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Simulation and Analysis of the Hybrid Operating Mode in ITER

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***Abstract*—The hybrid operating mode in ITER is examined with 0D systems analysis, 1.5D discharge scenario simulations using TSC and TRANSP, and the ideal MHD stability is discussed. The hybrid mode has the potential to provide very long pulses and significant neutron fluence if the physics regime can be produced in ITER. This paper reports progress in establishing the physics basis and engineering limitation for the hybrid mode in ITER.**

INTRODUCTION

The International Thermonuclear Experimental Reactor (ITER) project has identified three primary operating modes for demonstrating controlled burning plasmas, the ELMy H-mode, the Hybrid mode, and the Steady State Mode[1]. These modes of operation are motivated by experiments on existing tokamaks that demonstrate their potential for good performance. The reference operating mode is the ELMy H-mode, with I_P = 15 MA, B_T = 5.3 T, R = 6.2 m, a = 2.0 m, κ_x = 1.85, δ_x = 0.5, P_{alpha} = 80 MW, and P_{aux} = 40 MW, obtaining a fusion gain (Q = P_{fusion}/P_{aux}) of 10. The hybrid mode has the same geometry and field, but operates at lower plasma current, 12 MA. This leads to higher safety factor and less loop voltage to drive inductive current. Present experiments[2,3] on the hybrid configuration, which are heated by neutral beam injection (NBI), show that the plasma has higher energy confinement than the standard ELMy H-mode and can operate near the no wall n=1 β limit ($\beta_N \approx 3$) without neo-classical tearing modes (NTM) degrading its performance. This results in higher bootstrap current which further reduces the loop voltage. The hybrid mode obtains roughly 50% non-inductive current. The steady state (or advanced tokamak) mode has an even lower plasma current, 9 MA, and a slightly smaller minor radius, 1.85 m, with stronger shaping κ_x = 2.0, δ_x = 0.5. For these the non-inductive current is 100% in flattop, while inductive current drive would be used in the current rampup. The safety factor is higher thoughout the plasma, above 1.5-2.0 everywhere. Although steady state configurations can be found with β_N near the no wall n=1 β limit, it is desired to demonstrate sustained operation above this limit, to approximately β_N $\approx$ 4-4.5, with resistive wall mode feedback. It can be seen that the hybrid operating mode has physics features somewhat between the ELMy H-mode and steady state regimes, and can provide very long pulse lengths, giving it the potential of providing high neutron fluence (neutron wall load × flattop time) for nuclear testing. Systems Analysis, 1.5D Discharge simulations with the Tokamak Simulation Code (TSC)[4] and TRANSP[5], and MHD stability with JSOLVER/BALMSC/PEST2[6-8] of the hybrid operating mode in ITER will be discussed in the following.

I. SYSTEMS ANALYSIS OF HYBRID PLASMAS

A zero-dimensional systems code was developed for use in the ITER study. The analysis used for hybrid operating point calculations incorporates the plasma power and particle balance, Bosch-Hale fusion reactivity[9], Post-Jensen coronal equilibrium radiation model[10], Albajar cyclotron radiation model[11], Hirshman and Neilson flux consumption formulation[12], in addition to several other global parameter relations. In particular, the ITER98(y,2) scaling is assumed for the global energy confinement time. For the present application to hybrid plasmas, the major and minor radius, elongation, triangularity and aspect ratio are fixed.. An expression for the bootstrap current fraction is included and external current drive is included. Heating and current drive are provided by Negative Ion NBI (NNBI), at 1 MeV, and ICRF minority heating on He3. The current drive efficiency used in these scans is 0.3 A/W-m2 for NNBI, which is determined from TRANSP calculations. There is 33 MW of NNBI power and 20 MW of ICRF power available. The volt-second capability is 300 V-s with 10 V-s reserved for breakdown, an Ejima coefficient of 0.45 for the current rampup, and an li(1) of 0.8 for the flattop plasma, all derived from 1.5D TSC simulations. A large number of plasma configurations are generated by varying the I_P from 11.0 to 13.0 MA, β_N from 1.5 to 3.0, the ratio of line average density to Greenwald density n/n_{Gr} ($n_{Gr} = I_P/\pi a^2$) from 0.4 to 1.0, fusion gain from 3.0 to 12.0, the density peak to volume average from 1.05 to 1.25, temperature peak to volume average from 1.5 to 2.5, beryllium impurity fraction from 1 to 3%, carbon impurity fraction from 0 to 2%, and the argon impurity fraction from 0 to 0.2%. The toroidal field and ratio of τ_{He}^*/τ_E = 5 were held fixed. Parabolic temperature and density profiles are used, so that peak to volume average values are obtained by prescribing an exponent and an edge value.

