DEVELOPMENT OF PHYSICS AND ENGINEERING DESIGNS FOR JAPAN’S DEMO CONCEPT

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Abstract

Recent progress of Japan’s DEMO design is presented. The key concept is a steady-state DEMO with a major radius of 8 m class and fusion power of 1.5 GW level, which is proposed based on ITER physics and technology bases, and characterized by operational flexibility from pulse to steady-state operations. Even in a steady-state DEMO, the pulse operation is required for the commissioning of plant systems and also suitable for early demonstration of fusion electricity by moderate plasma performance. Regarding the physics design, divertor plasma simulation clarifies that the lower density to be compatible with detached plasma, which is consistent with the operational density of JA DEMO. Vertical stability evaluation by 3D eddy current and plasma control model shows that plasma elongation of 1.75 is sustainable by applying the double-loop type shells. The development of plasma operation scenario indicates the importance of off-axis ECCD for controlling the internal transport barriers. In addition to physics design, engineering designs are performed in wide area. The divertor cassette design is developed for reducing the fast neutron flux to protect the vacuum vessel and for replacement of the power exhaust units. The breeding blanket concept based on JA ITER-TBM strategy is developed to increase the pressure-tightness of the modules by considering safety assessment of in-box LOCA. On the TF coil design, assessment of the error field indicates that the fabrication tolerance can be mitigated by ~2.5 times as large as ITER’s with correction coil current of several 100 kAT/coil. The concept of remote maintenance for the blanket segments is developed such as the stable transfer mechanism in the vertical, radial and toroidal directions. The rad-wastes generated by the maintenance can be disposed of in shallow land burial after 10-year storage. The concept of primary cooling water system is developed for effective use of thermal power removed from not only blanket but also divertor.

1. INTRODUCTION

The Japan-wide pre-conceptual DEMO design activity is recently conducted by the Joint Special Design Team for fusion DEMO organized in 2015 to establish JA DEMO concept by narrowing down the design through the evaluation and integration of design elements, and to clarify R&D issues to be implemented during the conceptual design phase after 2020 [1,2]. The requirements for the DEMO presented by MEXT are to demonstrate (1) steady and stable electric power generation in a power plant scale, (2) reasonable availability using a remote maintenance scheme anticipated in a commercial plant, and (3) overall tritium breeding to fulfill self-sufficiency of fuel. The steady-state DEMO concept of JA DEMO [3] is used as the primary one, which has a relatively large major radius $R_p = 8.5$ m for inductive $I_p$ ramp-up and a relatively low fusion power $P_{\text{ fus}} = 1.5$ GW for divertor heat removal. Since the feasibility of such a steady-state DEMO concept has not been confirmed so far, the physics and engineering design activity is in progress to find a design solution meeting the above requirements.

2. OVERVIEW OF JA DEMO CONCEPT

2.1. Concept and design parameters

JA DEMO concept was proposed in 2014 by considering the technological feasibility based on the physics and technology basis for ITER [3]. In particular, components of divertor, breeding blanket and superconducting magnet are designed based on ITER technology bases as much as possible. The divertor concept is a lower single null W-shaped divertor configuration with water cooled tungsten mono-block target, while cooling tubes are composed of a Cu alloy for high heat flux region and a RAFM (reduced activation ferritic martensitic) for high neutron flux region, respectively [4]. The breeding blanket concept is water-cooled ceramic breeder based on ITER-TBM strategy in Japan [5]. Based on the toroidal field (TF) coil design in ITER, Nb$_3$Sn wire material, the cable-in-conduit conductor structure, the radial plate structure for conductor support, and the double pancake winding are also considered for TF coil design for JA DEMO, while the allowable design stress of 800 MPa is
larger than that of ITER. On the other hand, remote maintenance scheme, radioactive waste management and primary heat transfer system (PHTS) are required for developing the alternative concepts beyond the ITER technology bases for JA DEMO because of high radiation environment and electricity generation. By assessment of maintenance scheme of sector and segmented transfer methods, the segmented maintenance scheme provides advantages to the ease of handling of the in-vessel components, the layout of poloidal field (PF) coils and their magnetic energy, the size of TF coils, the support of turnover force acting on TF coils [6]. Therefore, the segmented maintenance scheme is applied to JA DEMO where the breeding blanket segments are vertically transferred from upper port, and divertor cassettes horizontally from lower port. Based on the assessment of radioactive nuclide transport via likely pathways in the biosphere, all waste is disposed in shallow land burial as a low-level radioactive waste after 10-year storage [7]. The concept of primary cooling water system is designed based on the pressurized water reactor, in which the coolant water condition is 15.5 MPa and the temperature range from 290 to 325 °C, to minimize the R&D items related to electricity generation.

