Concept of Compact Low Aspect Ratio Demo Reactor, SlimCS

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Abstract

A new concept realizing a compact demonstration reactor named "SlimCS" is presented. This is a compact reactor concept with a reduced-size center solenoid (CS), which can be envisioned by considering a major function of CS as plasma shaping rather than poloidal flux supply. Such a consideration on CS is reasonable when the plasma current is non-inductively maintained in steady state. SlimCS is as small as advanced commercial reactor designs (such as ARIES-RS and CREST) and can produce 1 GWe in spite of conservative design parameters. The reduced-size CS enables us to introduce a slender toroidal field (TF) coil system which contributes to reducing the weight and construction cost of the reactor. Moreover, SlimCS has an advantage of expanding the design window of fusion reactors to lower aspect ratio of around 2.5 which facilitates higher elongation and higher beta access with reasonable design margins.

1. Concept of reactor

In steady state operation of a tokamak reactor, the most important function of CS is plasma shape control rather than poloidal flux supply. This means that assuming reliable current ramp technology with non-inductive current drive, the CS diameter can be reduced as long as a sufficient CS current for shaping is obtained. This is because, in the conventional tokamak reactor design, the CS diameter is determined from the required volt-second for plasma current ramp-up plus extra flux. If one reconsiders fusion reactor design from this point of view, the CS radius is dramatically reduced. The fact that higher CS current density is expected with decreasing the CS diameter helps this reduction. Such a down-sized CS affects TF coil design, eventually contributing to a reduction of the tokamak reactor size.

As known by the Virial theorem, the weight of a TF coil system increases with its magnetic energy. Since the coil system usually accounts for a significant part of the reactor construction cost, a light (or low magnetic energy) TF coil system is required to reduce the cost. On the other hand, high field is also required for the TF coil system to attain high power density. An assembly of TF coils with a small inner leg radius (R_{TF}) can meet these requirements simultaneously[1].

We have been conceiving the demonstration reactor SlimCS using the reduced-size CS with an outer radius of 0.7 m which has the capability of plasma shaping (triangularity of ~0.4) [2] probably enough to obtain high confinement in high density region and possibly to avoid giant edge-localized modes.

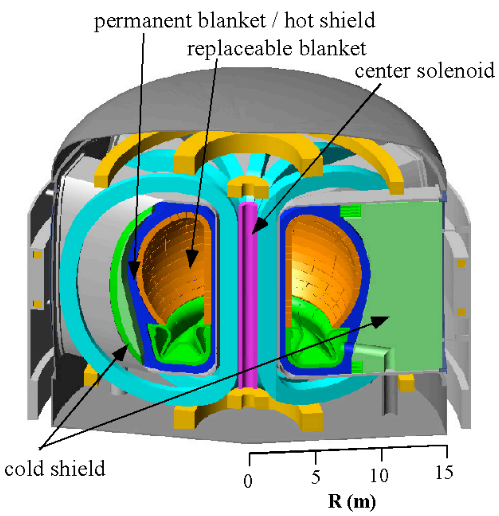

Fig.1 Conceptual view of SlimCS

2. Design parameters and technologies

SlimCS produces a fusion output of 2.95 GW with a major radius of 5.5 m, aspect ratio (A) of 2.6, normalized beta (β_N) of 4.3 and maximum field of 16.4 T. The conceptual view is depicted in Fig.1 and the design parameters are listed in Table 1. Note that the position of two poloidal coils outside the cryostat is under reconsideration to increase compatibility with sector transport maintenance scheme. It is expected that the zero output at the sending end is obtained at $\beta_N = 2$, $n/n_{GW} = 0.4$ and $f_{BS} = 0.35$ and that a commercial output level of 1 GWe is produced at $\beta_N = 4.3$, $n/n_{GW} = 1.1$ and $f_{BS} = 0.77$, where n/n_{GW} and f_{BS} are the line-averaged electron density against the Greenwald density and the bootstrap fraction, respectively. SlimCS uses technologies foreseeable in 2020's such as Nb_3Al superconductor, water-cooled solid breeder blanket, a reduced activation ferritic martensitic steel F82H as the blanket structural material, and tungsten monoblock divertor plate. Neutron wall load is designed at 3 MW/m^2. Divertor heat flux, which can be a critical issue for such a compact reactor, is mitigated to 10 MW/m^2 at the peak by small inclination (12-15°) of divertor plates and flux-tube expansion in the divertor region.

