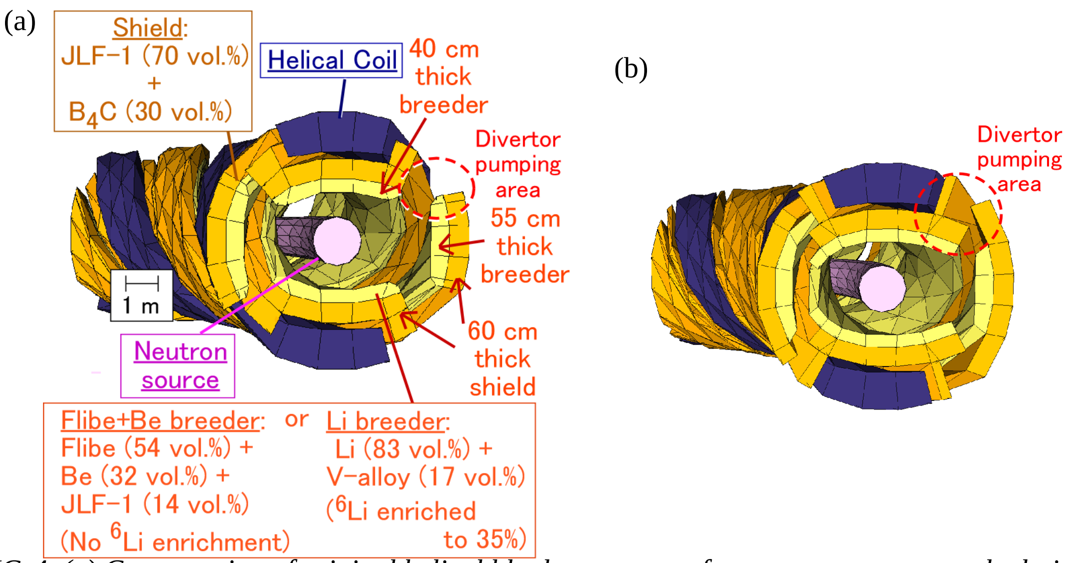

FIG. 4. (a) Cross-section of original helical blanket geometry for neutron transport calculation. (b) Cross-section of blanket geometry modified for improvement of tritium breeding.

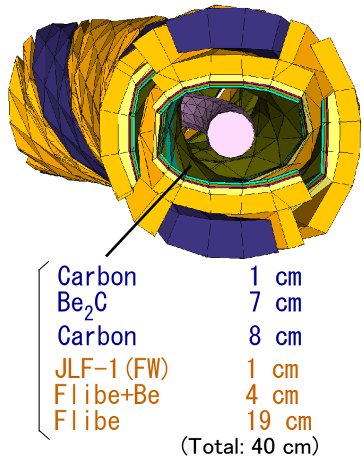

FIG. 5. Cross-section of 3-D geometry for STB concept.

areas. In order to enhance the tritium breeding ability and shielding efficiency, the shaping of blankets has been modified by expanding the edge of each blanket as shown in Fig.4 (b). For this modified blanket configuration, the effective coverage increased to ~80 % and the total TBRs in the Flibe+Be/JLF-1 and Li/V-alloy blanket systems were 1.08 and 0.98, respectively. The TBR of the Flibe/V-alloy blanket system, in which the volume ratios of Flibe and V-alloy are the same as the Li/V-alloy blanket, was 0.97 for the modified configuration. The three blanket systems would achieve the total TBR>1.0 by further design modification such as optimization of reflectors, dimension adjustment etc. Since the helical structures of FFHR2 is rotating with the period of 72^{o} along the troidal direction, it was confirmed in the TBR calculations that the number of cells can be reduced to ~600, i.e. 1/5, successfully by using the periodic boundary function of the MCNP code.

In contrast to the three blanket systems, simulation of multi-layer structure is necessary in the TBR evaluation for the STB concept (Fig. 5). Since the neutron energy from core plasma is dramatically attenuated with the thick carbon armor, the position and thickness of the carbon and Be_2C layers affect the TBR sensitively. By increasing the number of cells in the MCNP calculation to ~1,900 (72^{o} model), the TBR for the STB concept was obtained with the modified blanket configuration. In the previous optimization of the layer thickness and material compositions using the simple torus model, the thickness for JLF-1 first wall has been set to 1 cm. However, the total TBR calculated with the present 3-D geometry was 0.91 due to the drastic absorption of thermal neutrons by the JLF-1 first wall. The relation between the local TBR and the first wall thickness evaluated with the simple torus model [9] indicated that the value of the total TBR would be ~1.05 by reducing the first wall thickness to 5 mm in the design effort.

