Journal of Nuclear Materials 386–388 (2009) 888–892

Contents lists available at ScienceDirect

Journal of Nuclear Materials

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

Torus configuration and materials selection on a fusion DEMO reactor, SlimCS

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A R T I C L E I N F O

PACS:
28.52.−s (Fusion reactor)

A B S T R A C T

SlimCS is the conceptual design of a compact fusion DEMO plant assuming technologies foreseeable in 2020–2030s. For continuity of blanket technology from the Japanese ITER-TBM, the prime option of blanket is water-cooled solid breeder with Li_2TiO_3 (or Li_4SiO_4) and Be. A reduced-activation ferritic–martensitic steel (RAFM) and subcritical water are chosen as the structural material and coolant, respectively. The reactor has somewhat complex torus configuration with a sector-wide conducting plate slipped in between the replaceable (front) and permanent (back) blanket. In order to allow flexibility of maintenance in such a complex configuration, sector transport hot cell maintenance scheme is adopted. This paper describes characteristics of SlimCS with a focus on materials selection.

1. Introduction

SlimCS is a DEMO reactor concept featuring a low aspect ratio (A) of 2.6 with a downsized central solenoid [1,2]. In such low-A regime, the construction cost of a reactor can be potentially reduced. The reactor produces a fusion output of 2.95 GW with a major radius of 5.5 m and a maximum field of 16.4 T. The main design parameters of the reactor are listed in Table 1. In order to take advantage of low A, the reactor has characteristic toroidal coil system, torus configuration and blanket. This paper describes these components in terms of materials.

2. Operational scenario

In the planning of the operational timetable of DEMO, there are three points to be considered, which are all related with materials.

(1) Early power generation.
In order to acquire an important share in the energy market in the end of this century, early realization of fusion power (preferably, 2030s) is desirable [3]. At least, "zero output" in the net electric power should be demonstrated shortly after the start of operation. This means that DEMO should be designed with available materials at the present.
(2) Need for checkout of in-vessel components in the initial irradiation.
By the operation of DEMO, IFMIF will deliver the results of material irradiation test at several tens of dpa. Regardless of the results, however, scheduled checkouts of in-vessel components will be necessary in the first phase of operation. In particular, the plasma facing components suffer the simultaneous irradiation of neutrons and ions, whose impact on material will not be well understood at that point in time.
(3) Necessity of test phase for advanced blanket.
Other than RAFM, there is no material option that can be used as structural material of DEMO blanket in the early stage of operation. However, it is likely that an advanced material such as SiC/SiC and/or a high performance blanket will have been developed during the lifetime of DEMO. The material and/or blanket should be tested to make fusion energy more attractive, as pointed out by Ref. [4].

Fig. 1 is a provisional operation schedule of SlimCS that can satisfy the above requirements. Based on the schedule, fusion power generation above "zero output" will be demonstrated in five years after the first plasma. Since the DEMO is regarded as the first component test facility in a sense, the in-vessel components should be checked out once a year. Exchange of the replaceable blanket is planed when the average neutron wall loading reaches 11–12 MWa/m^2, corresponding to the average irradiation of about 90 dpa (120 dpa in peak) for RAFM of the replaceable blanket surface. In the second decade of operation, an advanced (or high performance) blanket can be introduced for the test toward the commercial stage. In order to avoid a large-scale replacement of activated piping and plant facilities, however, a small part of the original blanket should be replaced with the advanced one. Considering the introduction of advanced blanket, sector transport maintenance is favorable for a replacement of piping for coolant and tritium recovery in the hot cell. The average

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doi:10.1016/j.jnucmat.2008.12.197

Table 1
Main design parameters of SlimCS.

Major radius, R_p (m)	5.5
Minor radius, a (m)	2.1
Aspect ratio, A	2.6
Plasma current, I_p (MA)	16.7
On-axis magnetic field, B_T (T)	6.0
Maximum field, B_{max} (T)	16.4
Plasma volume (m^3)	941
Fusion output (GW)	2.95
Average neutron wall load (MW/m^2)	~3