Table 1 Design parameters of SlimCS

Major radius (m)	5.5	Poloidal beta, β_p	2.57
Aspect ratio, A	2.6	Normalized beta, β_N	4.3
Plasma current, I_p (MA)	16.7	Temperature, $\langle T_e \rangle$ (keV)	17.0
On-axis magnetic field (T)	6.0	Density, $\langle n_e \rangle$ (10^{20} m^{-3})	1.15
Elongation, κ_{95}	2.0	Confinement Enhancement, HHy2	1.3
Triangularity, δ_{95}	0.4	Bootstrap current fraction, f_{BS} (%)	77
Safety factor, q_{95}	5.4	Current drive power (MW)	60-100
Plasma volume (m^3)	941	Fusion output (MW)	2,950
Toroidal beta (%)	5.76	Neutron wall load (MW/m^2)	~3

3. Features of SlimCS

3.1 Advantages

As seen in Fig.2, even with the assumption of relatively conservative plasma parameters, SlimCS is as compact as advanced commercial reactor designs such as ARIES-RS [3] and CREST [4]. This is based on the characteristic that such a low-A plasma, being stable for higher elongation (κ), can have higher n_{GW} and β_N^{lim}. In the figure, κ^{lim} stands for the vertically stable κ limit for not using in-vessel coils. β_N^{lim} is the beta limit for fully bootstrap-driven plasma [5]. Note that β_N / β_N^{lim} can exceed unity when the plasma current profile is optimized with external current drive as assumed in ARIES-RS and CREST. Another merit of low-A is that the first wall area on the low field side, where smaller electromagnetic (EM) force acts on disruptions, is wide compared with that of conventional-A. This means that tritium can be efficiently breeded with large blanket modules on the side. As a result, the demand for tritium breeding on the high field side is comparatively reduced so that small blanket modules, being robust to larger EM force but less efficient for tritium breeding, can be arranged on the side.

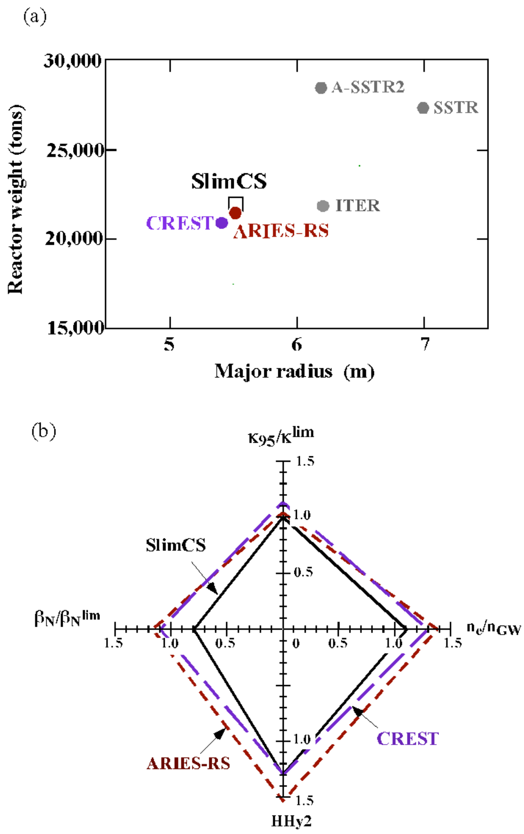

Fig.2 (a) Comparison of major radius and reactor weight for various fusion reactors, and (b) comparison of plasma parameters between SlimCS and compact commercial reactor designs (ARIES-RS and CREST).

3.2 Difficulties

Major issues of SlimCS are non-inductive plasma current ramp and plasma physics basis around A = 2.6. Although the reduced-size CS can control the position of x-point and divertor hit points independently of plasma current, the capability of inductive plasma current ramp is restricted to only 3.8 MA. Accordingly the plasma current must be raised using an overdrive with a combination of bootstrap current and non-inductive external current drive (CD). This technique is considered to be a continuation of the steady state operation technique of tokamak. As to the other issue on the low-A physics basis, NCT [6] designed to cover A = 2.6-3.1 will play an important role to resolve it.

