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A study regarding the optimal radial build of a low-aspect-ratio tokamak fusion system

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ABSTRACT

System parameters and the optimal radial build of a low-aspect-ratio (LAR) tokamak fusion system were found through the coupled analysis of a tokamak system and neutron transport. In a configuration with only an outboard breeding blanket, the minimum major radius to produce a given fusion power was determined by the shielding requirements and the magnetic field at the toroidal field (TF) coil. With a confinement enhancement factor $H = 1.4$, $Q > 10$ was possible for fusion power greater than 800 MW with an aspect ratio of $A = 1.5$; however, $Q > 10$ was possible for fusion power greater than 1800 MW with an aspect ratio of $A = 2.0$. The outboard radial build was determined by the tritium breeding and shielding requirements. The tritium breeding capability of blanket concepts proposed for testing in the International Thermonuclear Experimental Reactor (ITER) was evaluated by varying the outboard blanket thickness and the degree of lithium-6 (Li-6) enrichment. Cases with a smaller aspect ratio exhibited better performance since the number of fusion neutrons that contributed to tritium breeding were larger than the case with a larger aspect ratio. Among the blanket concepts, a helium (He)-cooled solid breeder (HCSB) concept showed the best tritium breeding capability and thus allowed for a smaller system size.

1. Introduction

LAR tokamak fusion systems have been studied [1–3] as an alternative path toward fusion reactors since they allow for both a large elongated plasma shape due to enhanced magnetohydrodynamic (MHD) stability and a high plasma beta, leading to a high-performance compact reactor. The need for concept development of a fusion demonstration (DEMO) reactor has grown as construction of the ITER has progressed, and successful operation of the ITER is foreseen. From a moderate extrapolation of the physics and technology adopted in the design of ITER, a study on the feasibility of LAR tokamak fusion systems is worthwhile.

For optimal design of the tokamak fusion system, radial builds of the system components must be self-consistently determined by accounting for plasma physics and engineering constraints, which inter-relate various system components [4]. For a given plasma performance, a small-sized system is desirable from an economic perspective; thus, a large toroidal magnetic field at a magnetic axis (B_T) with a small major radius is preferable since the fusion power, P_{fus}, varies as $P_{fus} \propto \beta^2 \ B_T^4$, where β is the plasma pressure. To increase B_T, the distance between the TF coil and the plasma must be short, while sufficient thickness is required to shield the superconducting TF coil. Thus, radial builds of components between plasma and the TF coil must be determined by accounting for the impact of 14 MeV neutrons produced during the fusion reaction.

The design of a blanket and shield played a key role toward determining the size of the tokamak fusion system and should be optimized with regard to the neutronic response. The blanket should produce enough tritium for tritium self-sufficiency, and the shield should provide sufficient protection for the superconducting TF coil against fast neutron fluence toward the superconductor, nuclear heating within the winding pack, radiation damage to the stabilizer, and radiation dose absorption by the insulator. To evaluate the tritium breeding capability and shielding characteristics, the neutron flux must be known, which depends on the dimensions, material, and configuration of the system components. Neutronic analysis must therefore be a part of the systems analysis for a self-consistent determination of the radial build of a tokamak fusion system; thus, we coupled the systems analysis to a one-dimensional neutron transport calculation. Neutronic effects were self-consistently incorporated in the systems analysis together with plasma physics and engineering constraints.

With the coupled system analysis, we studied the optimal radial build of a LAR tokamak fusion system. For the LAR tokamak, tritium self-sufficiency was shown [3] to be satisfied with only an outboard blanket. The maximum B_T and minimum major radius R_0 were thus determined by the shielding and maximum magnetic field requirements

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at the TF coil. With the minimum R_O determined this way, we investigated the dependence of the minimum R_O, the auxiliary heating power, and the divertor power handling index on the fusion power to evaluate the feasibility of the LAR tokamak fusion system.

Outboard radial builds for the LAR tokamak fusion system were determined by the tritium breeding and shielding requirements. Blanket concepts that were proposed for testing in the ITER as test blanket modules (TBM) [5] were evaluated for the feasibility of the relevant DEMO blanket by varying the outboard blanket thickness and the degree of Li-6 enrichment. The dependence of the tritium breeding capability on the aspect ratio was also investigated. The TBMs included a European helium-cooled solid breeder (HCSB) [6], a European helium-cooled lithium lead (HCLL) [6], a Chinese helium-cooled solid breeder (HCSB) [7], a helium-cooled ceramic reflector (HCCR) [8], a Japanese water–cooled ceramic breeder (WCCB) [9], and an India lead lithium ceramic breeder (LLCB) [10]. We simplified the solider breeder concepts by assuming the same breeding and structural materials to compare their breeding capability.

Analysis tools and models are explained in Section 2. The physical and engineering constraints are presented in Section 3, and the results of the coupled systems analysis for the LAR tokamak fusion system are given in Section 4. Section 5 provides a summary of this work.

