<Technical Note>

TOKAMAK REACTOR SYSTEM ANALYSIS CODE FOR THE CONCEPTUAL DEVELOPMENT OF DEMO REACTOR

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Tokamak reactor system analysis code was developed at KAERI (Korea Atomic Energy Research Institute) and is used here for the conceptual development of a DEMO reactor. In the system analysis code, prospects of the development of plasma physics and the relevant technology are included in a simple mathematical model, i.e., the overall plant power balance equation and the plasma power balance equation. This system analysis code provides satisfactory results for developing the concept of a DEMO reactor and for identifying the necessary R&D areas, both in the physics and technology areas for the realization of the concept. With this system analysis code, the performance of a DEMO reactor with a limited extension of the plasma physics and technology adopted in the ITER design. The main requirements for the DEMO reactor were selected as: 1) demonstrate tritium self-sufficiency, 2) generate net electricity, and 3) achieve a steady-state operation. It was shown that to access an operational region for higher performance, the main restrictions are presented by the divertor heat load and the steady-state operation requirements.

KEYWORDS : Fusion Reactor, DEMO Reactor, System Analysis Code

1. INTRODUCTION

The development strategy of nuclear fusion energy in the Korea National Basic Plan for the Development of Fusion Energy consists of several major programs. These are shown in Fig. 1, and include KSTAR for the study of a long-pulse, advanced tokamak operation, ITER for a burning plasma experiment, the DEMO reactor for the demonstration of producing net electricity from a fusion reactor, and a commercial fusion reactor. Material testing and integral testing of the reactor components must be performed using IFMIF and CTR (Component Test Reactor). The demonstration fusion power plant DEMO reactor is regarded as the last step before the development of a commercial fusion reactor. The primary requirements for the DEMO reactor can be summarized as follows: First, it should demonstrate net electric power generation. Second, it should demonstrate tritium self sufficiency. Lastly, it should demonstrate the safety aspects of a power plant and should be licensable as a power plant.

To develop the concepts of fusion reactors and identify the design parameters, dependence on performance objectives, design features and physical and technical constraints have to be considered. System analyses are necessary to find reactor parameters that can optimize figures of merit such as the major radius, ignition margin, divertor heat load, and neutron wall load. In a system analysis, effects of the plasma physics and technology constraints are expressed in simple mathematical model and are incorporated into a plant power balance equation and a plasma power balance equation. Thus, by solving the plant power balance equation and the plasma power balance equation, the reactor parameters that satisfy the plasma physics and technology constraints can be found simultaneously. A similar approach was used in the scoping studies for the ITER, the Conceptual Design Activity. The basis of the applied physics can be found in the ITER Physics Basis [1, 2].

To explore the range of concepts of a DEMO reactor and a fusion power plant, assumptions on the level of physical and technological development have to be made. There will be many reactor models depending on the assumptions, from the least ambitious plasma physics combined with the least ambitious technologies to the most ambitious in all areas. Therefore, it is stressed that the system analyses are intended to capture the range of likely outcomes and to identify the necessary R&D areas for the realization of the concept in terms of both physics and technology. In this study, as a part of a feasibility study for an early realization of a DEMO reactor, the performance of an ITER-like DEMO reactor was investigated; the plasma and machine size are identical to those of ITER