4. Physics design

4.1 MHD equilibrium

Taking advantage of vertical stability in low-A, SlimCS is designed to have high elongation as possible. On the other hand, the designed A of 2.6 is not low enough to lead to natural elongation. In consequence, SlimCS adopts CS to attain sufficient elongation as well as triangularity: κ_{95} = 2.0 and δ_{95} = 0.4. The designed triangularity is held down to some extent not to impair neutron shielding on the inboard divertor chamber side although higher δ_{95} may be favorable for advanced operation. The equilibrium is designed to have single null divertor considering an advantage in steady control of divertor radiative cooling and pumping. An important point to note in low-A reactor design is that the inboard SOL width increases with reducing A [7]. In the case of A = 2.6, the width of inboard SOL which is determined from the flux surface corresponding to the 3 cm outboard SOL is 13 cm. The inboard SOL width is considered in the determination of the radial build. Originally, two equilibrium coils out of eight were located near the outer equatorial plane but with subsequent design study, the plasma with κ_{95} = 2.0 and δ_{95} = 0.395 was also produced in the case of the vertical displacement of these two coils away from the equatorial plane (Fig.3), which is compatible with sector transport maintenance. However, the position of the outer-most poloidal coils should be shifted toward smaller R so that the cryostat can contain these coils.

4.2 Plasma profiles

Since plasma design parameters are

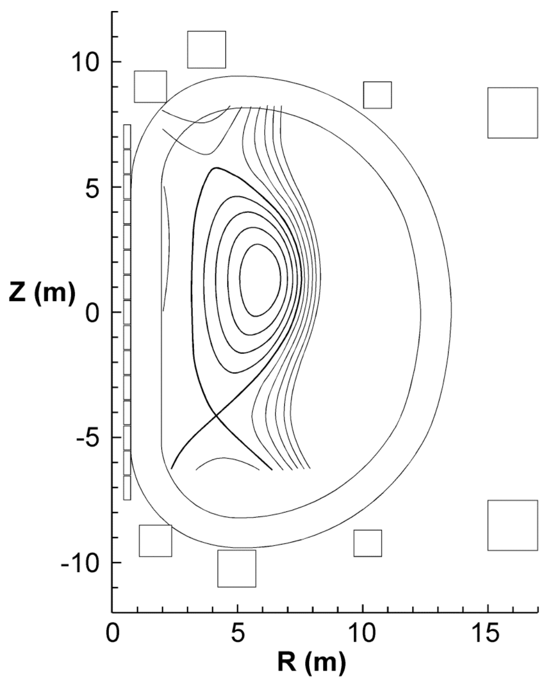

Fig.3 MHD equilibrium produced by poloidal coil configuration compatible with sector transport maintenance

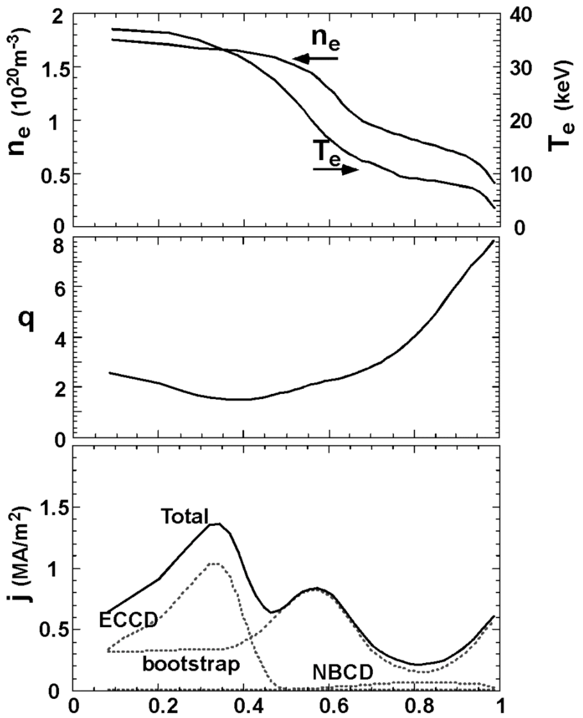

Fig.4 Example of weakly RS with consistency between pressure and current profiles. The central current by ECCD is important to attain the designed bootstrap current fraction.