In addition, the concept of JA DEMO is characterized by operational flexibility from pulse to steady-state operations. Even in a steady-state DEMO, the pulse operation is required for the commissioning of plant systems and also suitable for early demonstration of fusion electricity by moderate plasma performance. The latest view of JA DEMO and main parameters for both the steady-state and pulse operations are shown in Fig. 1. The requirements of plasma performance in confinement and stability are significantly reduced for the pulse operation. The major radius of 8.5 m enables to supply CS flux sufficient for Ip ramp-up inductively for steady-state operation with fusion power of 1.5 GW, and for 2 hours pulse operation with 1 GW. It should be noted that it is difficult to perform the pulse operation with reduced Ip in JA DEMO. Normally, the reduced Ip saves the CS flux consumption and allows pulse operation. However, compatibility with divertor detachment will be a problem, because the reduced Ip leads to lower operation density due to decrease of Greenwald density limit. Therefore, the pulse operation needs to be performed with the rated Ip in JA DEMO.

FIG. 1. Latest view of JA DEMO and main parameters.

2.1. Major issues raised for JA DEMO

A DEMO must be a highly integrated device for plasma physics and engineering. In particular, for the case of JA DEMO with R_p = 8.5 m, the following issues to be solved are raised. (i) The first is compatibility between core and divertor plasmas. Since the Greenwald density limit is inversely proportional to R_p, the operational density is lower in the large-size device. On the other hand, high density operation is suitable for sustainment of detached plasma. Therefore, it is necessary to clarify the required condition on the lower density for the sustainment of detached plasma. (ii) The second is to ensure vertical position stability by conducting walls and to access to the high beta above no-wall beta limit. In a DEMO, a thick breeding blanket is required for obtaining tritium self-sufficiency, which makes the conducting wall located behind the blanket far away, and then its effect on plasma would be limited. This issue would be impact on design elongation and achievable beta. Therefore, it is necessary to clarify the requirements of conducting wall position, shape and so on. (iii) The third is a countermeasure against a loss of coolant accident, because JA DEMO adopts water cooling system with the pressurized water condition. Therefore, when the cooling water leaks inside the blanket, the blanket module must be designed to withstand the inner pressure load. (iv) The fourth is management of radioactive waste, because the large amount of radioactive waste is generated in JA DEMO due to the periodic replacement of...
large in-vessel components. It is important to develop the maintenance scenario of in-vessel components, the management scenario of radioactive waste, and disposal classification. (v) The fifth is the requirement of fabrication tolerance to reduce the error field for avoidance of the locked mode. Since a high fabrication accuracy is required for the large TF coil, mitigation of the fabrication tolerance is required for JA DEMO.