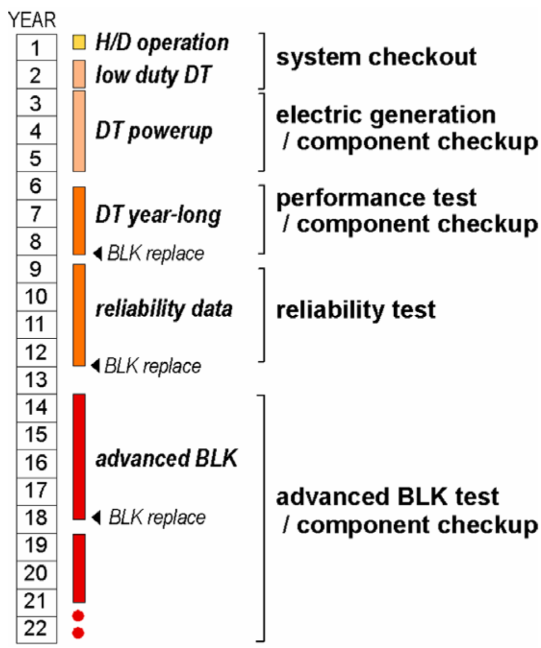

Fig. 1. Provisional operation schedule.

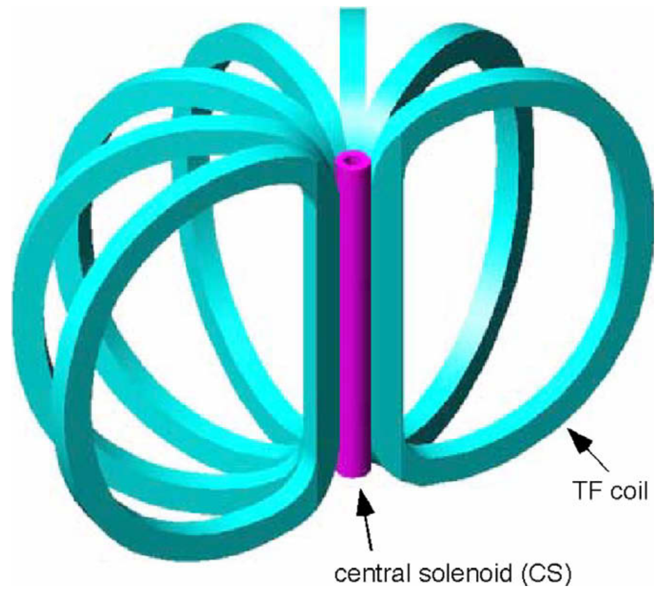

Fig. 2. Arrangement of TF coils and central solenoid in low aspect ratio tokamak.

availability throughout the life of reactor is anticipated to exceed 70%.

3. Superconducting magnet

Based on a point model formulation, the fusion power density (p_{fus}) is scaled as $p_{fus} \propto \beta^2 B^4$ with the beta value of plasma (β and the toroidal magnetic field (B). Therefore, high field is favorable to attain high fusion output for a given plasma volume. In a conventional tokamak reactor design, however, high field leads a challenge of supporting the resulting enormous centering force of toroidal field (TF) coils. For this reason, PPCS [5] and ARIES-AT [6] are designed at relatively low field: PPCS and ARIES-AT have the peak field (B_{max}) of 13.1–13.6 T and 11.4 T, respectively. In contrast with these designs with conventional A, a low-A reactor such as SlimCS can be designed at a high field without suffering the problem of the enormous centering force [7]. This is because the inner legs of TF coils are concentrated near the central axis of torus as shown in Fig. 2 and thus the field decays rapidly with increasing R. As a result, the stored energy of TF coil system is reduced dramatically.

For SlimCS, rapid-heating, quenching and transformation (RHQT) processed Nb_3Al is the prime candidate for the superconductor of TF coils because the critical current density (J_c) of the strand is as high as 1000 A/mm^2 at 16 T and 4.7 K [8]. In the design of the TF coils, the current density of the Nb_3Al wire is assumed to be 50% of J_c. This is based on the fact that the Nb_3Al wire shows a less degradation of J_c by strain. It is found that the magnetic field of the Nb_3Al TF coils reaches 16 T under the assumption although justification of the assumption is necessary throughout R&D of the Nb_3Al coil.

The estimation for high temperature superconductor (Bi-2212, $Bi_2Sr_2CaCu_2O_x$), which has higher current density than Nb_3Al, indicates that the attained magnetic field is about 17 T at most. This is because larger amount of structural material is necessary to reach higher field by Bi-2212. As a result, the composition of superconductor for Bi-2212 (1.8%) becomes smaller than that for Nb_3Al (2.2%) [9]. Taking the engineering feasibility of Nb_3Al in the near future as well, we regard Nb_3Al as the prime option of superconductor.