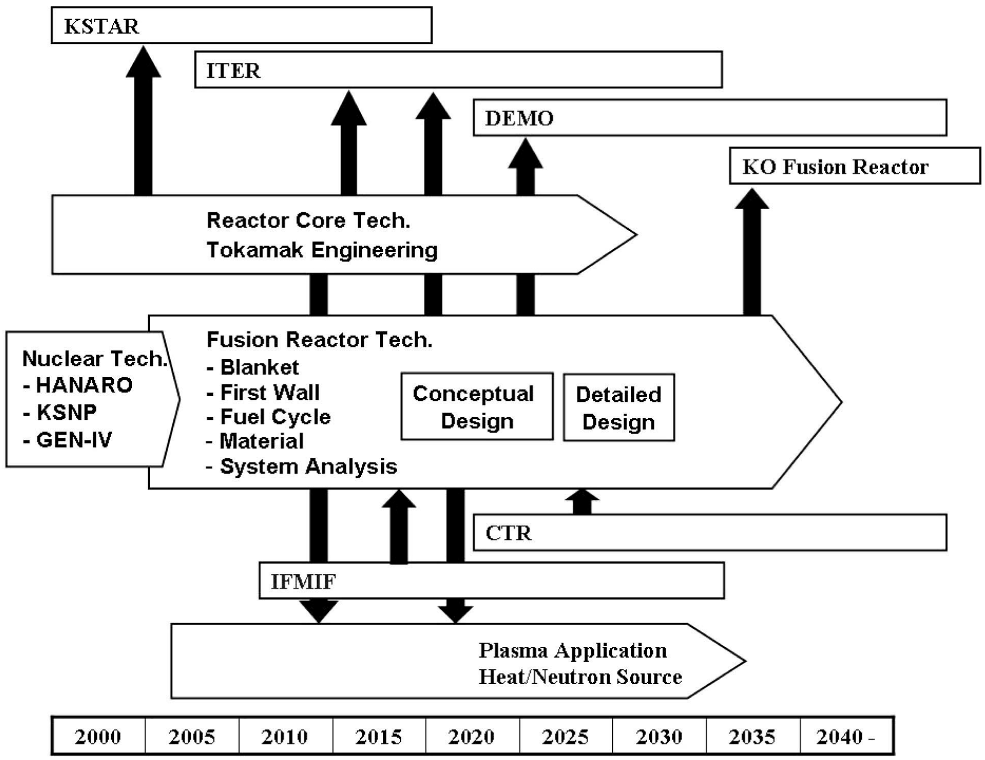

Fig. 1. Fusion Technology Roadmap of Korea

but the plasma physics and technology are assumed to be improved from those adopted in the ITER design.

Organization of this paper is as follows. The structure of the system code with the physical and technological constraints is explained in Sec. 2. The performances of a DEMO reactor for an early realization are given in Sec. 3. In Sec. 4, the conclusion is given.

2. DEVELOPMENT OF THE TOKAMAK REACTOR SYSTEM CODE

The system analysis code finds the design parameters that satisfy the plasma physics and technology constraints. It includes the range of likely outcomes for the development of the plasma physics and technology. The results arising from the system analysis are used to define the concept of a reactor and identify the necessary R&D areas to realize the derived concept.

2.1 Plasma and Plant Power Balance Model

In a system analysis code, the mathematical models to capture the physics and technologies are the overall plant power balance equation and the plasma power balance equation. The first equation is the plasma power balance equation, which is represented as

$$P_{con} + P_{rad} = P_{OH} + P_{\alpha} + P_{CD} \tag{1}$$

where the conduction (P_{con}) and radiation losses (P_{rad}) are balanced by α particle heating (P_{α}), auxiliary heating (P_{CD}) and ohmic heating (P_{OH}). These terms have a complex dependency on the plasma parameters. For the confinement scaling, the H-mode IPB98y2 scaling law [1] is used

$$\tau_E = H \tau_E^{IPB98(y,2)} \tag{2}$$

$$\tau_E^{IPB98(y,2)} = 0.0562 I_P^{0.93} B_0^{0.15} (P_{con} vol)^{-0.69} n_{19}^{0.41} M^{0.19} R_0^{1.97} \left(\frac{a}{R_0}\right)^{0.58} \kappa^{0.78} \tag{3}$$

where I_p is the plasma current (MA), P_{con}, is the power loss (MW), n_{19} is the line averaged density (10^{19}m^{-3}), B_o is the toroidal magnetic field (T) at the magnetic axis, M is the fuel mass number (amu), R_0 is the major radius (m), a is the minor radius (m), and k is the inverse aspect ratio. In

Eq. (2), H was introduced to represent the confinement enhancement factor.

The second equation is a plant power balance equation, which accounts for the energy multiplication, the efficiency of electricity generation and the power consumption in the current drive, cryogenics, and other systems. The overall plant power balance includes complex dependencies on plant parameters. Fig. 2 illustrates the power flow in plant systems.