3. PROGRESS OF PLASMA DESIGN

3.1. Divertor and pedestal plasma

One of the major physics issues in a large-sized steady-state tokamak is the compatibility between the operational density and the divertor detachment because of the low Greenwald density limit due to high $q_{95}$ and large $R_p$. Requirements for divertor plasma design are (i) peak heat load on divertor target plate less than 10MWm$^{-2}$ for heat removal by W mono-block and Cu alloy cooling pipe, (ii) electron temperature in attached area less than 20-30 eV for avoiding significant net erosion of W mono-block, and (iii) Ar impurity concentration in SOL less than 0.5-1.0%. In order to investigate the compatibility between the core operational density and divertor plasma, massive parameter scan has been performed by SONIC divertor plasma simulation for JA DEMO, as shown in Fig. 2. The results indicate that the separatrix density of $n_{sep} > 2.1 \times 10^{19}$ m$^{-3}$ satisfies above requirements, which is roughly consistent with the pedestal density of $\sim 6.6 \times 10^{19}$ m$^{-3}$ in JA DEMO [8].

![Fig. 2](image_url)

*FIG. 2. Dependence of (a) peak heat load on outer divertor target, (b) Ar impurity concentration in SOL, and (c) maximum electron temperature at attached area, on separatrix density.*

Based on the evaluation on the pedestal density, pedestal and ELM modelling are performed by EPED1 model and MARG2D stability code, suggesting that the pedestal temperature is $\sim 3$ keV and the equilibrium is near kink/peering stability boundary which can be regarded as the QH-mode regime. Furthermore, pellet ablation and drift simulation is performed by using HP12 code based on the evaluation of pedestal structure, indicating that the pellet with speed of 2 km/s and mass of $4 \times 10^{21}$ atom/pel can deposit at $r/a \sim 0.85$ from the high field side top injection with poloidal angle of 120 degrees [9].

3.2. Vertical stability

A Conducting shell is an essential component for vertical stability and high beta operation over the free boundary MHD limit. However, a thick breeding blanket makes the conducting shell located behind the blanket far away from the plasma ($r_W/a \sim 1.35$), and no in-vessel control coils due to large neutron irradiation environment and their maintainability. Vertical stability evaluation by 3D eddy current and plasma control model shows that plasma elongation of 1.75 is sustainable by applying the double-loop type shells [10].
3.3. Core plasma scenario development

In order to develop the integrated scenario modeling, the core plasma modeling has been performed by using 1.5-D time-dependent transport code (TOPICS). The CDBM, current diffusive ballooning mode, transport model is used in the simulation, because the CDBM transport model could well reproduce the ITB plasmas observed in JET and JT-60U tokamak experiments [11]. In the preliminary scenario modeling, the temporal evolution and profile of electron density are given, and fraction of impurities are fixed at nHe/ne = 0.07 and nAr/ne = 0.005, respectively. The basic scenario is as follows: (i) a plasma current (Ip) is ramped up with the rate of 0.15 MA/s which is the same in the ITER reference scenario and (ii) after increasing the target density during Ip flat-top phase, the neutral beams (NBs) are injected to produce a burning plasma by forming the internal transport barriers (ITBs) and edge transport barriers (ETBs). The typical waveforms obtained in the simulation are shown in Fig. 4. The NB power PB = 70 MW is injected with the beam energy of 1 MeV and tangency radius of injection Rtan = 9.0 m. The ECRF waves are injected from low field side and with toroidal angle of 150 degrees. Since L- to H-mode transition is not modelled at present, the edge transport barriers are artificially formed when the power across separatrix Psep exceeds the H-mode threshold power of ~100 MW. After ITB formation and growth of alpha heating power, ITBs cannot be sustained even by increasing PB, where the location of ITB is shrunk which might be related to the change in safety factor (q) profile. In order to sustain the ITBs, off-axis ECCD (PEC = 30 MW, O-mode, 190 GHz) is applied near the ITB foot location, and then the shrinkage of ITBs is successfully avoided and almost steady-state plasma is obtained with full non-inductive current drive condition as shown in Fig. 4. In this simulation, the confinement enhancement factor over the H-mode scaling HH98y2 = 1.41, $\beta_N = 3.6$, the bootstrap current fraction $f_{BS} = 0.69$ and sum of NBCD and ECCD fraction $f_{CD} = 0.32$ are obtained at $t = 200$ sec, respectively.

