PLASMA-SAFETY ASSESSMENT MODEL AND SAFETY ANALYSES OF ITER

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Abstract

A plasma-safety assessment model has been provided on the basis of the plasma physics database of the International Thermonuclear Experimental Reactor (ITER) to analyze events including plasma behavior. The model was implemented in a safety analysis code (SAFALY), which consists of a 0-D dynamic plasma model and a 1-D thermal behavior model of the in-vessel components. Unusual plasma events of ITER, e.g., overfueling, were calculated using the code and plasma burning is found to be self-bounded by operation limits or passively shut down due to impurity ingress from overheated divertor targets. Sudden transition of divertor plasma might lead to failure of the divertor target because of a sharp increase of the heat flux. However, the effects of the aggravating failure can be safely handled by the confinement boundaries.

1. INTRODUCTION

Fusion has attractive inherent safety and environmental characteristics, which are one of the motivations for developing fusion reactors. ITER has an important role in demonstrating realization of these characteristics. Fusion power dose not increase like chain reaction but is self-bounded by physical limits like the beta-limit. Fusion reactions can be easily terminated if necessary. On the other hand, the next experimental fusion machine already has a large fusion power which is compatible with that for a commercial power plant. Therefore, it is still necessary to address various plasma physics phenomena in safety analysis to demonstrate the fusion reactor's safety and environmental potential. This study presents a plasma-safety assessment model to simulate the plasma transients and discusses safety analyses of ITER.

2. MODEL AND CODE DEVELOPMENT

2.1 Plasma-safety assessment model

The basic plasma physics database has been provided by the ITER-JCT, the Home Teams, and the ITER Physics Expert Groups from data obtained in present tokamak experiments. The plasma-safety assessment model was developed from reasonable extrapolations of this database and principles of tokamak theory [1]. The model specifies an optimistic operation area for the plasma. In other words, the conservative operation limits provide severe operation conditions for the in-vessel components during abnormal plasma events (e.g., overpower). In the model, some physical limits, e.g. beta-limit (Troyon limit) and density limits (Borrass (BR) and Greenwald (GW) limits) and threshold power of H-L confinement mode transition, etc. for safety analyses were set on the basis of the ITER plasma physics reference so that the plasma could be optimistically operated.

The occurrence of disruptions can be assumed due to the beta-limit and density limits. The operational limits are classified as the upper bounds of the ITER class plasma. The beta-limit is set higher than the reference data by about 50 % and the density limits, which are an "AND" condition of BR limit and GW limit, are about double the reference judging from tokamak experimental data.

exceeds the plasma heating in each region (in other words, the energy balance collapses), the plasma is regarded as being terminated by a disruption.

The 0.85xITER93-H Law (baseline law) and ITER89-P Law have been taken as the energy confinement scaling laws. The H-to-L and L-to-H mode threshold powers are calculated conservatively on the basis of updated physics data, which means that the threshold power specified in the safety guideline is assumed to be lower than that in the reference data so as to keep the H-mode as long as possible.

A detached/attached state transition model of the divertor plasma is provided to simulate divertor plasma transition [2]. An operational table based on the present divertor plasma physics is developed, which gives the detached/attached operational area. The model can judge the divertor plasma state with the power flowing into the divertor region, the electron density and the impurity fraction in the divertor plasma. The edge and divertor radiation power partitioning and scaling are also provided to calculate the divertor heat flux linked with the main plasma.

Besides the above models and data, the standard fusion reaction parameter, <_v>, nominal definitions of plasma parameters, impurity transport probability from walls to the main plasma and conservative transport time of impurity, etc. are included in the plasma-safety assessment model.

