Fusion Engineering and Design 88 (2013) 488–491

Contents lists available at ScienceDirect

Fusion Engineering and Design

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

A preliminary conceptual design study for Korean fusion DEMO reactor

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HIGHLIGHTS

- Perform a preliminary conceptual study for a steady-state Korean DEMO reactor.
- Present design guidelines and requirements of Korean DEMO reactor.
- Present a preliminary design of TF (toroidal field) and CS (central solenoid) magnet.
- Present a preliminary result of the radial build scheme of Korean DEMO reactor.

ARTICLE INFO

Article history:
Received 13 September 2012
Received in revised form 8 February 2013
Accepted 20 February 2013
Available online 27 March 2013

Keywords:
DEMO
Reactor design
16-T Nb_3Sn TF magnets
Two-stage development plan

ABSTRACT

As the ITER is being constructed, there is a growing anticipation for an earlier realization of fusion energy, so called fast-track approach. Korean strategy for fusion energy can be regarded as a fast-track approach and one special concept discussed in this paper is a two-stage development plan. At first, a steady-state Korean DEMO Reactor (K-DEMO) is designed not only to demonstrate a net electricity generation and a self-sustained tritium cycle, but also to be used as a component test facility. Then, at its second stage, a major upgrade is carried out by replacing in-vessel components in order to show a net electric generation on the order of 300 MWe and the competitiveness in cost of electricity (COE). The major radius is designed to be just below 6.5 m, considering practical engineering feasibilities. By using high performance Nb_3Sn-based superconducting cable currently available, high magnetic field at the plasma center above 8 T can be achieved. A design concept for TF magnets and radial builds for the K-DEMO considering a vertical maintenance scheme, are presented together with preliminary design parameters.

1. Introduction

The ITER project mission is to provide the scientific and engineering feasibility of fusion energy by achieving extended D-T burning plasma of several hundred seconds; now in its construction phase [1]. But, at the same time, the next generation fusion demonstration reactor (DEMO) is being actively discussed worldwide to further accelerate the commercialization of fusion energy. Based on the so called fast track approach, EU, Japan, and USA have revised their roadmaps so that an economically viable demonstration fusion power plant can be operated during the 2040s [2]. A similar strategy is adopted in Korea as was announced in a National Fusion Roadmap released in 2005 [3]. Furthermore, Fusion Energy Development Promotion Law (FEDPL), the first legal act in the world for fusion energy development, was enacted in 2007 to promote a long-term cooperative fusion research and development among participating industries, universities and research institutes.

In order to meet this target schedule, a realistic and flexible conceptual study, which can incorporate the operation phase, needs to be initiated; as was discussed by Hiwatari et al. in their work on Demo-CREST [4]. Furthermore, for an immediate realization, a practical consideration of engineering feasibilities is a mandatory prerequisite. This paper reports on our preliminary work on a conceptual study for K-DEMO, providing design guidelines, preliminary design parameters, and radial build scheme based on practical design of toroidal field (TF) and central solenoid (CS) magnets. The radial build scheme also incorporates a vertical maintenance scenario [5]. The plasma exhaust system is not discussed here because of the limitation in current technologies for the divertor system

2. Design guidelines for K-DEMO

A conceptual design study for K-DEMO has been initiated and the overall time plan, based on the Korean FEDPL, can be

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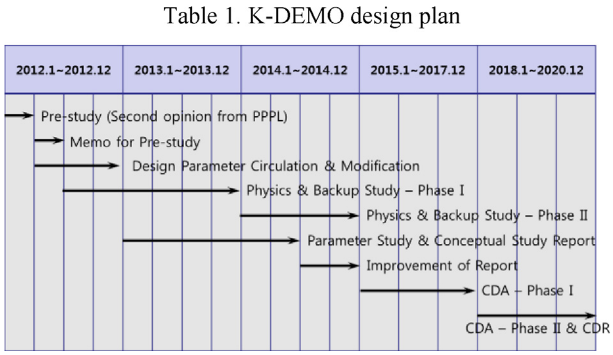
http://dx.doi.org/10.1016/j.fusengdes.2013.02.123

Table 1
K-DEMO design plan.



summarized in Table 1. At the moment, general requirements are discussed, in particular on a feasibility of two-stage development plan. In its first stage, K-DEMO is not targeted to demonstrate the competitiveness in COE. In other words, K-DEMO, in this stage, cannot be regarded as a final DEMO but rather a test facility for a commercial reactor. All plant core and subsystems shall be representative of those in the final plant. At first, several ports will be designated for diagnostics relevant to burning plasma operations and at least one port shall be used for component testing including blanket test. Still, electricity generation ($Q_{eng} > 1$) and a self sustainable tritium cycle (tritium breeding ratio, TBR > 1.05) needs to be demonstrated.