2.2 Physics Model

The plasma physics properties are expressed in a zero-dimensional model in the system analysis code. The zero-dimensional model cannot consider the profile effects precisely, such as the heating and current drive profile, the bootstrap current fraction, or advanced tokamak operation with a negative shear. However, this approach will provide satisfactory results in the selection of a reactor concept. For further development of the reactor concept, detailed analyses of the plasma performance including the MHD equilibrium, stability, transport and current drive analyses are required. The physics models used in the system analysis code are identical to those used for the design of ITER [1, 2].

The total plasma current I_p is limited by the limit of the safety factor q_{95} at the edge.

$$q_{95} = \frac{B_0 R_0}{2\pi} \int \frac{ds}{R_2 B_p} \cong q_* \frac{(1.17 - 0.65\frac{a}{R})}{(1-(\frac{a}{R})^2)^2} \tag{4}$$

$$q_* = \frac{5 a_2 B_0}{R_0 I_p} \frac{1 + k_{95}^2 (1 + 2\delta_{95}^2 - 1.2\delta_{95}^3)}{2} \tag{5}$$

Recent experiments in many devices have shown that the limit on the plasma beta value β, is imposed by MHD instability of neoclassical tearing modes (NTMs) or by the resistive wall mode (RWM). Appropriate control of the plasma shape and profiles will allow access to values of $\beta_N > 3$, which is a required value for a power plant. The plasma beta limit is typically expressed as [3]

$$\beta \le \beta_T = C_T (I_P / aB_0) \tag{6}$$

where C_T is a Troyon coefficient.

Operation at a high density is favored but there is a limit above which the plasma becomes disruptive. A further constraint on a plasma density arises from the need to limit the power flux to the divertor target, which limits the acceptable peaking of the density profile. The expression for a density limit is given by Murakami-Hugill scaling, Borass scaling or Greenwald scaling, as follows:

$$n_{MH} = \frac{2B_0}{R_0 q_*} \quad (10^{20} m^{-3}) \tag{7}$$

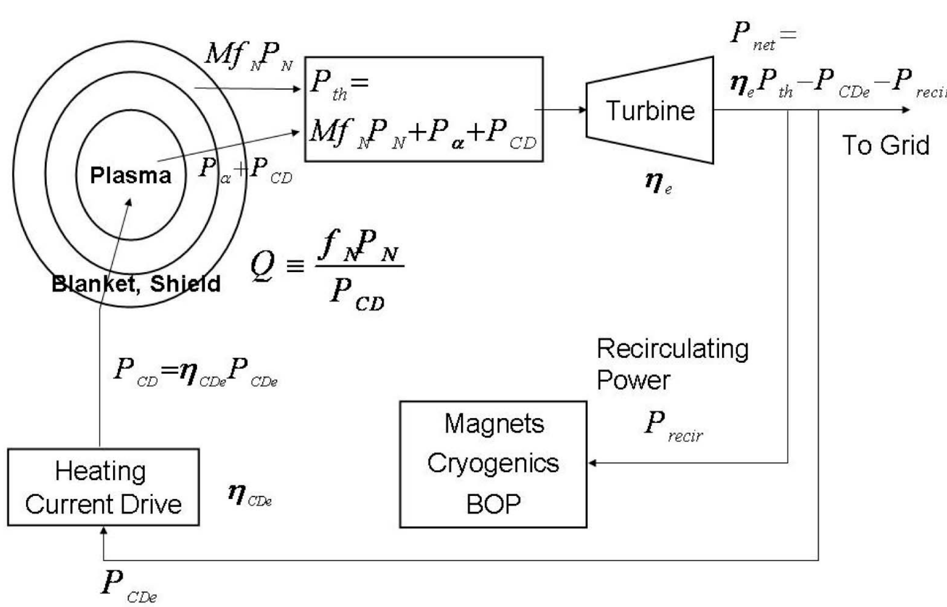

Fig. 2. Power Flow in a Tokamak Fusion Power Plant

$$n_B = \frac{Q^{0.57} B^{0.31}}{2(qR)^{0.09}} \quad (10^{20} m^{-3}) \tag{8}$$

$$n_G = \frac{I_P}{\pi a^2} \quad (10^{20} m^{-3}) \tag{9}$$

Steady-state operation requires that the plasma current be fully driven by the non-inductive current drive and that the radial distribution of the externally driven current complements the bootstrap current profile so that the total current profile satisfies any global requirements and is robust against MHD instabilities. However, the externally driven current is constrained by the acceptable level of the re-circulating power and the number of ports available for the external current drive. The formula for the bootstrap currently uses the following ITER physics guidelines [1, 2]:

$$I_{bs} / I_p = C_{bs} (\varepsilon^{1/2} \beta_{pa}) \tag{10}$$

$$C_{bs} = 1.32 - 0.235 \left(\frac{q_{95}}{q_0} \right) + 0.0185 \left(\frac{q_{95}}{q_0} \right)^2 \tag{11}$$

$$\beta_{pa} = \beta \left(B_0 \frac{2\pi \langle a \rangle}{\mu_0 I_p} \right)^2 \tag{12}$$

2.3 Technology Constraints

There are various technology constraints, such as the radial/vertical build, the ripple condition, critical current density in the superconducting coil, the startup & burn volt-sec capability, the stress limit, the divertor heat load limit, shield requirements, and the maximum TF field. These constraints will incorporate the prospects of the development of the relevant technology in the future.

Utilization of a strong magnetic field is very important because of its impact on plasma performance, and R&D activity is required to obtain a higher maximum magnetic field on a conductor than the current value of 13 T [4]. The constraints on the superconducting coils can limit the operating current density at various operating conditions.

The position and width of the components of the tokamak reactor, including the blankets, shields, central solenoid coils, and toroidal field coils, depend on the physics and technology constraints. The ripple requirement determines the location of the outer leg of the TF coil. Sufficient space for the blankets and the shields should be maintained to maximize the tritium breeding ratio and the energy multiplication factor. Shield thickness is also closely related to the neutron wall loading (fusion power).

If a plasma current ramp-up is provided with the magnetic flux of the central solenoid coils, it has to be larger than the required magnetic flux and this, in turn, restricts the position, size and current density of the coil. The required magnetic flux is expressed as

$$\Delta\Psi = L_p I_p + C_{Ejima} \mu_0 R_0 I_p \tag{13}$$

where L_p is the plasma inductance, $\mu 0$ is the vacuum permeability and C_{Ejima} is the Ejima coefficient [5].

A constraint imposed by the maximum tolerable divertor heat load has an impact on the machine size, plasma current and current drive power. The divertor heat load can be reduced by an impurity seeding of the edge plasma and the core by increasing the radiated power; otherwise, developments in divertor technology will reduce the penalty imposed by the divertor on the plasma performance. Developments in the technology for an increased tolerable heat load and physics to reduce the heat load and to improve the confinement are necessary. Thus far, a maximum peak power flux of 15 MW/m^2 would be permissible and the divertor plasma temperature would need to be reduced to below 20 eV to ensure that the erosion rate is acceptable.

The current drive power is required to sustain the plasma current; thus, it is important to develop efficient current drive systems that can run reliably in a steady state to reduce the resultant re-circulating power. The current drive power is limited due to the limited number of available ports and the necessity of a low circulating power.

2.4 Development of the System Code

In the system analysis code, the physics and technology constraints explained in the previous sections are modeled into the plant power balance equation. The system code finds the design parameters under the plasma physics and technology constraints or optimizes the design depending on the given figures of merits. The manner in which the systems code operates is such that n variables (normally physical parameters or device parameters) are found with given n constraints (physical or technology constraints), or a set of variables that optimize the given figure of merit (object function) is found. In the latter case, the number of variables can be larger than the number of constraints. The logical analysis flow is illustrated in Fig. 3. The main variables of the system code are plasma physics parameters such as the normalized beta value β_N, the confinement improvement factor H, the ratio of the density limit, the major radius, and the temperature. Engineering parameters include the maximum magnetic field on the

