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CONCEPTUAL DESIGN STUDY FOR DOWNSIZING OF FUSION DEMO REACTOR

H. UTOH
National Institutes for Quantum Science and Technology (QST), Rokkasho, Aomori, Japan
Email: uto.hiroyasu@qst.go.jp

Y. SAKAMOTO, S. SUGIYAMA, Y. SOMEYA, and the Joint Special Design Team for Fusion DEMO
National Institutes for Quantum Science and Technology (QST), Rokkasho, Aomori, Japan

Abstract

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-class 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. ITER-CLASS DEMO REACTOR 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 systems code TPC. As shown Table 1, the following three phases were envisioned. Table 2 shows the main parameters of the ITER-class 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), $P_{net} \sim 5$ 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). Figure 1 (a) and (b) show the cross section of phase I and phase II & III, respectively. To compensate for the reduced plasma volume, a conventional JA DEMO scenario (normalized beta β_N = 3.4, normalized density f_{GW} = 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.

TABLE 1. Phased approach strategy for ITER-class DEMO reactor

	Phase I: System Integrating Operation (Power Generation Demonstration)	Phase II: Blanket Functional Test (Fuel Breeding Demonstration)	Phase III: Extended Operation (Steady-State Operation Demonstration)
Objective	· Short pulse (several minutes) · P_{gross} > ~180 MW · P_{net} ~ 0	· Long pulse (several hours) · P_{net} ~0 · Self-sufficiency of fuel	· Steady-state operation · P_{net} > 0 (~100 MW) · Self-sufficiency of fuel
Specification	· ITER baseline scenario ✓ P_{fus}: 500 MW ✓ Q: 10 ✓ Pulse length: ~400 s · Electricity generation and shielding blanket ✓ Electricity generation and shielding blanket ✓ Same size as ITER shielding blanket · Heating and Current Drive Device ✓ ECH only	· Original JA DEMO baseline scenario ✓ P_{fus}: >500 MW ✓ Q: 10 · Fuel breeding demonstration ✓ Breeding blanket · Heating and Current Drive Device ✓ ECH and NBI	· JT-60SA scenario (High β & High confinement) ✓ P_{fus}: ~1 GW · Improved breeding blanket · Improved efficiency of heating and current drive device · Confirmation of the procedure and time for blanket replacement by remote handling.

TABLE 2. Main parameters of the ITER-class 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
A	3.1	3.76	3.76
V_p [m^3]	835	569	569
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
Pluse length	337 s	3.98 h	Steady-State
P_{fus} [MW]	492	510	820
Q	10	10	14
P_{net} [MWe]	7.3	9.3	82.5
P_{gross} [MW]	188	195	307
β_N	1.8	3.4	4.3
HH_{98y2}	0.95	1.41	1.50
f_{GW} (n_e/n_{GW})	0.85	1.19	1.20
Breeding/shielding zone [m]	- / 0.45	0.5 / 0.35	0.5 / 0.35
Net TBR	-	1.05	1.05

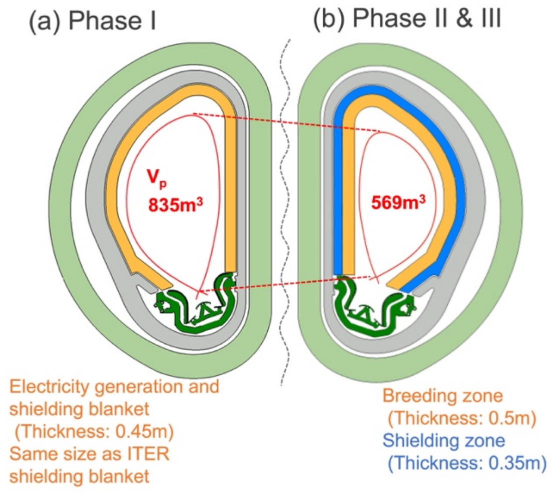

FIG. 1. Cross section of phase I and phase II & III.

In this concept, the phase I is based on the ITER baseline scenario and is reasonably promising. On the other hand, in the phase II, the breeding blanket will be installed, reducing the plasma volume by approximately 30%, requiring confinement performance 1.5 times that of the phase I. This means that a strong internal transport barrier (ITB) and a reversed magnetic shear plasma must be assumed. Furthermore, the beta value is 1.8 times that of the phase I, requiring a value above the no-wall beta limit. These plasma parameters have not yet been experimentally verified, and this remains a challenge. Furthermore, the phase III will require an even greater leap in plasma parameters than the phase II (beta value 2.4 times that of the phase III).