determined by a systems code based on a point model, the parameters should be checked for correctness using one dimensional (1-D) codes. For this purpose, an ACCOME code [8] was used to review the consistency between the assumed plasma profiles and key parameters such as P_{fus}, β_N, n/n_{GW}, HHy2 and f_{BS}. In the 1-D analysis, we attempt to find a solution with a q-profile other than strongly reversed shear (RS). This is because strong RS will be not appropriate as a standard operation mode of SlimCS from the points of view of disruptivity and the controllability of q in the central region.

Figure 4 shows a solution for weakly RS. Although the profile is not reasonably optimized because of a single ECRF beamline, the following information was obtained from the result:

1) Most of the design parameters by the point model are consistent with the 1-D calculation;
2) The location of NBCD is restricted to the peripheral region because of beam attenuation;
3) Use of ECRF as the main CD tool requires a high CD power input due to its lower current drive efficiency than NBCD.

In addition, the calculation indicates that q-profile control by ECRF in the central region is important to maintain the bootstrap fraction around the design value for various density and temperature profiles. For example, suppose the case that the BS current around an internal transport barrier is dominant. Then f_{BS} is strongly dependent on q-value around the ITB, i.e., the total current driven inside the ITB. For this reason, q-profile control is a key technology to maintain f_{BS} at a design value in fusion plasma especially with high f_{BS}. In connection to this, the interplay between q profile and pressure profile (including ITB structure) will be an essential issue governing the controllability on f_{BS}. A concern about the analysis is consistency between the obtained q profile and the given density/temperature profiles. This is an open question to be resolved with further understanding on plasma transport.

4.3 Divertor power handling

A rough metric for divertor power handling is P/R_p where P denotes the input power due to alpha heating and CD power. Hence divertor power handling is a more serious problem in a compact reactor. The divertor plate of SlimCS consists of monoblock armors of tungsten and water-cooling ferritic steel tubes [9]. The prime constraint in power handling is an allowable heat flux (about 10 MW/m^2) of the divertor plate, being lower than the design value of ITER (~20 MW/m^2). Thus, higher emphasis must be placed on radiative cooling in SlimCS because alpha heating is six times as high as that in ITER.

Radiation from the main plasma, which is the summation of Bremsstrahlung, synchrotron and impurity line radiation, is effective at lessening the requirement for radiative cooling in the divertor. For the parameters of SlimCS, Bremsstrahlung and synchrotron radiation are estimated to be 48 MW and 22 MW, respectively. Impurity line radiation being dependent on impurity content is assumed to be about 70 MW, which would be obtained by argon injection at the content of 0.2%. The total radiation power from the main plasma is designed at 140 MW. Neon injection is not favorable as impurity because the fuel dilution becomes lower than 80% to reach 70 MW of radiation power.

It is assumed that the 70% of SOL input is radiated in the divertor. In this condition, the allowable heat flux of 10 MW/m^2 (peak) at the divertor is satisfied when the effective area of the divertor plate is widen by flux-tube expansion (5-9×) and shallow inclination (inboard side 30º, outboard side 15º) of the divertor plate to field lines.

In order to suppress the physical sputtering of the divertor armor, a reduction in the divertor temperature is required. Considering the sputtering of tungsten armor by argon ions, the divertor temperature should be lower than 10 eV. On the other hand, detachment should be avoided to maintain effective evacuation of helium ash. After all, the target temperature in the divertor is 5-10 eV, which can be realized by the particle flux multiplication as high as ~150 at the plate.

