CONCEPTUAL DESIGN STUDY FOR DOWNSIZING OF FUSION DEMO REACTOR

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This paper reports a conceptual design study for a downsized fusion DEMO reactor. It is very important to conduct power generation demonstrations as early as possible for early social implementation of fusion energy. For a DEMO reactor with a mission of power generation demonstration, the reactor size, which is larger than the ITER currently under construction, entails a longer construction period and development risk due to its larger size. Based on the system code analysis, this conceptual design study investigated a reactor concept that can both demonstrate power generation and tritium self-breeding in an ITER-size DEMO reactor. By improving the in-vessel components step by step in a single device, the DEMO rector concept was presented that could achieve a net electric power of more than 0 with ITER-like parameters in Phase I, demonstrate comprehensive tritium breeding for self-realization with JA DEMO-like parameters in Phase II, and achieve the net electric power of 100 MW-class with JT-60SA-like parameters in Phase III. In addition, by evaluating the impact of key components for miniaturization in the DEMO reactor, such as superconducting magnets and blanket, the R&D items that are important for miniaturization of the reactor were clarified.

1. INTRODUCTION

The conceptual design of the Japanese demonstration (DEMO) reactor is being carried out by the Joint Special Design Team for fusion DEMO to establish the Japanese DEMO concept, named “JA DEMO” [1]. The following values are set for the main design parameters of JA DEMO to meet the requirements of the DEMO reactor [2]. The plasma major radius (R_p) is 8.5 m, fusion output (P_{fus}) is 1.5-2 GW, the net electric power (P_{net}) is 0.2-0.3 GW, and the magnetic field on the plasma axis (B_t) is 6 T. On the other hand, from the viewpoint of early power generation demonstration, the larger reactor in the conventional JA DEMO concept leads to a more extended construction period and higher development risk. Therefore, based on ITER's experience in manufacturing toroidal field coils and its ability to foresee burning plasma (high energy multiplication), for the early power generation demonstration, a conceptual design study was carried out on a DEMO reactor downsized from JA DEMO (R_p = 8.5 m) to the ITER size (R_p = 6.2 m), with a step-by-step approach to demonstrate early power generation and tritium breeding, and to obtain the net electric power of 100 MW-class.

2. MAIN CONCEPT

Reactor dimensions (plasma main radius and toroidal field (TF) coil dimensions) were studied using that of ITER as the initial values. The “generation demonstration” was defined as the excess of the generating end output over the on-site power, i.e., a positive net electric power. By improving the in-vessel components step by step in a single device, the concept of a 100 MW class net electrical output was explored. The furnace parameters were studied using the system code TPC. As a phased approach, the following three phases were envisioned. Table 1 shows the main parameters of the ITER-size DEMO reactor for each operational phase.

Phase I is the system integration phase, and the goal is to demonstrate power generation (net electric power P_{net} > 0) in this initial phase. In this phase, under the ITER baseline scenario (Q = 10), P_{net} ~ 5 MWe is obtained at a fusion power P_{fus} of about 500 MW by installing a power generation blanket of the same size as the ITER shielding blanket. Plasma heating is only electron cyclotron heating (ECH), with a pulsed operation of about 400 seconds.

Phase II is the functional test phase of the breeding blanket, which aims to demonstrate comprehensive tritium breeding for self-realization in addition to the power generation demonstration in Phase I. In this second phase, to obtain a tritium breeding ratio (TBR) of more than 1, a thicker breeding region in the radial direction than the ITER shielding blanket is required, resulting in a decrease in the plasma volume (decrease in plasma minor radius). To compensate for the reduced plasma volume, a conventional JA DEMO scenario (normalized beta β_N = 3.4, normalized density n_e/n_{GW} = 1.2) is assumed compared to the ITER baseline scenario (Q = 10). As a result, P_{fus} ~ 500 MW and P_{net} ~ 10 MWe are expected to be obtained at TBR~1.05. Plasma heating is a combination of ECH and neutral beam injection heating (NBI), and the plasma is operated in pulses of several hours.

Phase III is the extended operation phase and aims to achieve P_{net} ~ 100 MWe through steady-state operation demonstration, power generation demonstration, and tritium breeding. In this phase, the JT-60SA scenario (β_N = 4.3) is assumed to obtain even larger fusion power than in Phase II. As a result, steady-state operation of P_{fus} ~ 800 MW and P_{net} ~ 80 MWe is expected while maintaining TBR ~ 1.05. Further increase of P_{net} is expected if the heating and current drive system, which has been underway in parallel with the DEMO reactor operation, can be made more efficient.

Tab. 1. Main parameters of the ITER-size DEMO reactor for each operational phase.

	Phase I	Phase II	Phase III
R_p / a_p [m]	6.2 / 2.0	6.2 / 1.65	6.2 / 1.65
κ_{95}	1.7	1.7	1.7
q_{95}	3.0	4.0	3.7
I_p [MA]	15.0	7.4	8.0
B_t [T]	5.3	5.3	5.3
P_{fus} [MW]	492	510	820
Q	10	10	14
P_{net} [MWe]	7.3	9.3	82.5
β_N	1.8	3.4	4.3
HH_{98y2}	0.95	1.41	1.50
n_e/n_{GW}	0.85	1.19	1.20

3. SUPERCONDUCTING MAGNETS AND BLANKET

As initial parameters, the TF coil was assumed to have the same dimensions and performance as the ITER-TF coil (superconducting wire: Nb_3Sn, conductor current: 68 kA, design stress: 667 MPa (yield stress: 1000 MPa)). On the other hand, assuming the use of high-strength cryogenic steel with a yield stress of 1200 MPa [3] and 83 kA conductors, which have been developed for JA DEMO, the magnetic field on the plasma axis B_t will increase by 0.35 T and P_{net} by about 15 MWe. These are very important to ensure net electrical output in Phases I to III, and since TF coils are difficult to upgrade step by step, it is essential to complete these R&Ds and achieve higher magnetic fields as soon as possible.

In the power generation blanket (0.45 m thick) in Phase I, tritium can be produced by loading materials for fuel production (breeding and multiplying materials). In the case of a breeding area of 0.25 m and a shielding area of 0.2 m, the TBR is 0.84, which means that fuel production can be tested even in the power generation demonstration phase, although the TBR will not be higher than 1. Since the radial thickness of the breeding blanket and shielding directly affects the plasma volume, i.e., the fusion power, the thinning of the breeding blanket and shielding while maintaining the tritium breeding and shielding performance (high performance) is an essential R&D item for this reactor concept and commercial reactors.

4. CONCLUSION

In a DEMO reactor downsized to ITER size, the goal is to demonstrate power generation with positive net electric power and fuel self-sufficiency. This is followed by a step-up of in-vessel components and core plasma performance based on various R&D results in addition to ITER and JT-60SA to achieve a fusion reactor with a single device. The concept of a fusion energy reactor that can generate 100 MW-class net electric power was established by stepping up the performance of the in-vessel components and core plasma based on the results of various R&D activities.

The construction of the reactor will start by conducting the R&D necessary for the early power generation demonstration in advance. The R&D for upgrading the in-vessel components (blanket and shielding) to be replaced periodically and for regularization and higher efficiency of the heating devices (NBI and ECH) will be conducted in parallel with the construction, which may accelerate the power generation demonstration period.

REFERENCES

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