3. OPTION OF HIGHER MAGNETIC FIELD & LARGER TF COIL

To ease the plasma performance requirements of the original ITER-class DEMO reactor, we considered options for increasing the magnetic field on plasma axis B_t and plasma volume V_p. To achieve these goals, we assumed that the TF coils would be higher magnetic field and the coil bore would be enlarged. However, increasing the coil bore too much would increase development risks, as in the previous JA DEMO. Therefore, we decided to extend the bore from the ITER TF coil to 1.2 m, keeping the ITER TF coil fabrication facilities as far as possible within the usable range. To minimize alpha particle loss due to TF ripple, we fixed the outer plasma surface position at 8.6 m, where the TF ripple value is 1% for 16 TF coils. Since higher ellipticity and aspect ratios are more likely to cause Vertical Displacement Event (VDE), set $\kappa 95 = 1.7$ and $A \leqq 3.1$, respectively, similar to ITER.

Figure 2 shows the system code analysis results for each operational phase. Table 3 lists representative plasma parameters for each operational phase. The higher magnetic field and larger coil bore reduce the aspect ratio A and increase the safety factor q_{95} compared to the original ITER-class DEMO concept. As a result, in the phase I, plasma stability was improved, the pulse length was increased, and $P_{net} > 0$ was achieved at low beta values. In the phase II, similar effects were observed, resulting in a reduced confinement improvement factor HH_{98y2} and a reduced normalized density f_{GW}, achieving a P_{net} equivalent to the original concept even below the no-wall beta limit. However, the reduced pulse length due to the reduced magnetic flux supplied by the CS is a disadvantage in the phase II. In the phase III, in addition to improving stability through a reduced aspect ratio and increased safety factors, $P_{net} > 100$ MW is within reach, and the normalized density and beta values have also been reduced. However, the plasma parameters in the phase III remain challenging, and the development of high-performance plasma is essential. It is also important to reduce the required beta value by increasing the plasma radius by reducing the thickness of the breeding blanket and shielding.

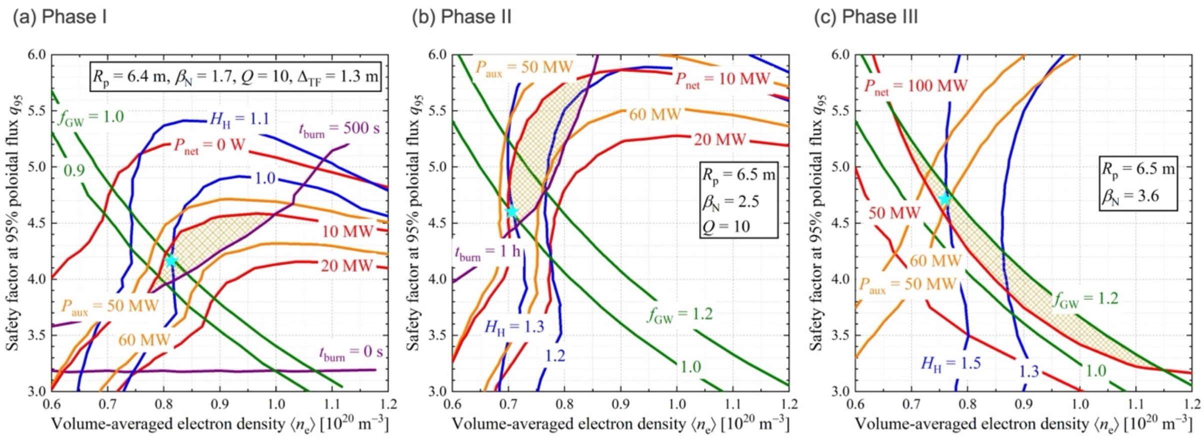

FIG. 2. System code analysis results of the higher magnetic field and larger TF coil option for each operational phase.

TABLE 3. Main parameters of the higher magnetic field and larger TF coil option for each operational phase.

	Phase I	Phase II	Phase III
R_p / a_p [m]	6.4 / 2.2	6.5 / 2.1	6.5 / 2.1
A	2.91	3.10	3.10
κ_{95}	1.7	1.7	1.7
q_{95}	4.16	4.60	4.71
I_p [MA]	13.5	10.6	10.4
B_t [T]	5.56	5.47	5.47
Pluse length	564 s	1.08 h	Steady-state
P_{fus} [MW]	540	531	917
Q	10	10	16
P_{net} [MWe]	11.3	10.9	107
P_{gross} [MW]	206	203	342
β_N	1.7	2.5	3.6
HH_{98y2}	1.00	1.30	1.50
f_{GW} (n_e/n_{GW})	0.99	1.00	1.15