FIG. 3. (a) Conducting structures such vacuum vessel, backplate and double-loop type shells. (b) Plasma displacement and (c) controlled PF coil power for plasmas with $\beta_N = 0.1$ and 1.8, plotted against plasma elongation.

FIG. 4. Typical waveforms simulated by TOPICS (left) and profiles of ion temperature and safety factor.
The MHD stability analysis for JA DEMO has been performed by using MARG2D which is a linear ideal MHD stability code [12]. The beta limit without conducting wall is $\beta_N \sim 2.6$. The beta limit with conducting wall is improved to $\sim 3.5$ when the wall is located at $r_w/a = 1.35$. Further improvements are observed with decreasing wall radius, for example $\beta_N \sim 3.9$ at $r_w/a = 1.30$. It should be noted that the shorter wall radius is required to achieve $\beta_N \sim 3.4$ assumed in JA DEMO, indicating that reconsideration of the aspect ratio or width of blanket module might be required for moving the conducting shells relatively close to plasma.

4. PROGRESS OF KEY ENGINEERING DESIGNS

Progress of engineering design has been made in wide area, where the ITER technologies are utilized as much as possible in divertor, blanket and TF coil designs, while remote maintenance, plant systems and rad-waste management are developed as a challenge inherent in DEMO.

4.1. Divertor

The design concept of divertor is basically similar to that of ITER, that is a water-cooled tungsten mono-block target. By considering the neutron irradiation environment, a Cu-alloy heat sink is used only in the large heat flux region, where DPA rate is less than $\sim 1.5$ dpa/year, while a RAFM (reduced activation ferritic martensitic) steel heat sink is applied in the large neutron flux region [3]. An arrangement of the coolant rooting for the Cu-alloy and RAFM pipes in the divertor cassette is developed and the heat removal capability of the divertor target was evaluated, indicating that the peak heat load of 10 MWm$^{-2}$ for Cu-alloy and 5 MWm$^{-2}$ for RAFM can be handled. However, replacement of divertor is expected every 1-2 years due to reduction in fracture toughness of the Cu-alloy pipe, which leads to increase the amount of radioactive waste ($\sim 920$ ton every replacement). To improve the situation, new design of divertor cassette is developed, where the heat sink unit of tungsten mono-block with the Cu alloy pipe is independently replaced as a module type structure as shown in Fig. 6. In addition to the target design, a conceptual design of the divertor cassette is developed for reducing the fast neutron flux to protect the vacuum vessel, handling the nuclear heat of the cassette, and replacement of the target units with the Cu-alloy heat sink as shown in Fig. 6.

FIG. 6. Divertor concept consists of the inner thick plate with water puddles for reducing the fast neutron flux to protect the vacuum vessel and of pipe arrangement for replacement of the target units.
4.2. Breeding blanket

The design concept of breeding blanket is based on water-cooled pebble bed, which is the same as JA concept of ITER-TBM. Considering assessment of safety analysis of in-box LOCA, which is an accident of cooling pipe break inside blanket module, the blanket module design is developed to withstand against the in-box LOCA by increasing pressure-tightness (17.2 MPa), where the cross-section of module is divided into square with 0.1 m per side by 0.015 m-thick ribs as shown in Fig. 7. By optimizing cooling route based on thermal analysis, the total length of cooling pipe is shortened, which contributes to reduce the pressure drop of coolant and to increase the tritium breeding volume. The tritium breeding ratio of TBR~1.05 is achievable with the blanket thickness of ~ 0.7 m [13].

FIG. 7. Breeding blanket concept with pressure tightness against in-box LOCA.