4. Torus configuration

In SlimCS, each sector has a sector-wide conducting plate slipped in between the replaceable (front) and permanent (back) blanket to attain a high normalized beta (β_N) of 4.3. This is because, for high β_N access, the sector-wide conducting plate is needed to be placed at a distance of $r_w/a \leqslant 1.3$, where a and r_w are the minor radius of plasma and the distance of the conducting plate from the plasma center, respectively. For SlimCS with $a = 2.1$ m, the plate must be located at 0.6 m or closer from the plasma outboard surface to sustain such a high β_N by stabilizing magneto-hydrodynamic (MHD) modes of plasma. It should be noted that blanket modules divided into hundreds of small pieces do not work as conducting plates. The reason is that the mutual inductance between the plasma and such a blanket module is too small to induce the eddy current on the module box enough to stabilize the MHD modes. Fig. 3(a) illustrates the torus configuration of SlimCS. The poloidal ring supporting replaceable blanket has separated box structure inside as shown in Fig. 3(b), being used as permanent blanket on the inside and shield on the outside. The thickness of replaceable and permanent blanket is 0.3 and 0.9 m, respectively. The conducting plate is 0.01 m-thick copper. Each fin of the conducting plates has an important function to cancel harmful components of eddy current loops by superposition with the eddy current passing on to the neighboring fin [10]. The conducting plate is irradiated up to 5–10 dpa during 2–3 year operation. Since the eddy current induced on the conducting plate is of the order of kA in

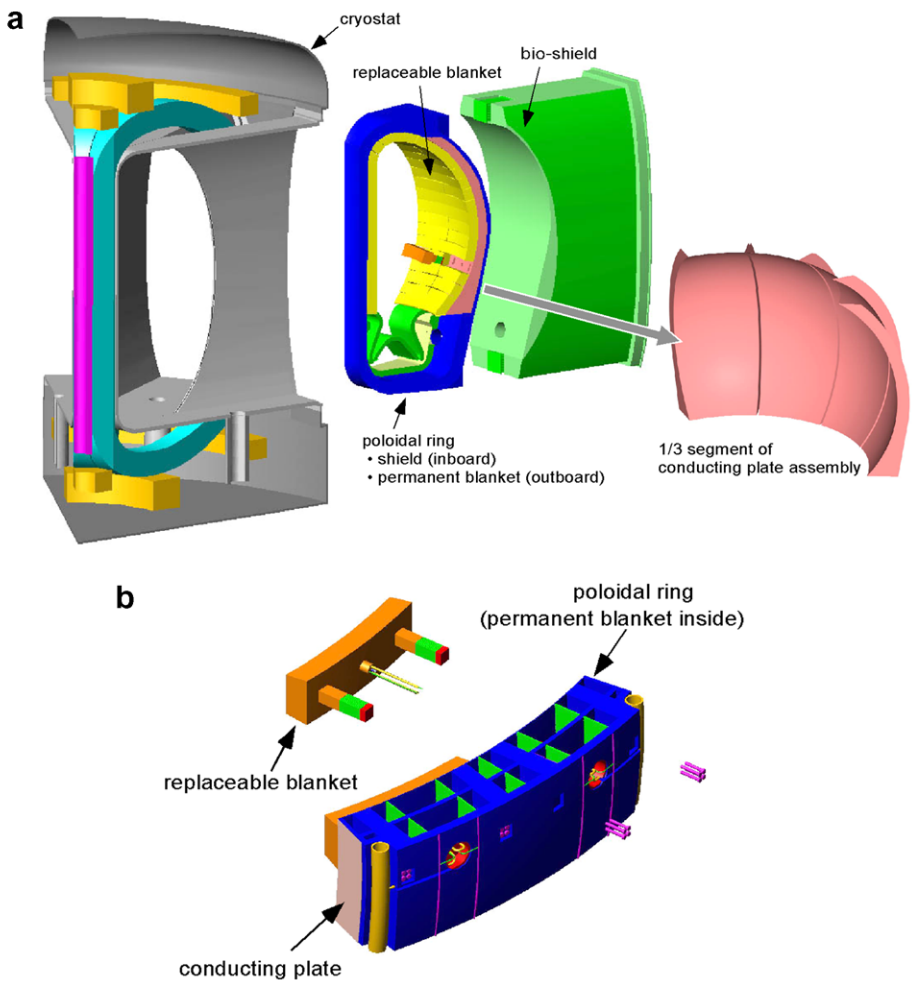

Fig. 3. Torus configuration of SlimCS: (a) setup of in-vessel components; (b) conceptual view of the attachment of replaceable blanket to the poloidal ring.

magnitude, the plate is requested to tightly contact with the permanent blanket not to buckle under the action of the resulting electromagnetic force.