4.2 Shielding Performance

Issues of neutron shielding in helical reactors have been discussed in the previous neutronics studies for the Heliotron-H reactor design [15,16]. Importance of shielding for neutrons flowing in the tangential direction of the torus has been pointed out by neutron transport calculations with a coarse 3-D geometry due to lower computer ability at that time. In the present investigation for the FFHR2m1 design, neutron streaming through the openings of the divertor pumping areas is the main feature to be simulated with the 3-D neutronics calculation system. The target of the shielding performance in the FFHR2 design is to suppress the fast neutron flux (>0.1 MeV) lower than 1.0 x 10^{10} n/cm^2/s at the superconducting magnet system to avoid the critical damage during 30 years operation [17].

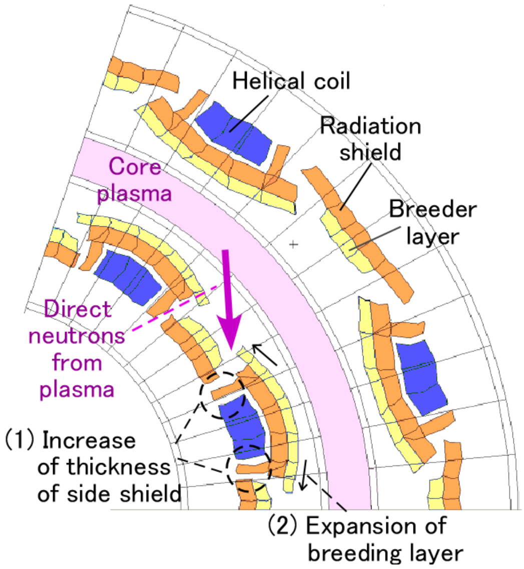

FIG. 6. Horizontal cross-section of blanket system and modification of shielding performance against direct neutrons from core plasma.

Shielding ability of the helical blanket system against direct neutrons from the core plasma is evaluated with the geometry shown in Fig. 4(b) and the Flibe+Be/JLF-1 blanket system. The intensity of the neutron source was fixed to 6.6 x 10^{20} n/s for the 360^o full torus geometry, which gives the neutron wall load of 1.5 MW/m^2 in the simple torus geometry shown in Fig. 2. The distribution of fast neutron flux on the helical coil surface was investigated by 3-D visualized drawing of the calculated results. The results indicated the inhomogeneous distribution of fast neutron flux on the surface. The maximum values was ~9 x 10^{11} n/cm^2/s at the inner side of the torus. From the horizontal cross-section of the blanket system shown in Fig. 6, it has been understood that the direct neutrons from the core plasma are hitting the thin side shield of ~30 cm in thickness for the helical coils especially at the inner side of the torus. By increasing the thickness of the side shield to ~50 cm, the maximum fast neutron flux on the helical coil has been reduced to ~9 x 10^{10} n/cm^2/s. Further reinforcement of the shielding performance has been attempted by expanding the breeder layers to shield the side surfaces from the direct neutrons. By the expansion, the maximum fast neutron flux could be suppressed to ~3 x 10^{10} n/cm^2/s. The previous neutronics investigations using the simple torus geometry indicated that the fast neutron flux would be ~1/3 by increasing the ratio of B_4C in the shield layer [4] and the shielding performance against direct neutrons from the core plasma could achieve the design target.

As shown in Fig. 7(a), non-helical structures of the poroidal coils, supporting structures, vacuum vessel, beljar and floor panel were added around the helical blanket components for evaluation of fast neutron flux on the poroidal coils and effect of neutron reflection. An example of 3-D distribution of fast neutron flux is shown in Fig. 7(b). In the figure, results on the upper poroidal coils, one of the helical coils and the support structures are extracted and drawn. The gradation of the color is by smoothing function of the visualization software. The distribution indicated that the fast neutron flux was extremely high on the outer side support structure due to neutron streaming through the divertor pumping areas. The maximum fast neutron flux was 1.7 x 10^{11} n/cm^2/s on the poroidal coils and exceeded the design target also due to the neutron streaming. In addition, the maximum neutron flux on the side surface of the helical coil increased to 1.1 x 10^{12} n/cm^2/s near the outer support structure. From comparison to the neutron flux of ~3 x 10^{10} n/cm^2/s obtained for the geometry without the support structure (Fig. 6), this is considered due to reflection of neutrons from the support structures. By taking into account that the large ports of the support structures for blanket maintenance (Fig.1) have not been simulated in the present 3-D geometry data, the neutron reflection from the support structures would be lower in the actual reactor geometry. However,