2.2 Plasma-safety analysis code

The plasma-safety assessment model has been installed in a safety analysis code (SAFALY) consisting of the plasma dynamics model and thermal behavior model of the in-vessel components [3]. A zero-point plasma dynamics model, which takes account of energy and particle conservation, would be enough to simulate the thermal plasma transients. The energy conservation treats ions and electrons separately and the particle conservation includes fuel ions, alpha-particles and impurity ions. The code can determine the steady state parameters on the basis of ion temperature, fusion power and Q-value (fusion power/auxiliary power) given as input. Uncertainties arising from the plasma physics can be covered by a parameter survey using the adaptability of the code. SAFALY can also treat impurity transport from plasma facing components (PFCs) by a simplified model with a transport probability from the wall into the main plasma through the scrape-off layer (SOL) and a time delay due to the transport (typically set to be equal to the energy confinement time).

Verification studies between the results by SAFALY and by a 1.5-D plasma transport code (PRETOR) have been carried out [4]. The PRETOR code was used for plasma performance assessments for many ITER design related problems. Several plasma abnormal transients were considered for the test cases. It was confirmed that SAFALY is capable of reproducing the results of the more sophisticated PRETOR code, and for all significant parameters, SAFALY gives a more conservative prediction as a safety analysis code.

For the thermal behavior of the in-vessel components, a one-dimensional time-dependent model is adopted in the radial direction of the structures, emphasizing the adaptability. The structures in the code are the PFCs, i.e., first wall, divertor, and blanket. To obtain the temperature distribution in the poloidal direction, the structures can be divided into 20 calculation regions and radiation between the surface of each region can be also considered. In this sense, the code can carry out a 1.5-D heat transfer analysis. Furthermore, melting and evaporation (sublimation) of PFC material can be taken into account. Each coolant channel is treated as a constant convective boundary during the steady state, which can simulate the changes of the heat transfer coefficients and temperatures during hydraulic accidents using other data.

In comparisons with results by the MELCOR code and with JCT analytical results (the MELCOR code was used for many safety analyses of the hydraulic accidents of ITER), the temperature distribution in the first wall/blanket in ITER obtained by SAFALY shows good agreement with a difference of 10 % [5,6]. The thermal characteristics are calculated by coupling with from PFCs into the plasma can be considered.

3. SAFETY ANALYSES

Some postulated initiating events leading to overpower in ITER were selected, and the plasma and in-vessel components behaviors were calculated on the assumption of a combined failure of plasma control and machine interlock [7]. The initial fusion power was assumed as 1.65 GW considering deviation of fusion power control by + 10%.

Figure 1 shows the transient behaviors in case of overfueling by a factor of 2.2. The factor was found by a pre-parameter study to lead to the maximum fusion power. After overfueling, the fusion power reaches the maximum of 3.1 GW at 24 s. During the overpower, the heat flux on the divertor target increases to 17 MW/m^2 under the detached state before 20 s. Just after that, the divertor plasma changes to the attached state and the heat flux jumps to 31 MW/m^2. The surface temperature of the target plate increases to 3000¡C. Then, sublimated impurities (carbon) abruptly enter the main plasma and plasma burning passively terminates when the density limits are exceeded.

In the event of a sudden auxiliary heating of 100 MW (Fig. 2), the divertor plasma transits to the attached state just after injection. After that, plasma burning terminates due to impurity ingress from the target plate at 13 s due to collapse of energy balance in the edge plasma. When the plasma confinement is improved by a factor of 2 (Fig. 3), fusion power reaches 3.3 GW just before the beta-limit. The divertor transits into the attached state at 1 s and plasma burning terminates at 3.5 s due to the beta-limit. The parameter scan shows that the maximum fusion power almost saturates at more than the improvement factor provided by the beta limit.