2. Analysis tools and models

A one-dimensional radiation transport code, ANISN [11], was used for neutronic calculations and was embedded as a subroutine in the tokamak system code. ANISN is a multi-group 1-D discrete ordinate multi-group transport code with anisotropic scattering, solving the Boltzmann transport equation for neutrons and gamma rays in slabs, spheres, or cylindrical geometries. With regard to cross-section libraries and activity tables, MATXS-format libraries [12] based on JENDL-3.3 generated via NJOY99 code were used. These libraries were further processed to produce 42 neutron group transport tables for the ANISN through TRANSX 2.15 [13]. The tritium breeding ratio (TBR) was the ratio between the amount of tritium generated in a blanket and the quantity of tritium consumed in a fusion reaction and was calculated using the lithium cross-sections from the ENDF/B-VII.1 library.

The systems analyses determined plasma and system parameters that satisfied the plasma physics and engineering constraints, meshes were subsequently prepared from the radial builds of the system, and neutron transport calculations were performed for the blanket and shield system. Calculated neutron flux and activities were used to calculate the TBR and shielding parameters. This procedure continued until all the plasma physics and engineering constraints were simultaneously satisfied. A schematic diagram of the calculation procedure is shown in Fig. 1.

We considered the LAR tokamak configuration with only an outboard blanket; its radial build, material, and volume fraction are summarized in Table 1. Neutronic calculations were performed for a cylindrical geometry including both the inboard and outboard regions.

For a super-conducting (SC) material of the TF coil, Nb_3Sn was assumed. The number of TF coils was set to 16. The vacuum vessel was assumed to be 0.1 m thick and was composed of borated steel, which was the same material adopted in the ITER vacuum vessel. A lead (Pb) layer was added to the inboard shield to effectively improve the TBR since Pb possesses a larger (n,2n) cross-section for high energy neutrons. Neutron reflection and multiplication effects by Pb had an impact

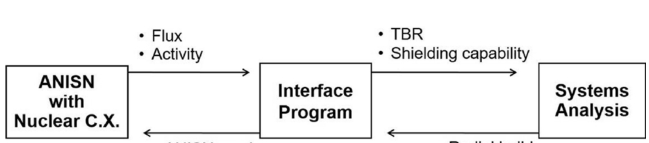

Fig. 1. A schematic diagram of the calculation procedure.

Table 1
Component and material composition of the LAR tokamak fusion system.

Component	Materials (Volume%)
Toroidal field coil	SUS316, L. He, Nb_3Sn, Cu, Epoxy
Vacuum vessel	Borated steel (60), H2O (40)
Shield	WC (80), H_2O (20)
First wall	FMS (60), H_2O (40)
Plasma	D, T
First wall	FMS (60), H_2O (40)
Breeding blanket	He (10), Breeder (80), FMS (10)
High temp. shield	WC (60), H_2O (40)
Low temp. shield	WC (80), H_2O (20)
Vacuum vessel	Borated steel (60), H_2O (40)
Toroidal field coil	Nb_3Sn, Cu, Epoxy, SUS316, L. He

on the TBR and outboard blanket thickness. The shield material was tungsten carbide (WC). The first wall was made from low-activation ferritic-martensite steel (FMS), and the thickness of the scrape off layer (SOL) was set to 0.1 m. With regard to the outboard blanket, concepts based on the ITER TBMs were compared. Blanket materials were assumed to be homogeneously mixed in this study.

3. Physics and engineering constraints

The plasma physics parameters could be determined by the choice of aspect ratio, A, and the plasma performance was limited by the beta limit, plasma current limit, and plasma density limit. Appropriate models for the plasma composition, bootstrap current fraction, non-inductive current drive, divertor heat load, etc. were also necessary to calculate the plasma performance. Although these dimensionless models are unable to reflect plasma profile effects, they provide satisfactory results in scoping studies of the tokamak fusion system concept. For the physics models, we refer to Ref. [4].

The maximum elongation κ was assumed to depend on the aspect ratio as $\kappa_{\max} = 2.4 + 65\exp(-A/0.376)$ [14]; the dependence of the maximum normalized beta β_N on the aspect ratio and elongation was assumed as follows [15]:

$$\beta_{N,\max} = \frac{-0.7748 + 1.2869\kappa - 0.2921\kappa^2 + 0.0197\kappa^3}{\tanh[(1.8524 + 0.2319\kappa)/A^{0.6163}]A^{0.5523}/10}$$

The plasma current was calculated according to

$$I_p = \frac{5a^2 B_T}{R_0 q_a}\frac{1+\kappa^2}{2}$$

where R_o is the major radius (m), a is the minor radius (m), and q_a is the safety factor q at the edge.

The electron density limit was given by the Greenwald limit:

$$n_G = \frac{I_p}{\pi a^2}$$

Density and temperature profiles were assumed to be parabolic ~ $(1 - r^2/a^2)^{\alpha}$ with the exponent of the density and temperature profiles, α_n, α_T.

We assumed a 50/50 mixture of D/T fuel, and neon was assumed to be a radiating impurity. The He ash fraction f_{He}, impurity ion charge Z_{imp}, and impurity ion fraction f_{imp}, were 0.02, 10, and 0.01, respectively. The ion fraction f_i and Z_{eff} were calculated as follows:

$$f_i = \frac{1 - 2f_{He} - f_{imp}Z_{imp}}{Z_i}$$

$$Z_{eff} = f_i + 4f_{He} + Z_{imp}^2 f_{imp}$$

In this work, we considered a case in which the plasma current was fully driven by a combination of the non-inductive current drive and the bootstrap current. The bootstrap current fraction f_{bs} was calculated using the formulation developed by Pomphrey in the arbitrary aspect