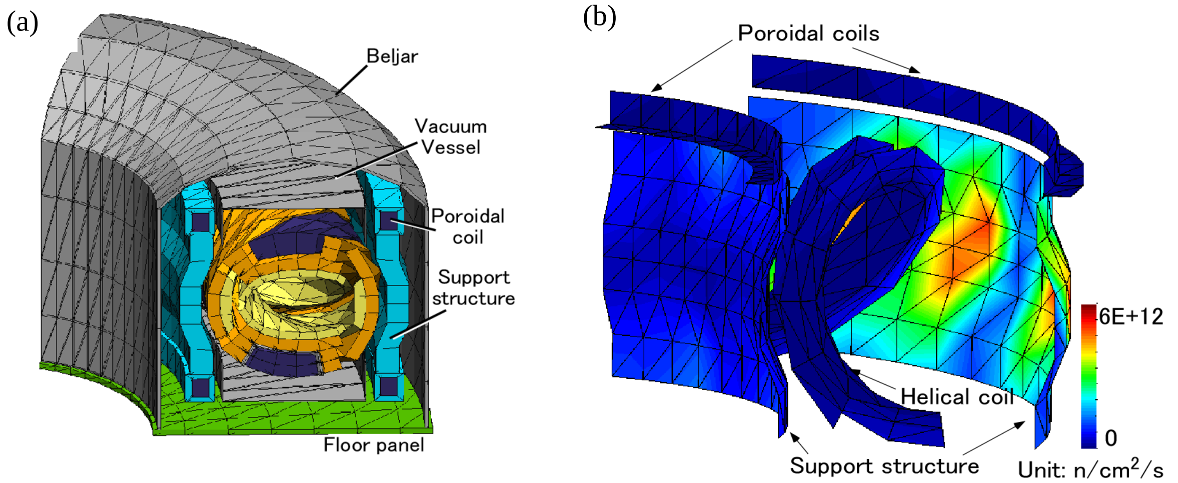

FIG. 7. (a) Addition of non-helical structures for investigation of neutron streaming and reflection. (b) Distribution of fast neutron flux on helical coil, troidal coils and support structures.

the values of neutron flux are indicating the necessity of enhancement of the shielding ability to reduce the reflection. This is one of most important key issues, and detailed design modifications will be presented in elsewhere [18].

4.3 Toroidal and helical effect on neutron wall loading

While the averaged plasma radius and neutron wall load have been assumed 1.73 m and 1.5 MW/m^2, respectively, in the design parameters of the FFHR2m1 design, the shortest distance between the troidal axis and the first wall is 1.45 m for the helical blanket system. The distribution of the neutron wall load on the first wall was calculated to evaluate the maximum value. Since it has been considered that the neutron wall load distribution would be affected by the plasma distribution shown in Fig. 1, the evaluation was performed by assuming the helically rotating plasma distribution with an elliptical cross-section. The long radius and short radius of the cross-section was 2.4 m and 1.8 m, respectively. The second order distribution was assumed for the ion density and temperature in the plasma. The intensity of the neutron source also in this investigation was fixed to 6.6 x 10^{20} n/s for the 360^o full torus geometry, which gives the neutron wall load of 1.5 MW/m^2 in the simple torus geometry shown in Fig. 2. From the results shown in Fig. 8, the maximum neutron wall load has been estimated to be 1.8 MW/m^2. Since the maximum neutron wall load of 2.0 MW/m^2 was obtained for the uniform torus-shaped neutron source shown in Figs. 4 and 5, comparison of the results indicated that the peaking factor of the neutron wall load on the first wall would be moderated by the plasma distribution in the helical reactor. The simulation of the helical-shaped plasma decreased the total TBR by ~4 % for the Flibe+Be/JLF-1 blanket system with the geometry shown in Fig. 4 (b) compared to the evaluation simulating the uniform torus-shaped neutron source. The decrease in the total TBR could be recovered by adjustment of the dimension, e.g. expansion of the breeder layers.

