SIII-6

原型炉設計に関する数値シミュレーション研究
Numerical Simulation Study on DEMO Design

星野　一生
Kazuo HOSHINO

原子力機構
JAEA

In DEMO design activity of BA-IFERC project, key design issues and options are being investigated for future DEMO reactor design. Most of design elements are under development and there are large gap from present knowledge and technology. Therefore numerical simulation has an important role. In project cycle 1 of CSC, the project "SSDEMO[1]" (Simulation Study for fusion DEMO Reactor Design) has been accepted. In the project, the power handling in the divertor and neutronics design have been investigated.

Investigation of power handling in the divertor

Huge power handling in the SOL/divertor region is one of the crucial issues for a tokamak fusion reactor. In the present scenario for power handling in tokamak DEMO reactor SlimCS [1], the exhausted power into the SOL/divertor region is expected to be as high as 500-600 MW, which is 5-6 times larger than that of ITER. On the other hand, the desirable heat load on the divertor target is well below 10 MW/m^2 from the engineering point of view, being lower than that of ITER.

Divertor design of SlimCS DEMO reactor has progressed using a suite of integrated divertor codes SONIC [2, 3] with the non-coronal impurity radiation model. Recently, SONIC code has been improved, and then the impurity Monte-Carlo model coupled with SONIC becomes available for DEMO divertor simulation. While the impurity transport can be treated appropriately, the required computer resources increase. Therefore, CSC is necessary for wide range parameter survey and high-precision simulation.

Improvement of the divertor geometry is important in order to reduce the peak heat load. In the project cycle 1, effect of "longer leg divertor" was investigated [4] for the SlimCS, where the divertor plasma temperature T_d is expected to decrease because of longer connection length to the target ($L_{//}$), i.e. $T_d \propto L_{//}^{-4/7}$ from a simple 2-point model. Outer divertor leg is extended to 2.5 m, while the magnetic flux expansion at the target is reduced to a half compared to the reference case. Figure 1 shows the impurity radiation distribution in the outer divertor region. In the reference case, the large impurity radiation region is close to the target. The divertor heat load by the impurity radiation is about 5.4 MW/m^2 at the peak. Total heat load including contributions from the plasma transport, the surface recombination, the impurity radiation and the neutral transport, is 19 MW/m^2 as shown in Fig. 2.

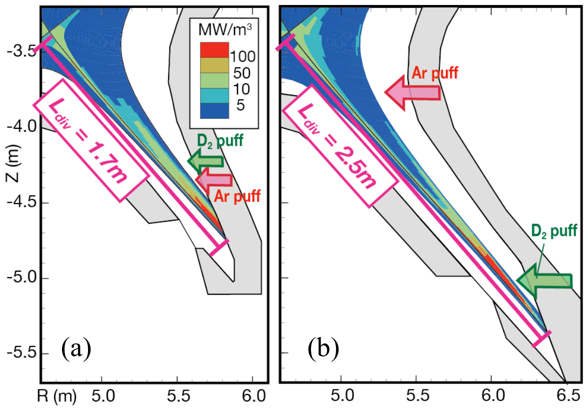

Fig. 1 Spatial profile of the impurity radiation power. (a) reference case and (b) longer divertor leg case.

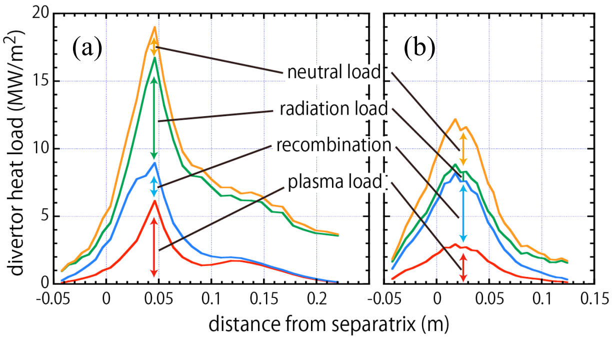

Fig. 2 Divertor heat load along the outer divertor target. (a) reference case and (b) longer divertor leg case.

[1] Member of SSDEMO project:
K.Hoshino, Y.Someya, N.Asakura, S.Tokunaga, H.Utoh, M.Nakamura and K.Tobita

In the longer divertor leg case, the large radiation region moves further upstream from the target, and both T_i, T_e and n_e at the divertor decrease. As a result, peak heat load decreases to 12 MW/m^2 due to a reduction of heat loading by the plasma transport and impurity radiation. Effect of the long field line is efficient for a reduction in the peak heat load, while the reduction in the magnetic expansion.