Economic feasibility requires a demonstration of the competitiveness in COE. This will be achieved during the second stage development and operation, which involves a major upgrade of in-vessel components including blankets, divertor system and affected interfacing systems and services. The second phase of operation is expected to generate net electricity on the order of >300 MWe with a planned plant availability over 70%. Remote-handling will definitely be incorporated from the first stage operation period.

To incorporate the plasma physics accumulated during the ITER operation and to utilize the experience acquired from the ITER construction, the size of K-DEMO shall be relevant to that of ITER. Furthermore, we expect the extrapolation of technologies from K-DEMO to a commercial fusion power plant should be straight forward. Practically, all technology to be used in a commercial reactor needs be demonstrated by K-DEMO. Also, extrapolation of performance parameters between K-DEMO and a commercial plant should be minimized. To fix the size of K-DEMO based on plausible engineering limits, preliminary TF magnet designs and radial build schemes have been studied.

3. Engineering considerations on the physical size of K-DEMO

Higher toroidal field is usually preferable and could offer engineering margins for uncertainties that might be encountered during the transfer of the ITER plasma physics. Recently, high performance Nb_3Sn strands with critical current densities over 2600 A/mm^2, more than twice than that of current ITER-type strands, was developed [6]. The ac-performance of these strands is not yet confirmed but can be considered as the most prominent candidate superconducting strands for TF coils. The ITER-type Nb_3Sn strands could be utilized for the central solenoid (CS).

Two options for the ITER relevant size of TF magnets are shown in Fig. 1, which correspond to the major radius of 6.0 and 6.5 m, respectively, as listed in Table 2. Both magnets are designed to produce the toroidal field of ~7.72 T at the plasma center and the peak field of ~16 T. An improved feature of the current magnet designs is that for both the option I and II, the magnets are wound with two different cable-in-conduit conductors (CICCs) in order to reduce the fabrication cost. The inner and outer magnets are connected serially. Possible joint schemes are being studied, together with the overall magnet structure analysis. The nominal currents for option I and II are about 78.7 kA and 85.2 kA, respectively. It is expected that temperature margins for option I and II are around 0.9 K and 1.0 K, respectively. To accommodate enough

