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Physics and Engineering Studies of the Advanced Divertor for a Fusion Reactor & DEMO Concept Development and Assessment of Relevant Technologies

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A
Magnetic configurations of advanced divertor, i.e. super-X (SXD) and snowflake, were recently proposed, and the concepts have been demonstrated in experiments. For the application to the Demo reactor, engineering design as well as the plasma performance should be determined. A short super-X divertor (short-SXD) is proposed as a new option for Demo divertor, where field line length from the divertor null to the outer target was largely increased (more than two times) in a similar size of conventional divertor. Physics and engineering design studies of a fusion reactor with the short-SXD installed at the outer divertor have progressed. Minimal number of the divertor coils (1 or 2) were installed inside the toroidal field coil, i.e. interlink-winding (interlink). Arrangement of the poloidal field coils (totally 8 or 9) and their currents were determined, taking into account of the engineering design such as vacuum vessel and the neutron shield structures, and maintenance scenario of the divertor and blankets. Divertor plasma simulation (SONIC) showed that large radiation region is produced between the super-X null and the target, and the plasma temperature becomes low (1-2 eV) both at the inner and outer divertors, i.e. fully detached plasma was obtained efficiently.

B
Recent development of a DEMO concept with a medium size (major radius of ~8.2 m) and a lower fusion power (~1.5 GW) is presented together with assessment of relevant technologies. The maximum toroidal field is evaluated at ~13 T, which is nearly independent on strand materials (Nb3Sn or Nb3Al) unlike a compact DEMO, while the increase of the allowable design stress has a large impact on that. The divertor simulation study indicates that the tolerable level of divertor heat flux (~5 MW/m^2) is foreseeable for the medium size DEMO, and the design study of short super-X divertor as an option is progressing to efficiently obtain the fully detached plasma. The assessment of various maintenance schemes indicates that the vertical port maintenance scheme provides advantages in the easy handling, the layout of poloidal coils, the size of toroidal coils, and separate maintenance of the in-vessel components. Finally, the study of the waste management suggests that the ratio of the radioactive waste to be disposed of in shallow land burial can be increased thanks to the lower fusion power.

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Primary authors: Mr ASAKURA, Nobuyuki (Japan); Dr SAKAMOTO, Yoshiteru (Japan Atomic Energy

Agency)

Presenter: Mr ASAKURA, Nobuyuki (Japan)

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