One of the missions of DEMO is to prove a high availability of plant. In terms of this point, the marginal extension of the maintenance scheme for ITER (in-vessel maintenance with small blanket modules) will not be a solution to DEMO. SlimCS adopts sector transport hot cell maintenance so as to minimize the shutdown period for scheduled maintenance and to allow maintenance of such a complex torus configuration. Flexibility of replacing the original blanket with an advanced one is an additional advantage of sector transport maintenance.

5. Divertor

A difficulty on divertor in DEMO is that copper alloy used in ITER will be no more adaptable as major parts of the divertor due to radiation embrittlement. For this reason, a divertor concept with a RAFM (F82 H [11]) cooling tube and tungsten mono-blocks is considered [12]. In order to ensure the heat flux allowance of 10 MW/m^2, a thin-wall tube of 0.8–1.0 mm in thickness is adopted. However, the problem is that the inlet temperature of coolant should be as low as 200 °C to keep the working temperature of RAFM below 550 °C. This means that F82H will be used below DBTT (~350 °C). The inconsistency of divertor design may be resolved by reducing the heat flux allowance to about 5 MW/m^2 and increasing the inlet coolant temperature to about 300 °C. On the other hand, such a reduction of the allowance seems to make it more difficult to find a solution of power balance on core and divertor plasma. In addition to the difficulty in handling heat, divertor includes a lot of open issues on plasma wall interaction, which drives further DEMO design study.

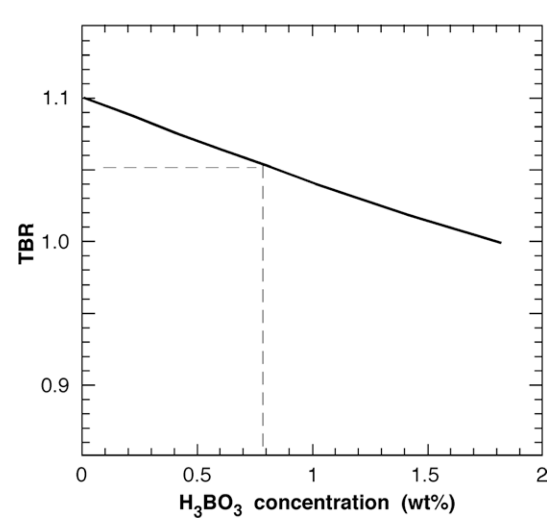

Fig. 4. TBR Control with borated water.

Table 2
Component breakdown in weight (tons).

	F82H	SS	Cryo. steel	Be	Li_2TiO_3	W	SC (CIC)	Insulat.	Subtotal
TF coils			5000				1100	80	6180
PF & CS			950				1600	40	2540
R-Blk	300			240	300	10			850
Divertor	450					230			680
P-Blk/Shld	2700			540	710				3950
Port/BioShld		900							900
Cryostat		4200							4200
Support		4300							4300
Subtotal	3450	9400	5950	780	1010	240	2700	120	23650

(R-Blk: replaceable blanket, P-Blk: permanent blanket, Shld: shield, BioShld: bio-shield, SS: austenitic stainless steel; SC(CIC): cable-in-conduit superconductor).

6. Blanket

Considering a solution to early realization of fusion power narrows down materials for the DEMO. The blanket materials are required to utilize the Japanese prime option of ITER-TBM. The most likely option for blanket structural material is F82H. In the previous DEMO design of JAERI (DEMO-2001), the combination of oxide dispersion strengthened (ODS) steel and supercritical water (25 MPa, 280–510 °C) was chosen [13]. In contrast, we chose technically matured F82H as structural material of SlimCS. On the other hand, since F82H is not compatible with supercritical water due to corrosion, the combination of F82H and subcritical water (23–25 MPa, 290–360 °C) was selected in SlimCS.