In addition to investigation of the divertor geometry, numerical analysis for improvement of the detachment modeling, surface erosion etc., were carried out on the CSC Helios super computer.

Neutronics Design

In DEMO design, there is a conflicting requirement regarding the gap between blanket modules. In order to ensure the self-sufficient production of tritium, the gap being a non-breeding zone must be minimized. In addition, such a minimized gap is favorable to suppress neutron flux streaming and to protect the superconducting coils against neutron/gamma-ray irradiation as well. On the other hand, the gap between neighboring blanket modules is necessary to allow a thermal distortion of blanket during the operation and remote handling access for blanket replacement. Moreover, for arrangement of plasma diagnostics and NBI/ECW port for fueling and current drive, larger spaces between blanket modules are necessary. From these viewpoints, the gap width is an important “trade-off” problem in DEMO design.

In this study, the trade-off problem is studied by using the three-dimension Monte Carlo N-particle transport code, MCNP-5 [5]. The neutron and gamma-ray fluxes and tritium breeding ratio (TBR) is calculated in three-dimensional precise geometries including blanket, cooling tubes, back-plates, shield, cryostat, superconducting magnet and structural materials. Moreover, allowable spaces for diagnostics and NBI/ECW ports are evaluated in light of tritium breeding ratio. Important findings obtained from the first cycle are that 1) higher TBR is anticipated for lower neutron wall loading (NWL) because of a relative increase of the breeding zone as a result of reduced cooling channels at lower NWL, and that 2) when the gap between the neighboring blanket modules is changed from 0.5 cm to 10 cm, decrease in the calculated TBR is little for the gap of less than 4 cm. Such an allowance of the gap will facilitate access of remote handling equipment for replacement of the blanket modules and improve access of diagnostics.

In addition, a position of the diagnostics and a maintenance process of component inside the reactor are studied. For evaluation of dose map, calculation of neutron and gamma-ray fluxes in precise 3D reactor geometry is necessary. In the 1st cycle, the preliminary calculation was carried out. Figure 3 shows the 3D model of fusion DEMO with a fusion output of 3 GW for MCNP-5 calculation and the neutron flux map above 100 keV during the operation. The large neutron streaming in the NBI duct can be seen. Decrease in the neutron flux along the duct is only 3 orders of magnitude. Therefore, dose ratio after the operation increases around the NBI duct.

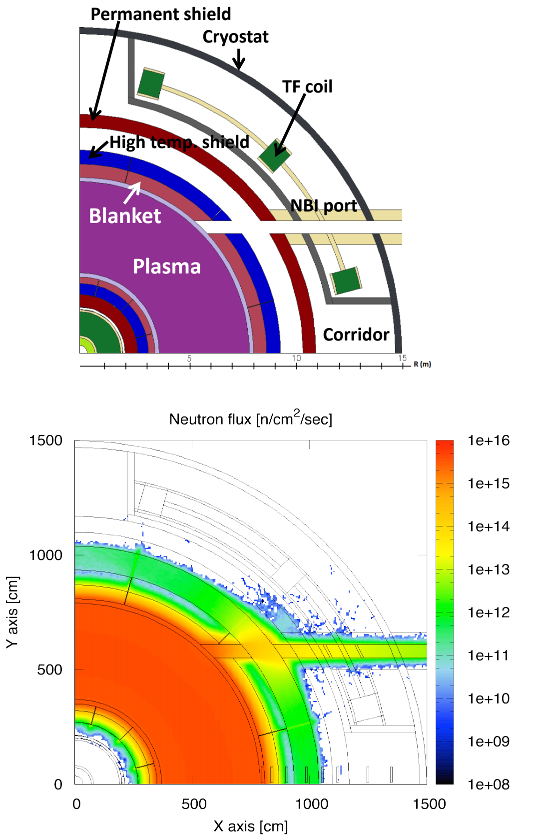

Fig. 3 Horizontal cross section of 3D calculation model (up) and neutron flux map above 100keV (down)

References

[1] K. Tobita, *et al.*, Nucl. Fusion **49** (2009) 075029.

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[4] N. Asakura, *et al.*, 24th IAEA FEC, San Diego, USA, Oct. 2012.

[5] X-5 Monte Carlo Team, MCNP-A General Monte Carlo N-Particle Transport Code, Version 5, Vol. II: Users Guide, La-CP-03-0245.