The resulting physics operating points are further constrained by engineering limitations, such as the fusion power/pulse length determined by the heat rejection system, the maximum fusion power determined by the cryoplant, the maximum peak heat flux to the divertor, volt-second capability of the PF coils, first wall maximum surface heat flux, and installed auxiliary powers for heating and current drive.

Results show that the fusion power/pulse length limitation of the existing ITER design is the most limiting to the operating space for the hybrid operating mode. Although the PF coils can provide very long pulses, > 3000 s, for the loop voltages expected in the Hybrid, the flattops are severely constrained for fusion powers above approximately 350 MW.

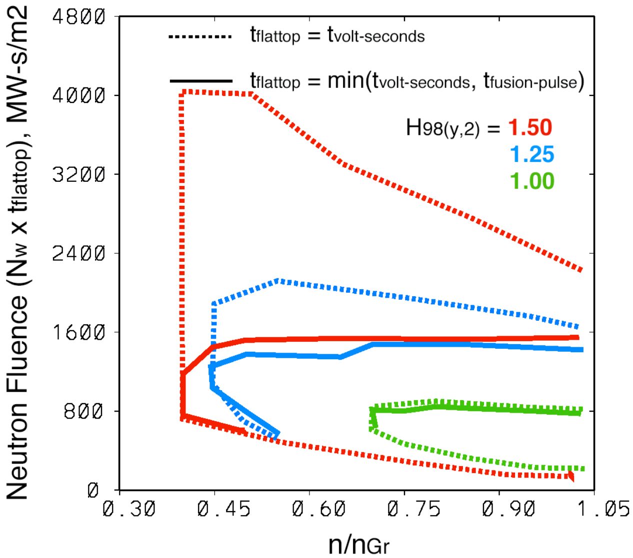

FIGURE 1. The neutron fluence within a discharge versus the density divided by the Greenwald density from a large plasma parameter scan, showing the influence of the fusion power/pulse length limitation of the existing ITER design on the operating space of the ITER hybrid operating mode. The very long pulse lengths available from the PF coils volt-second capability and higher β_N values can not be accessed.

The existing heat rejection system can provide 3000 s flattops for P_{fusion} = 350 MW, 400 s for 500 MW, and about 150 s for 700 MW. The hybrid only reaches β_N values of about 2.0 at P_{fusion} = 325 MW, which may not be consistent with the physics of this operating mode seen on existing experiments. Therefore, the heat rejection system needs to be upgraded in order for the hybrid plasmas to access the long pulses provided by the volt-second capability. Shown in Fig. 1 is the neutron fluence as a function of the ratio n/n_{Gr}, with $H_{98(y,2)}$ values of 1.0, 1.25, and 1.5, if the flattop is determined by the available volt-seconds from the PF coils, or the pulse length allowed at the given fusion power. The reduction in available operating space to maximize neutron fluence within a discharge is clear. The divertor heating is the next most significant constraint, and is determined by the power radiated from the core plasma (bremsstrahlung, cyclotron, and line), the power radiated in the region of the divertor, and transients in the power leaving the plasma (ELMs). Although the maximum peak heat flux in the divertor can reach > 20 MW/m^2, a nominal value of 5-10 MW/m^2 is preferred. Sufficient impurities in the plasma are required to reach these conditions, including the intrinsic impurities like Be and C, and intentional ones like Ar. Finally, in order to provide useful nuclear material testing, the ratio of $t_{flattop}/(t_{flattop} + t_{dwell})$, where t_{dwell} is the total time between discharge flattops, must be sufficiently high for extended periods of time (order of months to a year). A recent paper[13] indicated that this ratio is 25%, regardless of the operating mode. The cryoplant for the superconducting PF and TF coils is what is limiting this, and will preclude significant neutron fluences without an upgrade.

II. 1.5D SIMULATIONS OF ITER HYBRID OPERATING MODE

TSC and TRANSP are used for the 1.5D modeling of ITER. TSC is a predictive free-boundary evolution code. It solves the MHD-Maxwell equations on an axisymmetric 2D grid, and 1D transport equations for density, temperature, and current density given the various transport coefficients and source descriptions. TRANSP is an interpretive code, usually applied to experiments, expecting the temperature, density, current density (or q profile), and equilibrium geometry to be given. It solves for the transport coefficients, applying the necessary source models. The source models and fast particle treatment in TRANSP are considered its best attributes and the free-boundary capability in TSC, including PF coils, structures, and feedback systems, is considered its best attribute. These are combined by creating an iteration between the codes, refining the heating and current drive sources and the discharge scenario.