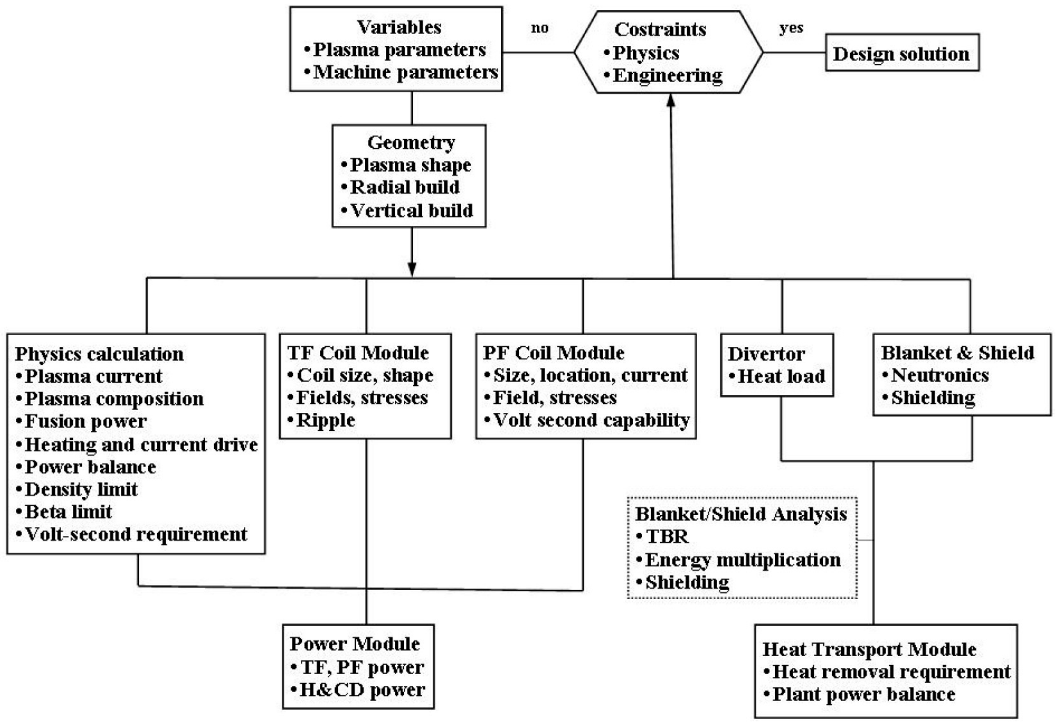

Fig. 3. Structure of the Tokamak System Analysis Code

TF coils, the current drive power, and the divertor heat load. These processes are iterated until solutions that satisfy the given constraints are found.

3. PERFORMANCE OF AN ITER-LIKE DEMO REACTOR

The tokamak reactor system analysis code was utilized to investigate the operational space for the DEMO reactor and to develop the concept of the DEMO reactor. For the DEMO reactor, widely accepted common requirements are that it must 1) demonstrate tritium self-sufficiency, 2) generate net electricity, and 3) demonstrate a steady-state operation.

To investigate the performance of the DEMO reactor, it was assumed that the plasma physics and technology in DEMO reactor to be improved compared to those adopted in ITER. The plasma parameters of the major radius, aspect ratio, plasma elongation, plasma triangularity, and edge safety factor were set to be identical to those of ITER. The plasma parameters that characterize the performance, i.e., the normalized beta value β_N, the confinement improvement factor for the H-mode H, and the ratio of the Borass density limit n/n_B were assumed to be improved beyond those of ITER, i.e., $\beta_N \geq 2.0$, $H \geq 1.0$ and $n/n_B \geq 1.0$, as these regimes must be achieved in a fusion reactor. For technology conditions, the maximum magnetic field of 13 T, thermal efficiency of 30% and current drive efficiency of 50% were considered as achievable in the near future.

For a blanket, a He-Cooled Molten Lithium (HCML) blanket was chosen. A previous study [6] showed that a total blanket and shield thickness of 2.5 m provides a tritium breeding ratio larger than 1.05. In addition, the neutron energy multiplication factor was calculated to be larger than 1.15.

For a steady state operation, the plasma current was driven by a combination of the bootstrap effect and a current drive by external heating. In this study, the plasma current ramp-up is assumed to be provided by the magnetic flux of the central solenoid coil, and the external current drive is provided by NBI and LHCD, which is the favored current drive scenario for the ITER in a steady state.