Table 4 shows the TF coil specifications for the options of increasing the magnetic field and enlarging the coil bore. The superconducting strands are assumed to be Nb_3Sn, as in ITER. To minimize design changes from the ITER TF coil, a maximum toroidal magnetic field of approximately 13 T is assumed using radial plates with the same number of turns as the ITER TF coil, resulting in a conductor current of 85 kA. Assuming the radial thickness of the TF coil is approximately the same as ITER's (0.91 m), the outer diameter of the conductor strands must be approximately ϕ39 mm compared to ϕ39.7 mm of ITER. This value requires an increase in the conductor current density from I_c = 55 A/mm^2 to I_c = 71 A/mm^2. This value corresponds to 1.3 times that of the ITER TF conductor. This increase may be achievable by increasing the current density of the Nb_3Sn strands and reducing performance degradation due to short twist pitch stranding, but verification through the fabrication of a full-scale conductor and conductor testing is essential. As the magnetic field increases, the electromagnetic force acting on the coil case will increase by approximately 200 MPa, requiring the cryogenic steel for the coil's inner legs to be stronger than those used in ITER. While Japan's previous research and development efforts into the development of cryogenic steel capable of withstanding a design stress of 800 MPa have shown promise [3], further R&D is needed to verify weldability and the feasibility of fabricating large structures. Furthermore, since the coil will be somewhat larger than the ITER TF coil, the development of new attachments for TF coil fabrication may be necessary. Furthermore, since the coil will be larger than the ITER TF coil, increased fabrication costs are expected. Therefore, rationalization of coil fabrication based on ITER's lessons-learn will be important. Given these factors, the issues of the options of increasing the magnetic field and enlarging the coil bore are the increased development time associated with the increased magnetic field due to R&D and fabrication costs.

TABLE 4. Main parameters of the TF coil of the higher magnetic field and larger TF coil option.

	ITER	Option
SC strand	Nb_3Sn	Nb_3Sn
Number of TFC	18	16
B_{tmax} [T]	11.8	~ 13
Conductor current [kA]	68	85
Number of turns per TFC	134	134
Total magneto motive force [MAT]	164	182
Total magnetic energy [GJ]	41	~55
Design stress [MPa]	667	800
Width / Height of TFC [m]	8 / 12.3	9.2 / 14.2

4. OPTION OF BLANKET AND SHEILDING

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. Figure 3 shows the calculated shielding performance and neutron energy multiplication factor on a shielding thickness of 0.45 m in the phase I. The left vertical axis represents the shielding performance (i.e., the lifetime of the insulation on the TF coil), and the right vertical axis represents the neutron energy multiplication factor. The shielding candidates considered included a mixture of low-activation ferritic steel F82H and water, zirconium hydride ZrH_2, titanium hydride TiH_2, tungsten boride W_2B_5, hafnium hydride HrH_2, tungsten carbide WC, and various mixture ratios of tungsten carbide and water. The analysis code used was MCNP-5, the nuclear data was FENDL-2.1, the analysis system was a stacked cylinder model, and the source strength (neutron wall load to the inner blanket first wall) was 0.56 MW/m^2. Considering the need for cooling water to remove heat from the shield, tungsten carbide offers relatively high shielding performance and is a promising candidate. The breeding blanket with high tritium breeding and shielding performance and thinning of the shielding layer are important research topics for the future.

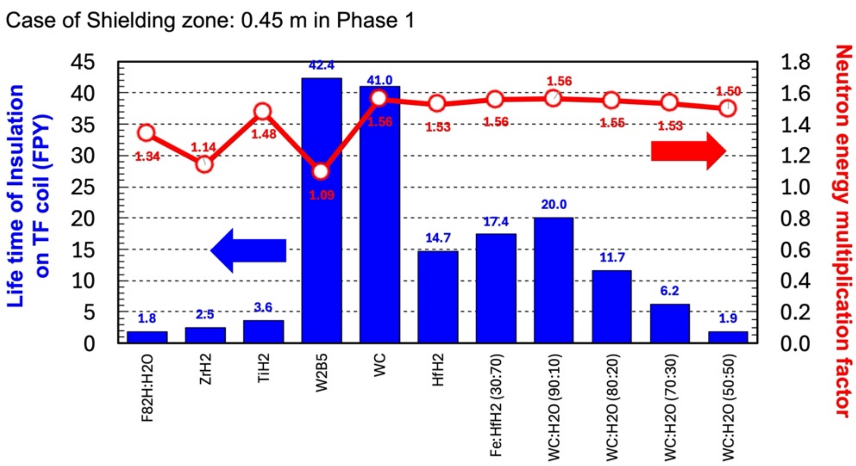

FIG. 3. Calculation results of shielding performance and neutron energy multiplication factor on a shielding thickness of 0.45 m in the phase I.

5. 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.

To mitigate the core plasma performance required to achieve the target, it is effective to increase the magnetic field and enlarge the TF coil bore, but to realize this option, early start of the necessary R&D and rationalization of manufacturing costs are required. In addition, thinning the breeding blanket and shielding while maintaining the tritium breeding shielding performance (high performance) is an essential R&D item for this reactor concept and compact fusion reactors.

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