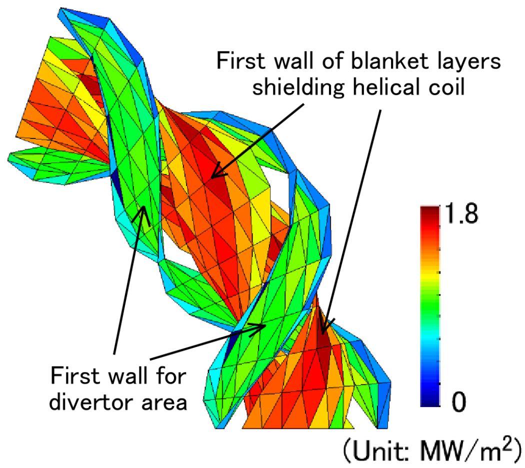

FIG. 8. Top view of calculated neutron wall load distribution. (72^o of the torus)

4.4 Improvement of calculation system

Quick feedback between investigation of the neutronics performance in the helical blanket system and design modification has been performed effectively by using the present 3-D neutronics calculation system. In the system, the helical and rotating structures of the reactor components could be simulated from input of small size data, which are the coordinates of the quadrangular meshes generated on the cross-section drawings of the components. However, the investigation of the shielding performance is indicating the importance of simulating the maintenance ports on the support structures, the localized ports for divertor pumping on the shielding layers etc. in further neutronics studies. Addition of the function to import CAD data is planed for the detailed simulation of the structures and ports. The target of the present development of the 3-D neutronics calculation system is to understand the neutron flow in the reactor system by meshing the vacuum areas around structures similarly to the reactor components and by 3-D visualization of neutron flux distribution.

5. Conclusion

Neutronics performance of advanced self-cooled liquid blanket system for the helical-type power reactor FFHR2 has been investigated in the design activity. In the present study, the 3-D neutronics calculation system has been constructed to simulate the helical configuration of the blanket system. The tritium breeding abilities of the four types of Flibe cooled and Li cooled blanket systems have been evaluated with 3-D blanket geometry data. After the modification to improve the ability, the total TBRs were 1.08 (Flibe+Be/JLF-1), 0.98 (Li/V-alloy) and 0.97 (Flibe/V-alloy), respectively. The three blanket systems would achieve the total TBR>1.0 by further design modification such as optimization of reflectors, dimension adjustment etc. The Flibe cooled STB (Spectral-shifter and tritium breeding blanket) system employing a thick carbon armor would also achieve the TBR > 1.0 in FFHR2 by reducing the thickness of the JLF-1 first wall to ~5 mm. In the investigation of the shielding performance against direct neutrons from the core plasma, the maximum fast neutron flux on the helical coil surface could be suppressed to ~3 x 10^{10} n/cm^2/s by enhancement of the side shield for the helical coils and expansion of the breeder layers. Further suppression of the flux would be possible by optimization of the shielding materials and compositions. However, it was indicated that the neutron flux on the helical coils dramatically increased to 1.1 x 10^{12} n/cm^2/s due to the neutron reflections from the support structures around the blanket system. The maximum neutron flux of 1.7 x 10^{11} n/cm^2/s on the poroidal coils also requires improvement of the shielding performance for the neutron streaming. Suppression of the neutron reflection and streaming is the most important neutronics issues in the helical reactor to be improved by modification of the shielding design. Investigation of the neutron wall loading indicated that the peaking factor of the neutron wall load distribution would be moderated by the toroidal and helical effect of the plasma distribution in the helical reactor.

By using the 3-D neutronics calculation system developed for the helical reactor design, quick feedback between neutronics evaluation and design modification has been performed successfully with small size input data. Investigation of the tritium breeding ability has indicated the configuration and dimensions to achieve the tritium-self sufficiency in the helical reactor. The neutron shielding performance in the helical reactor has been understood by the 3-D visualization of calculated neutron flux and the issues have been discussed quantitatively. After the present fundamental investigation of the neutronics performance, this study for the FFHR2 design will proceed into optimization in accordance with the studies from other aspects such as plasma confinement, divertor system, superconducting magnet system, blanket design etc.

6. Acknowledgement

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