The outboard blanket consists of 0.3 m-thick replaceable and 0.9 m-thick permanent blanket. In contrast, only the 0.3 m-thick replaceable blanket is arranged on the inboard side. This is because the contribution of the inboard side to tritium breeding for such a low-A reactor is low compared with a conventional-A reactor with $A = 3$–4 [2]. The local TBR of 1.38 is expected using Be as neutron multiplier and Li_2TiO_3 or Li_4SiO_4 with 90%-enriched ^6Li as tritium breeder, resulting in the net actual TBR of 1.05. Suppose that the actual TBR exceeds a designed value due to uncertainty in cross sections. For example, when the actual TBR is 1.10 exceeding a designed value of 1.05, surplus production of tritium amounts to about 25 g/day, which will be extracted from the fuel cycle system and stored in the on-site fuel storage. Since the surplus production of tritium reaches 9 kg for one-year operation, a "in situ" TBR control method may be required to avoid excessive production if the tritium is not used as the starting stock for the next generation of plants. Borated-water is promising for the purpose in that water borated with 0.8 wt% of H_3BO_3 reduces the TBR by 0.05, corresponding to a reduction of the TBR from 1.10 to 1.05 as shown in Fig. 4. Such a controllability of TBR is a merit of water-cooled blanket.

From the point of view of waste management, it is very important to dispose of the waste by shallow land burial because the only existing repository in Japan is located in Rokkasho where low level waste is disposed of in concrete pit storage in shallow land. Critical nuclei of F82H regarding waste classification are ^{14}C originating from N, ^{94}Nb from Nb, ^{93}Mo and ^{99}Tc from Mo, and ^{192}Ir from W contained in F82H. For the shallow land burial, we set the target of detrimental elements other than W (2 wt.%) in F82H: N < 20 ppm, Nb < 1 ppm, and Mo 100 ppm. When the condition is satisfied, most part of blanket will have qualification of shallow land burial 50 years after the decommissioning. There is a viewpoint that much more nitrogen should be contained for toughness and reliability in mass production. The requirement can be acceptable by enriching ^{15}N (natural abundance 0.37%) without increasing the production of ^{14}C. When ^{15}N is enriched to 95%, the nitrogen content of 200 ppm in F82H can be allowable in terms of waste management [14]. Incidentally, ^{15}N is most likely to be separated with thermal diffusion isotope separation technique, for instance.

In contrast with the reduced waste strategy, we may take another strategy focused on a cost reduction and reality. The idea is to use already existing blast furnaces, which have been contaminated with various impurities, for the production of F82H in lieu of the abandonment of chemical composition and impurity control for F82H. This seems to be realistic compared with constructing a furnace for exclusive use in DEMO. This is an issue of choice which to take reality or public acceptance.

7. Component breakdown

The provisional component breakdown is listed in Table 2. SlimCS requires about 18800 tons of steel. For public acceptance regarding waste, the shield should be made of RAFM (F82H) rather than austenitic stainless steel (SS). As a result, F82H amounts to as high as 3400 tons. As to superconductor (SC), the amount is estimated in the cable-in-conduit (CIC) conductor basis composed of superconductor strand, niobium, copper, SS conduit and insulator. The amount of tungsten of blanket (9 tons) corresponds to the 0.5 mm coating on the first wall. It should be stressed that the total reactor weight of 23650 tons is as low as a commercial reactor concept assuming advanced physics parameters such as CREST and ARIES-RS [2]. In particular, TF coils of 6180 tons are remarkably light compared of that of SSTR [15] of 11200 tons. Such a reduction in TF coils weight of SlimCS is attributed to low aspect ratio.

8. Summary

The philosophy of SlimCS design is to reduce the construction cost. At the same time, we attempted to avoid technology challenges in the materials selection as possible. The choice of water-cooled solid breeder and wide usage of SS is consequences of the philosophy. On the other hand, we are aware of a problem that considerations on environmental aspects such as waste management have been left behind in the present design. This will be a challenging issue that we should deal with in the future. As suggested by the previous study [16], a dramatic reduction of radioactive waste seems difficult as far as the present materials selection of SlimCS is concerned. Introduction of an advanced blanket with environmentally aware, for example, the combination of liquid breeder and SiC/SiC, in the later phase of DEMO operation will be a possible idea to stress environmental awareness in fusion energy development.

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