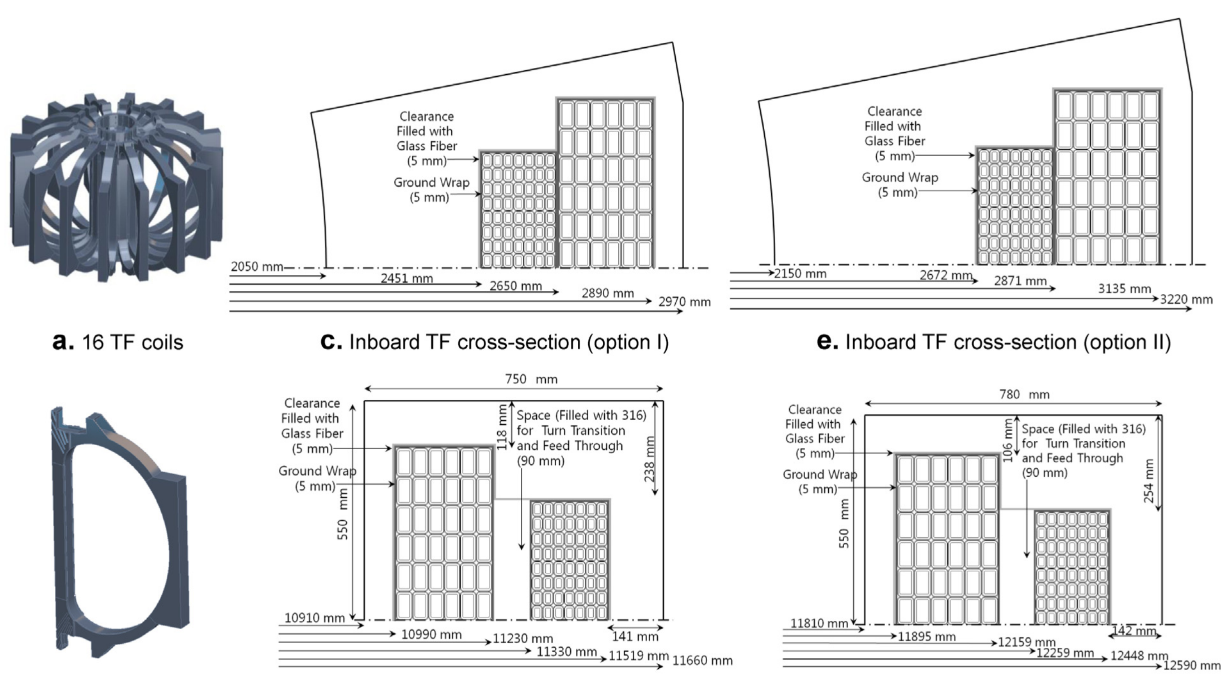


Fig. 1. A preliminary 3D modeling for TF coils (a, b) and detailed inboard and outboard TF magnet cross-sections for both option I and II (c–f).

Table 2
Parameters and operational capabilities of K-DEMO.

Parameter	Option 1	Option 2
Major radius, R_0 (m)	6.0	6.5
Minor radius, a (m)	1.8	2.0
Aspect ratio, A	3.33	3.25
Toroidal field on axis, B_T (T)	7.72	7.72
Plasma current, I_P (MA)	11	12.5
Elongation, κ	1.8	1.8
Triangularity, δ	0.4	0.4
Normalized beta, β_N (%)	4.2	4.2
Safety factor, q_{95}	3.5–6.0	3.5–6.0
Energy confinement time, τ_E (s)	2.27	2.36
Bootstrap current fraction, f_{bs}	0.6	0.6
Averaged electron temperature (keV)	19	19
Averaged electron density ($10^{20}/m^3$)	1.08	1.12
Greenwald limit, n_{GW} ($10^{20}/m^3$)	1.08	0.99
n/n_{GW}	1.0	1.13
Total fusion power (MW)	1708	2400
Q-value	24.4	30
Total H&CD power (MW)	70	80
L–H mode transition power (MW)	98	113
Average wall loading (MW/m^2)	2.0	2.34
Z_{eff}	1.4	1.4

space for a vertical maintenance, 16 TF coils are considered, at the moment.

The physical dimensions for the assumed CICCs, considering plausible cabling schemes, are presented in Fig. 2. The CICC structures for the outer TF coils are the same for both options. The cabling scheme considered at the moment is (3SC) × 4 × 4 × (5 + cooling spiral) (240 SC strands). 5 sub-cables are helically wound together without a central cooling spiral and the void fraction is ∼30%. About 600 m long CICC will be wound forming double pancakes. The cabling scheme of the CICC for the option I inner TF coils is (3SC) × 4 × 5 × 6 × (5 + cooling spiral) (1800 SC strands) and the void fraction is ∼29%. The unit length of CICC for quadruple pancakes is around 920 m. The cabling scheme of the option II inner TF CICC is (3SC) × 4 × 6 × 6 × (5 + cooling spiral) (2160 SC strands) with the void fraction of ∼30% and the unit length, around 990 m. The cabling scheme for CS coils CICC also can be similarly designed, (3SC) × 3 × 4 × 4 × 5 (864 SC strands) with the void fraction of ∼36%. The unit length of CICC for sextuple pancakes is around 950 m. Because of the incompleteness in vertical maintenance system and advanced divertor concepts, PF coil system design is not set yet. Also a detailed structural analysis is not completed yet.