4. CONCLUSION

The plasma-safety assessment model was provided for safety assessment of fusion reactors. In the calculation, optimistic density and beta limits, threshold power of the H-L confinement mode and divertor plasma state transition model, etc., were set so that a conservative result was given for safety assessment. The model was implemented in the SAFALY code which couples the plasma dynamics mode to the thermal behavior model of the components and some plasma events for ITER were calculated. It was found that plasma burning safely self-terminated by ingress of evaporated impurities from PFCs or by physical limits. Transition to the attached divertor state might lead to dry out of the coolant for high heat flux components. This would lead to coolant pipe damage and in-vessel coolant leaks. However, another safety analyses of ITER show that effects of aggravating failure of the divertor can be safely handled by the confinement boundaries, the vacuum vessel and its pressure suppression system [8].

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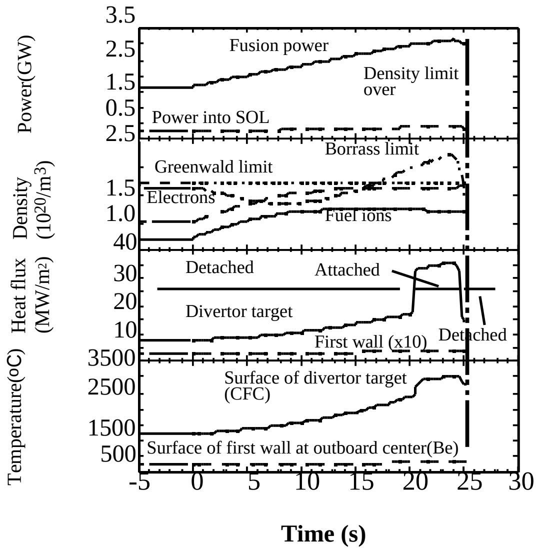

FIG.1. Time evolution of plasma parameters and the in-vessel component temperatures of ITER for an increase of fueling rate by a factor of 2.

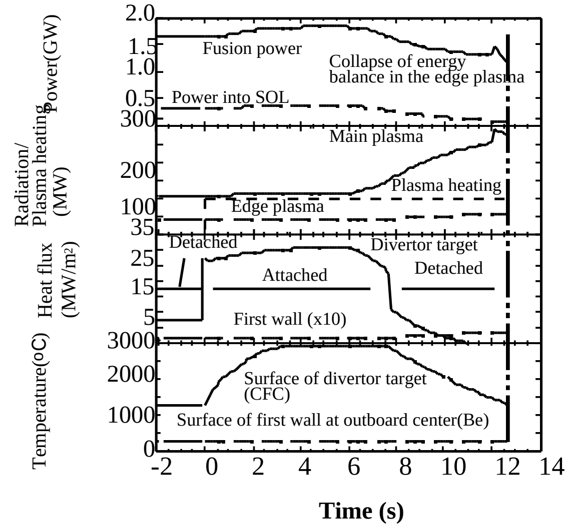

FIG.2. Time evolution of plasma parameters and the in-vessel component temperatures of ITER for a sudden increase of injection auxiliary heating of 100 MW.

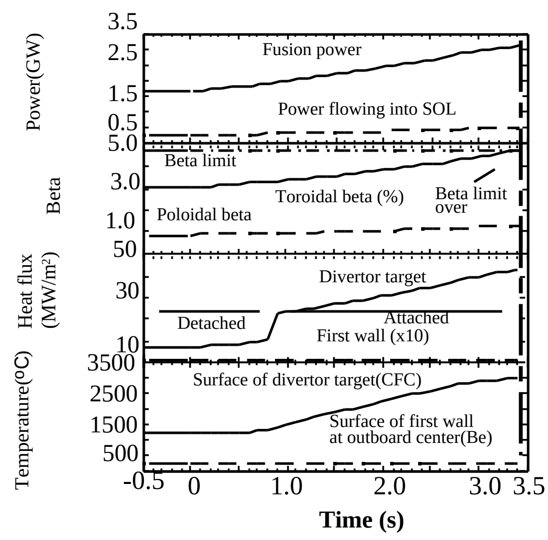

FIG.3 Time evolution of plasma parameters and the in-vessel component temperatures of ITER when plasma confinement is improved by a factor of 2.