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Coupling of Neutronic Analysis in Tokamak Reactor System Analysis

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1. Introduction

The reactor concept with small size for given performance is desirable in economic point of view. The design of blanket and shield play a key role in determining a size of a reactor since it has impact on the various reactor components. Blanket should produce enough tritium for tritium self-sufficiency and the shield should provide the sufficient protection of the super-conducting toroidal field coils (TFC); the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, the radiation damage to the copper stabilizer and the radiation dose absorbed by the Epoxy resin insulator.

For neutronic optimization of the blanket and the shield, the quantities such as the tritium breeding ratio (TBR), nuclear heating, radiation damage to the toroidal field coil have to be calculated and neutronic analysis need to be coupled in the system analysis. In most of the previous system studies, neutronic calculation and plasma analysis are performed separately, so blanket and shield size was determined independently from the reactor size. In this work, to account for the interrelation of blanket and shield with the other components of a reactor system, we coupled the system analysis with one-dimensional neutronic calculation to determine the reactor parameters in self-consistent manner.

2. Tokamak Reactor System Analysis Coupled with Neutron Transport Analysis

2.1 Physics model

In a system analysis, the main mathematical model to capture the physics and technologies are the plasma power balance equation which is represented as

$$P_{con} + P_{rad} = P_{OH} + P_{\alpha} + P_{CD} \qquad (1)$$

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, 2] is used

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_o is the major radius (m), a is the minor radius (m), and κ is the plasma elongation. In Eq. (2), H represents the confinement enhancement factor.

The plasma physics properties are expressed in a zero-dimensional model in the system analysis code. They impose limitation to the possible plasma performance through beta limit, plasma current limit imposed by a limit on the safety factor q at the edge, and the plasma density limit. For temperature and density profile, we assume a simple dependence on the minor radius of $[1-(r/a)^2]^{\alpha}$.

Steady-state operation requires that the plasma current be fully driven by the non-inductive current drive and that the bootstrap current. Total current profile has to satisfy any global requirements and has to be robust against MHD instabilities.

2.2 Radial build and engineering constraints

There are various engineering constraints, such as the radial/vertical build, the startup and burn volt-second capability, critical current density in the superconducting coil, the maximum TF field, the stress limit, the ripple condition, the divertor heat load limit and the shield requirements.. In this section we elaborate on the constraints that contribute to the radial build since they mainly determine the size of a reactor.

We consider the standard configuration where the blanket and shield are installed inside the vacuum vessel surrounding the plasma. Then the radial build of a reactor consists of central solenoid coils, toroidal field coils, shield, blanket and plasma. The radial position and width of these components should be determined by the physics and engineering constraints which they should satisfy.

Operation scenario has an impact on the design of the CS coil and it is a important design driver to determine the reactor size. 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

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

(3)

$$\Delta\Psi = L_p I_p + C_{Ejima}\mu_0 R_0 I_p \qquad (4)$$

where L_p is the plasma inductance, $\mu0$ is the vacuum permeability and C_{Ejima} is the Ejima coefficient.

There is a limit to the maximum magnetic field of CS coil and current density in superconducting conductor.

$$J_{CS} \le J_{CS,C} \qquad (5)$$

, where $J_{CS,C}$ is calculated depending on the operation scenario. Also the peak in-plane stress is limited to an appropriate allowable value.

Toroidal magnetic field and the current density at that field have an impact on the system design. Ampere's law relates the maximum toroidal magnetic field at the inner leg of the TF coil, B_{max} to the operating current density and the width of winding pack. The operating current density is limited by the critical current density of superconductor.

$$J_{TF} \le J_{TF,C} \qquad (6)$$

The constraint that in-plane stresses in the winding pack has to be within the allowable stress has a impact on the design of coil case. And the ripple requirement determines the location of the outer leg of the TF coil.

Sufficient space for the blankets should be maintained to maximize the tritium breeding ratio and the energy multiplication. Shield thickness is also closely related to the nuclear heating, radiation damage to the toroidal field coil. For estimate of these constraints, neutronic analysis are necessary.

2.3 Coupling with Neutron Transport Code

Neutronic calculations are carried out with the one dimensional radiation transport code, ANISN, and a transport group cross section library which consists of 30 neutron groups and based on JENDL-3.1. For the estimation of the local TBR, we have used the JENDL dosimetry file. ANISN is multi-group 1-D discrete ordinates transport code system with anisotropic scattering, and solves the Boltzmann transport equation for neutrons and gamma rays in slab, sphere, or cylinder geometry[#] and it has been used for the fission neutron design.

In the system code, main plasma physics parameters are the normalized beta value β_N, the confinement improvement factor *H,* the ratio of the density limit, the major radius, the aspect ratio, etc. Engineering parameters include the maximum magnetic field on the TF coils, the current drive power, the divertor heat load, etc. To find design parameters, design parameters and constraints have to be selected depending on the figures of merit which needs to be optimized. Starting from the initial guess for the design parameters, all the physical and engineering parameters of a reactor are calculated. From the calculated fusion power, ANISN code is called to calculate the tritium breeding ratio and the radiation shielding effect of the blanket and shield, and then needed dimension for blanket and shield are calculated. This, in turn, determines the magnetic field at magnetic axis and calculate the fusion power. The system code iterates the design parameters until they satisfy all the physical and engineering constraints.

3. Conclusion

For self-consistent calculation of the physical and engineering constraints which relate the various components of a tokamak reactor, the system analysis code was coupled with the one dimensional radiation transport code, ANISN. The coupled system analysis code can provide a comprehensive optimization study of a tokamak reactor with various blanket and shield.concepts.

REFERENCES

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