A preliminary concept for blankets can be found in Fig. 3. A water-cooled solid breeding blanket using lithium orthosilicate

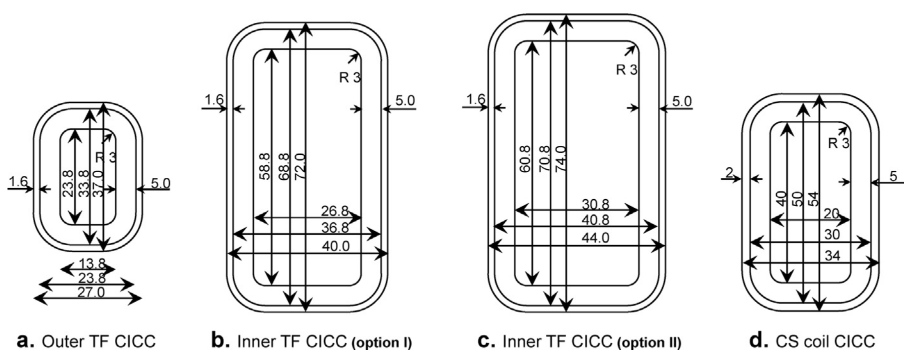


Fig. 2. Physical dimensions of the assumed CICCs (unit: mm).

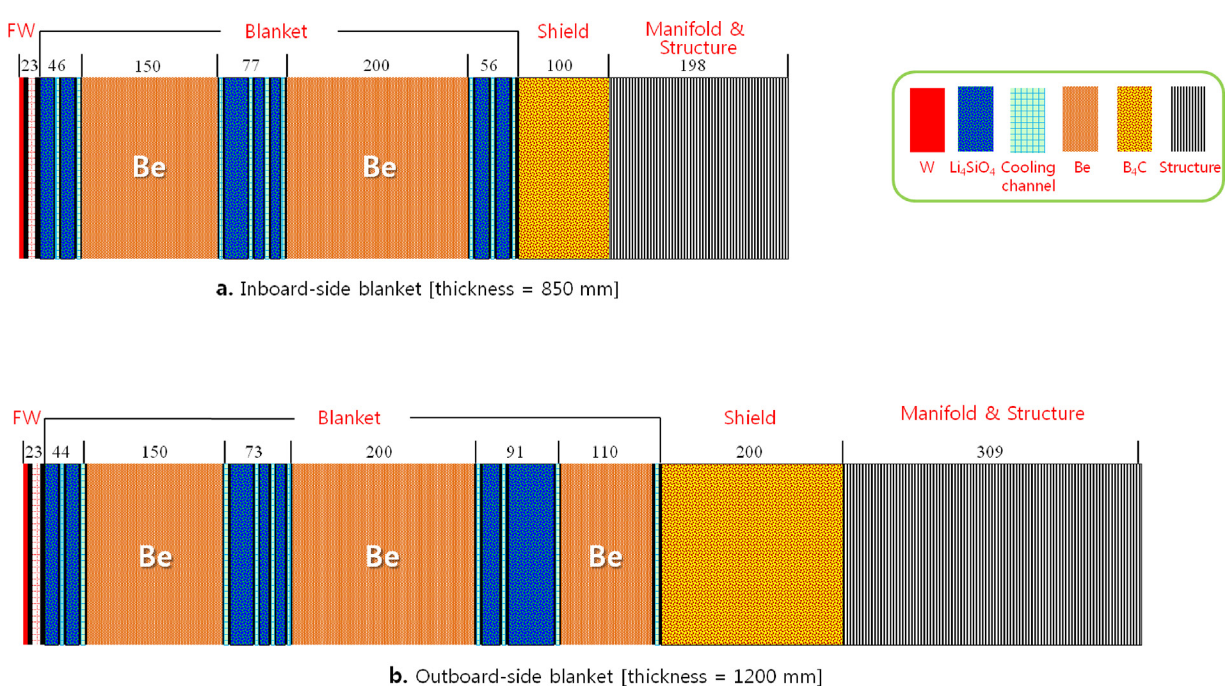
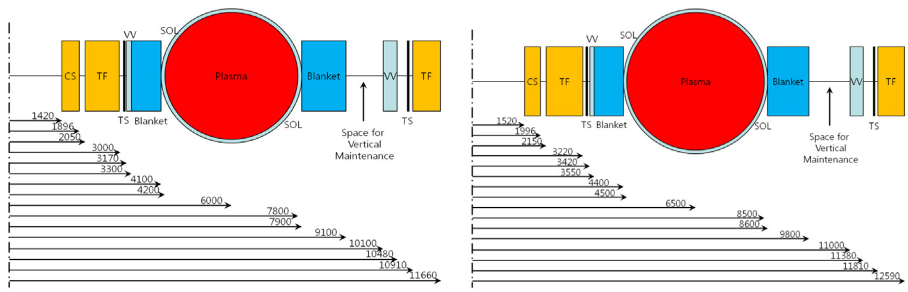