4.3. Toroidal magnetic field coil

The TF coil design is considered based on the ITER technology. The problem of the large size coil is the requirement of fabrication tolerance to reduce the error field for avoidance of the locked mode. The TF coil of JA DEMO is about 1.6 times larger than that of ITER. Assuming the fabrication tolerance of ITER, a high fabrication accuracy is required for JA DEMO. In order to mitigate the fabrication tolerance, the relation between the error magnetic field and the tolerance in the fabrication and installation of the superconducting coils are evaluated. Figure 8 shows the error magnetic field histogram at q = 2 surface obtained by error magnetic field analysis using the Monte Carlo method, where the fabrication and installation tolerances are mitigated by about 2.5 times as large as ITER’s. The error magnetic field at the cumulative relative frequency of 95% is 5.2 G, which satisfy the error magnetic field target of $B_{em}/B_T < 10^{-4}$ for JA DEMO. Furthermore, the correction coil current of several 100 kAT/coil can reduce the error magnetic field of about 3 G, which is required for avoiding the locked mode [14].

FIG. 8. Results of error magnetic field analysis. Error magnetic field histogram at q = 2 surface shown in red and cumulative relative frequency shown in blue.
4.4. Remote maintenance and radioactive waste

The concept of remote maintenance for the blanket segments is developed as shown in Fig. 9, in particular, the stable transfer mechanism in the vertical, radial and toroidal directions. After removed from the vacuum vessel, the blanket segment is installed in cask to control the scattering of radioactive dust [15]. It is found that the rad-wastes generated by the maintenance can be disposed of in shallow land burial after 10-year storage [7].

![Fig. 9. Concept of remote maintenance with stable transfer mechanism in three-axial directions for blanket segments.](image)

4.5. Cooling water system

The concept of primary cooling water system is developed for effective use of thermal power removed from not only blanket but also divertor. The difficulty in case of direct connection of the cooling lines of blanket and divertor is to raise the large pressure drop, which requires a large pumping power. In the concept shown in Fig. 10, thermal power removed from the divertor is used for preheating the blanket coolant, and a bypass line is installed to control the coolant flow for reducing the pressure drop [16].

![Fig. 10. Concept of primary cooling water system to utilize the heat generated in blanket and divertor for power production.](image)

5. SUMMARY

The Japan-wide pre-conceptual DEMO design activity is recently conducted by the Joint Special Design Team for fusion DEMO organized in 2015 to establish JA DEMO concept by narrowing down the design through the evaluation and integration of design elements, and to clarify R&D issues to be implemented during the conceptual design phase after 2020. The JA DEMO concept is characterized by operational flexibility from pulse to steady-state operations, which has the major radius \( R_p = 8.5 \) m for Ip ramp-up inductively and the fusion power \( P_{fus} = 1.5 \) GW for divertor heat removal in steady-state operation, and 1 GW in 2 hours pulsed operation for early demonstration of a fusion electricity by a moderate plasma performance. Recent development of JA
DEMO concept is summarized as follows: (i) Divertor plasma simulation indicates that the compatibility between the core operational density and divertor plasma. (ii) Vertical stability evaluation by 3D eddy current and plasma control model shows that plasma elongation of 1.75 is sustainable by applying the double-loop type shells. (iii) The importance of off-axis ECCD for controlling the internal transport barriers. (iv) The divertor cassette design is developed for reducing the fast neutron flux to protect the vacuum vessel and for replacement of the power exhaust units. (v) The breeding blanket concept based on JA ITER-TBM is developed to increase the pressure-tightness of the modules by considering safety assessment of in-box LOCA. (vi) Assessment of the error field indicates that the tolerance of TF coil can be mitigated by ~2.5 times as large as ITER’s with correction coil current of several 100 kAT/coil. (vii) The concept of remote maintenance for the blanket segments is developed such as the stable transfer mechanism in the vertical, radial and toroidal directions. (viii) The rad-wastes generated by the maintenance can be disposed of in shallow land burial after 10-year storage. (ix) The concept of primary cooling water system is developed for effective use of thermal power removed from not only blanket but also divertor. It is concluded that the feasibility of a steady-state DEMO concept with a large size of 8 m class and a lower fusion power of 1.5 GW level is basically confirmed.

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