With these physics and technology assumptions, the operational space for the ITER-like DEMO reactor was investigated. In Fig. 4, the plasma density is assumed to be above the density limit, $n/n_B = 1.2$, and the divertor is protected from an excessive heat load by radiation through an impurity (Fe) seeding of the divertor and the main plasma. When Fe = 0.1 %, electric power generation of Pe = 500 MWe is possible in the space of $\beta_N > 4.5$ for

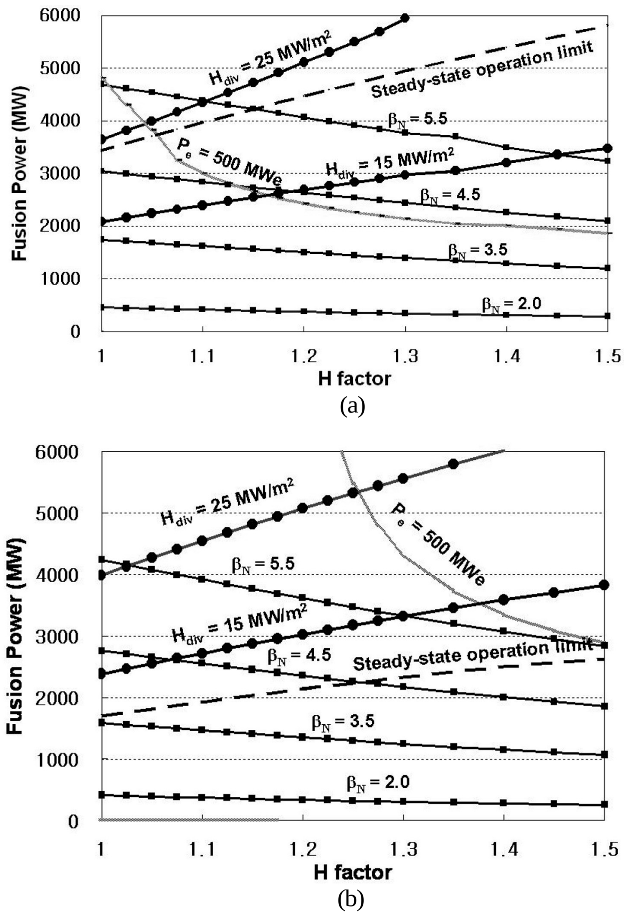

Fig. 4. Plasma Performance (a) When Fe= 0.1 %, and (b) When Fe = 0.2 %

H > 1.1. For a steady state operation, a high plasma beta of $\beta_N > 4.5$ is required. However, in this region, the divertor heat load H_{div} is well above 15 MW/m^2. If the impurity seeding is increased to Fe = 0.2 %, power loss by radiation increases and additional heating power is required to produce the same fusion power compared to the case with Fe = 0.1 %. For the same reason, the region with the lower plasma beta becomes accessible for the steady state operation. Thus, the higher plasma beta of $\beta_N > 5.5$ and good confinement of H > 1.3 are required for electric power generation of Pe = 500 MWe. Inside this region, operational space with a low beta value and a high confinement value is allowed for the divertor heat load H_{div} to be below 15 MW/m^2.

Thus, to access the operational space in which a steady state operation is possible and the operation space for higher electric power, the bootstrap current must be high enough and the current drive efficiency must be substantially higher than that expected in the ITER. The higher bootstrap current reduces the current drive power requirement from the auxiliary heating systems, which plays an important role in a reactor. In addition, both improved plasma physics and technology are required to handle a high heat load on the divertor to maintain the divertor within the engineering constraints.

4. CONCLUSION

A tokamak system analysis code that is necessary for developing the concepts of the DEMO reactor was developed in this study, and the necessary R&D areas in terms of physics and technology to realize the concepts were identified.

Using the developed system analysis code, an investigation of the operational space for an ITER-like DEMO reactor was made with the plasma size identical to that of ITER and with a limited extension of the plasma physics and technology adopted in the ITER. To access the operational space in which a steady state operation and a net electric power of Pe > 500 MWe are possible, a high β_N value and a high confinement value are required; the accessible space is determined mainly by the divertor heat load and steady-state current drive limit. To expand the operational space, better methods to handle the divertor heat load, a higher bootstrap current, and better current drive efficiency are required.

For further development of the reactor concept, a detailed analysis of the plasma performance and an engineering analysis are required.

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