Fig. 3. Assumption for the thickness of inboard and outboard-side blankets (unit: mm).



a. Radial build option I **b.** Radial build option II

Fig. 4. Radial build of K-DEMO option I and II (unit: mm).

Table 3
Power balance of K-DEMO.

Power balance	Option 1	Option 2
Fusion power, P_{fus}	1708 MW	2400 MW
Heating & current drive power, P_{heat}	70 MW	80 MW
Total Heat, $(0.8f_m{}^{a} + 0.2)\, P_{fus} + P_{heat}$	2325 MW	3248 MW
Q (P_{fus}/P_{heat})	24.4	30.0
Thermodynamic efficiency, η_{th}	0.35	0.35
Gross electric power	598 MWe	840 MWe
Recirculating fraction	0.85	0.65
Recirc. electric power	508 MWe	546 MWe
Net electric power	90 MWe	294 MWe

[a] Energy multiplication factor of blanket f_m is 1.4.

(Li_4SiO_4) and beryllium (Be) as a ceramic breeder and a neutron multiplier, respectively, is being considered. The first wall (FW) will be made of 5 mm-thick tungsten plate. The pebble layers of beryllium and breeder are arranged in parallel to the first wall along the radial direction as shown in Fig. 3. In-between layers, cooling plates are located for heat removal. Tritium breeding ratios (TBR), calculated by a Monte Carlo simulation code (MCNPX version 2.6.0) [7] for the inboard and outboard blankets at 60% Li-6 enrichment, are about 1.17 and 1.18, respectively. More detailed analysis can be found in our recent work [8].

With these engineering design concepts, possible radial builds can be constructed, as shown in Fig. 4. The K-DEMO reactor will be optimized further to meet the constraints of plasma physics and engineering simultaneously. The tentative parameters and operational capabilities of K-DEMO reactor are presented in Table 2. The elongation factor of K-DEMO is around 1.8 with a single-null configuration. System analysis codes are being used for the analysis of plasma and engineering parameters with feasible boundary values, such as steady-state maximum neutron wall loading, maximum peak heat flux in the divertor, and thermal and other efficiencies in the power conversion cycle. For example, the power balance of K-DEMO is presented in Table 3. Since the advanced divertor concept is being considered for the K-DEMO divertor system, it is too early to discuss about the power exhaust system under the assumption of current technologies. Detailed analysis results can be found in our recent work [9].

4. Summary

In summary, a preliminary conceptual design study for a steady-state Korean DEMO Reactor (K-DEMO) is presented. The main focus has been to address practical engineering considerations to meet the tight schedule for the fast-track approach. A two-stage development plan was presented, which includes testing of components needed for a commercial reactor. In addition to component testing, electricity generation and a self-sustainable tritium cycle will be demonstrated. In the second stage of operation, a major upgrade will be carried out with the goal of providing competitiveness in COE and net electric generation, over 300 MWe. Other considerations include near-future engineering feasibilities. The lesson-learned during the ITER project should be easily transferred, which invokes keeping the major and minor radii of K-DEMO around 6 m and 1.8 m, respectively. Utilizing the high current density Nb_3Sn strand, allows the magnetic field at the plasma center to reach ~8 T, providing good physics operating margin. Preliminary blanket design concept has been carried out to develop a reasonably sized K-DEMO device that incorporates a vertical maintenance scheme. The preliminary design parameters and operational capabilities of K-DEMO reactor have also been presented.

Acknowledgments

This work supported by the Ministry of Education, Science and Technology of Republic of Korea and also partially supported by the U.S. DOE under Contract No. DE-AC02-09CH11466 with Princeton University. Authors also want to say thanks to Antonio della Corte and his colleagues for the effort in the test of CICC jacketing and discussions.

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