4.4 Toroidal field ripple

The TF coil size is determined to be compatible with sector transport maintenance. This means that the TF coils should be oversized. Since an excessive reduction of TF ripple is not necessary from the viewpoint of ripple loss, the number of TF coils of SlimCS is reduced to twelve so as to have the ripple amplitude of 0.3% at the most on the plasma surface. Generally, in low-A tokamaks, TF ripple amplitude sharply damps with distance from the rim of TF coil [10]. As a result, the ripple loss of SlimCS can be minimal compared with conventional tokamak designs that have similar TF ripple. In fact, Monte Carlo calculations indicates that the alpha particle ripple loss is as small as 0.06±0.03% for the standard weak RS profiles with q(0) = 2.5. The loss is only 0.19±0.1% even for strong RS with q(0) = 7. The loss power of alpha particles is 1 MW at the most, probably leading to an acceptable heat load on the wall.

5. Engineering design and research issues

5.1 Torus configuration

The conceptual torus configuration of SlimCS is illustrated in Fig.5. As discussed in Sec. 5.3, a compact reactor like SlimCS requires segmentation of blanket to setup an inbetween conducting shell for vertical stability and high beta access. For this reason, the blanket is separated into replaceable and permanent blanket by the conducting shell. The replaceable blanket is further segmented into small modules whose support has leaf springs to allow differential thermal expansion between the blanket module and the anchor plate. To meet the requirement that the conducting shell must be sector-wide, sector transport maintenance is adopted so as to allow the maintenance check and repair of the permanent blanket behind the conducting shell.

The sector transport maintenance provides significant advantages in 1) compatibility with the sector-wide conducting shell, 2) accessibility for maintenance and repair, and 3) extensibility for replacement with an advanced core component. From an opposite point of view, in-vessel maintenance scheme is excluded in SlimCS due to its practical incompatibility with the complex blanket configuration with the inbeween conducting shell. Contrarily, the sector transport maintenance has critical issues in dispute regarding 1) the feasibility of a enormous transport cask with double-seal doors, 2) transport and anchoring mechanism for the sector, 3) how to support the turning-over force of TF coils without outboard shear panels, and 4) the feasibility of a hot cell containing a set of activated sectors. The adoption of the

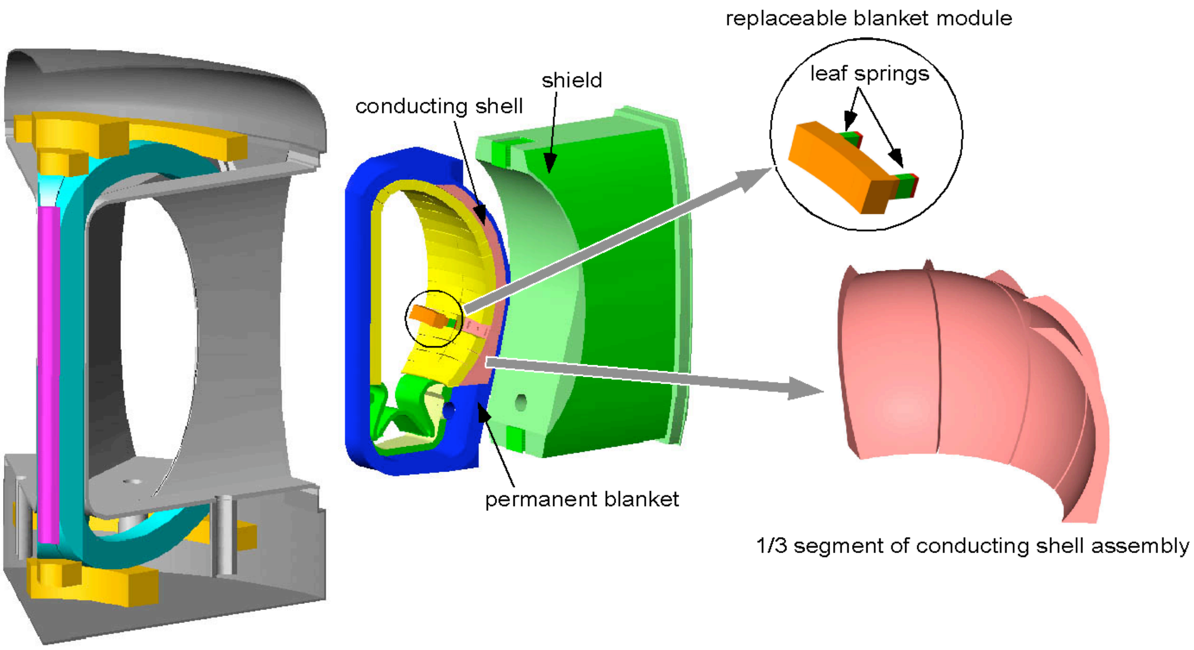

Fig.5 Concept of torus configuration

in-vessel maintenance scheme for ITER has created the present situation in which there are no development programs of the sector transport maintenance. However, possible solutions to these issues should be vigorously explored because the sector maintenance scheme has an economic impact for commercial plants by improving system availability [11].