For the simulations reported here, the density profile and magnitude are prescribed, while the energy transport is simulated with the GLF23[14] core transport model, which is calculated both with ExB shear stabilization of the turbulence and without this ExB shear included. The plasma rotation speed used in the ExB shear analysis comes from the TRANSP calculation. Amended to this model is a prescribed pedestal location and magnitude, which is varied to demonstrate the dependence of fusion performance. The impurity assumption is 2% Be, 2% C, and 0.12% Ar. The plasma is grown from a limited starting point on the outboard limiter, and it is found that early heating of ≤ 10 MW is required to keep q(0) above one, which is within the limiter capability. The plasma current, radial position, vertical position, and shape are feedback controlled. In addition, the ICRF power level is in a stored energy feedback loop, while the NNBI power is fixed at the maximum to provide the most current drive possible. In TRANSP the NNBI calculation uses the Monte Carlo orbit following method, NUBEAM. These calculations indicate that the NNBI system, at 33 MW and 1 MeV particle energy, can drive approximately 1.4-3.0 MA of current depending on the precise values of Z_{eff}, density, temperature, and on/off-axis steering, The ICRF calculation is done with the upgraded SPRUCE reduced order full wave analysis in combination with a Fokker Planck calculation of the distribution function. A 2% (of DT density) He3 fraction is assumed, and the frequency is 52.5 MHz. For a typical hybrid case in ITER, 20 MW of injected power delivers 13.8 MW to He3, 3.9 MW to other ions, and 2.3 MW to electrons, with the He3 reaching energies of up to 120 keV.

Initial discharge simulations showed that hybrid plasmas could be established and sustained with fusion powers ≤ 350 MW, which would allow them to access the very long pulse capability of the PF coils since the existing ITER designed heat rejection system can provide 3000 s pulses for these fusion powers. A typical plasma in this class had a peak density of 0.77×10^{20} /m^3, peak temperatures of 23 keV, n/n_{Gr} = 0.8, $H_{98(y,2)}$ = 1.3, non-inductive current fraction of 0.45, li(1) of 0.8, with Z_{eff} = 1.3-2.2. However, the pedestal temperature required for these plasmas was about 7.5 keV, and β_N only reached 2.0. Since tokamak experiments indicate that the hybrid configuration needs to operate close to the no wall n=1 β limit to avoid confinement degradation from NTMs these plasmas may not satisfy this requirement.

Higher β_N ($\approx$ 3) hybrid plasmas were generated to determine their requirements. A typical hybrid plasma from this class had a peak density of 0.93×10^{20} /m^3, peak temperatures of about 30 keV, n/n_{Gr} = 0.93, $H_{98(y,2)}$ = 1.6, a non-inductive current fraction of 0.65, li(1) of 0.77, with Z_{eff} = 2.2. Here again, the required temperature pedestal was 9.5-10 keV, β_N reached 3, and the fusion power was 500 MW. These high pedestal temperatures are above those predicted by the pedestal database scaling[15], which would predict a value of roughly 5 keV. Another complication of these high pedestals is weaker line radiation, since the volume where the Ar radiation would maximize, near the plasma edge, is strongly reduced. This makes the divertor solutions for the hybrid unacceptable at the present time.

Since the temperature pedestal plays such an important role in the overall energy confinement and fusion performance, a scan is done as a function of auxiliary power, using GLF23 core transport both with and without ExB shear stabilization. The plasma toroidal rotation speed is determined from TRANSP assuming the momentum diffusivity is equal to the ion thermal diffusivity. Shown in Fig. 2 is the fusion gain, Q, as a function of the pedestal temperature, for P_{aux} = 38, 43, and 53 MW. It can be seen that the plasma rotation speed is clearly too low to improve the energy confinement in ITER hybrids, although this is observed to be a significant effect on present tokamak hybrids. Lower auxiliary powers are giving similar fusion gains at lower pedestal temperatures, however, the β_N values are lower, the sawtooth radii are larger, and the non-inductive current fractions are lower.

These results raise the important question of how different ITER's physics regime is likely to be than present tokamaks, and this should be considered when establishing the physics basis for an ITER hybrid. Present tokamak hybrid plasmas have five important characteristics; high toroidal rotation from NBI, $T_i > T_e$, some degree of density peaking ($n(0)/\langle n\rangle \geq 1.25$, a suppressed or no sawtooth, and benign NTMs (in particular, the 3/2). The first three provide for enhanced energy confinement, but are likely to be missing in ITER. The fourth relies, at least in DIII-D, on a tearing mode that appears to stop the current profile from diffusing toward the core and becoming more peaked. The fifth relies on operating at sufficiently high β_N that the NTM is significantly weakened, and appears similar to the FIR-NTM regime identified on ASDEX-U and JET[16]. However, the resistive MHD regime in ITER is likely to be different as well, since many of the dimensionless plasma parameters that influence this physics will be different. Therefore, projecting the performance of the ITER hybrid based directly on existing hybrid experiments may be optimistic.