5.2 Superconducting magnets

For the reason described in Sec. 1, R_{TF} has an important impact on the stored energy and weight of TF coils. On the other hand, these benefits are obtained by a tradeoff of the maximum field. By assuming the use of Nb_3Al conductor for SlimCS, the achievable magnetic field was estimated as a function of R_{TF} [12]. In the estimation, grading the windings is considered to compensate for a decrease in the critical current density of Nb_3Al wire. In view of the tradeoff, TF coils are designed to have the maximum field of 16.4 T at R_{TF} = 2.0 m. The stored energy is as low as 50 GJ, being smaller than SSTR (140 GJ, B_{max} = 16.5 T) [13]. A major technical issue of the TF coils is how to suppress a displacement caused by the turning-over force. This is because the sector transport maintenance does not allow the placement of shear panels between neighboring TF coils across a wide area on the outboard. To resolve the displacement problem, inbetween three-dimensional supports are considered. The supports are set up in the upper part of TF coils and suppress the toroidal displacement of TF coils by maintaining the inter-coil gap at a constant distance.

The superconductor for CS is Nb_3Sn, which is operated at 13 T. The radius of the outer-most winding is 0.7 m. The gross current density of CS is 29 A/mm^2 and the ampere-turn is 10 MAT/m producing a magnetic field sufficient for plasma shaping of δ_{95} ~ 0.4. As to CS, further investigation is needed on 1) a standby CS current for steady state operation and 2) the minimization of CS segmentation. CS is most efficient to maintain the plasma current constant using its quick-response feature. For full use of this feature, CS should be ready to properly respond to both overdrive and underdrive conditions of non-inductive current drive. This means that in most instances the CS current should be 60-80% of the maximum CS current in preparation for occasional flux swing. Furthermore, the poloidal coil system has the capability of controlling the null point and divertor hit points in place on the occasion of CS flux swing for I_p-constant control. Because of the limited space for CS, feeder-routing to inner CS segments is a difficult engineering issue. Minimizing the number of CS segmentation mitigates the difficulty, which may limit the controllability over plasma shaping in a tradeoff.

5.3 Blanket

Generally, the blanket segmented into several hundred of modules do not have the function of conducting walls. In a reactor with the minor radius (a) of 2.5 m or larger (like PPCS Model C and D [14]), the backplate supporting the blanket modules play the role of a conducting wall. In contrast, the backplate of SlimCS with a = 2.1 m is located too far to produce the effect reaching the designed β_N of 4.3. Accordingly, the outboard blanket of SlimCS is designed to consist of 0.3 m thick replaceable and 0.6 m thick permanent blankets and toroidally 12-segmented conducting walls are located inbetween so as to form a conducting surface at the position of r_W/a = 1.3, where r_W denotes the distance of the conducting wall surface from the plasma center. Functionally, a toroidal assembly of the conducting shells shown in Fig.5 is considered as the conducting wall favorable to vertical stability and high β_N access [15]. Incidentally, each fin of the conducting shells works to cancel harmful components of eddy current loops by superposition with the eddy current passing on the neighboring fin.

We envision the blanket for Demo as an extension of the present R&D program based on water-cooled solid breeder blanket [16]. The structural material is low activation ferritic steel, F82H. As described in Sec. 3, since the first wall area on the outboard side is wide compared with conventional A (inboard 27%, outboard 73%), the demand for tritium breeding on the high field side is comparatively reduced. This leads to the breeding blanket concept consisting of small inboard blanket and large outboard blanket modules. The concept is reasonable because comparatively large modules are acceptable on the outboard with ~4.2 T in light of the EM force acting on disruptions. In contrast, on the inboard side where B_T can be as high as 10T, the modules must be small enough to be robust to the EM force. Although the dimension of the blanket modules on both sides is under study, possible combinations of breeding materials were investigated. The result for three candidates is summarized in Table 2. All these three options are based on water-cooled solid breeder blanket. Although Option C without inboard breeding blanket was proposed as a new concept possibly matching low-A, the concept is ruled out from the candidates from the viewpoint of the net tritium breeding ratio (TBR). Here the coverage of effective breeding region was estimated using the data of the water-cooled pebble blanket in Ref [16].

Table 2 Considered blanket options and evaluation

Option	Constituent	Features	Evaluation
A	•Inboard:Li_2TiO_3/ $Be_{12}Ti$ *0.4m thick* • Outboard: same *0.9m thick*	• $Be_{12}Ti$: low swelling and resistant to Be-water reaction • Least robust to EM forces • Local TBR = 1.35	Good (Net TBR = 1.05)
B	•Inboard:Li_2TiO_3/ Be *0.2m thick* • Outboard: same *0.9m thick*	• Be: swelling by irradiation and concern about Be-water reaction • More robust than Option A • Local TBR = 1.35	Good (Net TBR =1.05)
C	• Inboard: Pb reflector *0.2m thick* • Outboard:Li_2TiO_3 / Be *0.9m thick*	• No breeder on inboard • Most robust to EM forces • Local TBR = 1.25	No good (Net TBR = 0.97)

5.4 Current drive system

ECRF and NBI are considered as CD systems. As to ECRF, the required frequency is around 190 GHz so that the ITER 170 GHz gyrotron can be used with reasonable extension [17]. Because of its outstanding positional controllability of power deposition, ECCD is of great value as a CD method in the central region. CD in the central region dramatically changes q-profile, perhaps effectively controlling the bootstrap fraction at a target value. On the other hand, compared with NBI, the current drive efficiency of ECCD is roughly a half of that of NBI at given n_e and T_e.In order to overcome this difficulty, an improvement in system efficiency up to a theoretical value (~70%) is desirable.

From the point of view of system efficiency, NBI should adopt electrostatic acceleration in which around 50% of system efficiency is foreseeable in 2020-2030. Based on this acceleration, the beam energy of NBI is restricted to be lower than 2 MeV [18] and thus the beam mainly deposits in the peripheral region at standard parameters of SlimCS. In spite of such a peripheral deposition, the electrostatic acceleration is favorable compared with the higher energy NBI based on RFQ acceleration in terms of system efficiency. From the point of view of reactor design, there are three points to be considered regarding CD tools: 1) interference with the maintenance space; 2) compatibility with shielding around the injection port; 3) impact of port size on TBR. In order to avoid the interference with the sector maintenance route, the NBI port must be transported to elsewhere in the torus hall. In addition, the NBI port including the neutron shield becomes as large as in ITER. These two issues seems to be critical. On the other hand, according to our estimation, the ratio of the non-breeding zone to the blanket coverage is 11% (71 m^2). The foreseeable power density at the port for ECRF and NBI in the DEMO stage are ~300 MW/m^2 and ~50 MW/m^2, respectively. Although the power density of NBI is comparatively low, the required port size is not critical in light of TBR even for 100 MW injection.

6. Summary

Regarding a major role of CS as plasma shaping, one can reduce the CS size of a tokamak reactor. Such a reduced CS enables to decrease the stored energy of TF coil system, eventually leading to a compact low-A tokamak reactor. The Demo reactor concept SlimCS is based on this idea, which is as compact as advanced commercial reactor designs such as ARIES-RS even with the assumption of relatively conservative plasma parameters. Although design basis for such a compact low-A (A=2.6) reactor is at a premature stage, these notable features can be a motivation for further concept study. Research issues posed to the reactor concept are plasma design being reasonable in the DEMO stage, divertor power handling, blanket concept, maintenance scheme, etc. It should be noted that a feasible reactor structure concept with conducting shells in between replaceable and permanent blanket is a challenge facing the design study of a compact high β_N reactor like SlimCS. In order to resolve these issues, different concepts for constituent technologies will be considered in the Broader Approach.

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