III. VERIFICATION OF GLF23 TRANSPORT MODEL ON EXPERIMENTS

In order to project energy transport in ITER, the trend has been to rely on theoretically based turbulence models, of which GLF23 is an example[14]. The model is based on fits to gyro-fluid, and more recently gyro-kinetic, simulations of plasmas with parameters that bracket those expected in a wide range of tokamak plasmas at standard aspect ratios. This allows the model to produce a thermal diffusivity which can be used in a predictive transport code like TSC. Considerable effort has gone into improving this model, with the latest version available from the National Transport Code Collaboration (NTCC) library. It is necessary to apply the model to experimental discharges produced in tokamaks to demonstrate its capability to reproduce the transport observed. As part of the study of the ITER hybrid, a TSC simulation of a DIII-D hybrid discharge, shot 104276, was produced using a new algorithm[17] allowing a predictive code to reproduce experimental temperature profiles. In this simulation, the density profile and toroidal rotation profile is given by experimental data, and the NBI heating deposition profile is given by a TRANSP run of the discharge. The ExB shear stabilization associated with the plasma rotation is included. Once the discharge simulation is established, it is rerun with GLF23 to provide thermal diffusivities, and the resulting temperature profiles are compared. Comparisons of the peak electron and ion temperatures, and the profiles in the flattop phase of the discharge show that the GLF23 model produces a reasonable fit to the experimental data when the ExB shear stabilization is included. However, it completely misses an ion internal transport barrier (ITB) that exists in the earlier times when the plasma is in L-mode. ITBs in the ion channel are a common feature of many DIII-D advanced tokamak plasmas, and the transport physics of this phase is included in the GLF23 model. This part of the discharge is often difficult to model due to ramping plasma current and density, and varying NB power losses. Work is continuing to examine the GLF23 model in the TSC code to guarantee that it is implemented properly and will produce consistent results. This issue is particularly critical with GLF23 since the model can suddenly suppress or enhance the turbulence depending on the temperature gradients relative to critical gradients. Several parameters in the TSC code are being examined, including the radial mesh, time step, maximum allowable thermal diffusivity, time and space relaxation procedures, treatment of the nonlinear thermal diffusivities, and the numerical algorithm used to integrate the transport equations. It is critical to use theoretical transport models that are verified on experiments as well as possible, since we are projecting to ITER which has a different physics regime in terms of dimensionless parameters, such as gyro-radius normalized to the plasma minor radius, collisionality, etc.

IV. IDEAL MHD STABILITY OF ITER HYBRID

Experimental hybrid discharges on present tokamaks indicate that this plasma configuration exists in a β_N window, below the n = 1 no wall kink limit and above a not so well defined NTM limit. Below the NTM limit the 3/2 mode will significantly degrade energy confinement, and above the n=1 no wall kink limit, either an n=1 RWM or a 2/1 NTM will appear and result in a discharge termination. An advantage of the hybrid configuration is that it should not require feedback control of the RWM as in the steady state (advanced tokamak) operating mode. Ideal MHD stability of plasma hybrid discharges produced in TSC at β_N = 3 show that they are stable to the n=1 external kink mode without a conducting wall. However, the on-axis safety factor is typically below one, so these plasmas are susceptible to an internal n=1 or sawtooth instability. The JSOLVER fixed boundary equilibrium code is being used to produce model ITER hybrid configurations to study the MHD behavior as β_N is varied, using NNBI current profiles from TRANSP and self-consistent bootstrap current profiles. In particular, the sawtooth radius is found to increase as the β_N drops. In addition, the Porcelli sawtooth model[18] is being used in the TSC discharge simulation to determine if the fast particles can fully stabilize the sawtooth in ITER hybrid discharges.

V. DISCUSSION

A study of the ITER hybrid operating mode has begun to establish the physics basis and operating space within ITER's engineering constraints. 0D analysis is being used to show the influence of ITER's present design on the hybrid mode, in particular, on how the goals of long pulses and high neutron fluence are affected. Upgrades to the heat rejection and cryoplant systems appears to be required to take full advantage of the hybrid plasma configuration in ITER. 1.5D calculations with TSC and TRANSP are used to identify attractive discharge scenarios and more detailed physics characteristics for the hybrid plasmas. Using the GLF23 core transport model, the hybrid plasmas appear to require higher temperature pedestals than one would expect from the pedestal database, and needs to be at high n/n_{Gr} in order to reach a β_N of 3.0. These results indicate that some caution should be used when projecting to ITER hybrids directly from existing tokamak hybrid discharges. Efforts to verify the GLF23 model in TSC on a DIII-D hybrid discharge show reasonable agreement, and are continuing. The ITER hybrid plasmas produced in TSC simulations are found stable to the n=1 external kink without a conducting wall, and work to examine the sawtooth instability is continuing.

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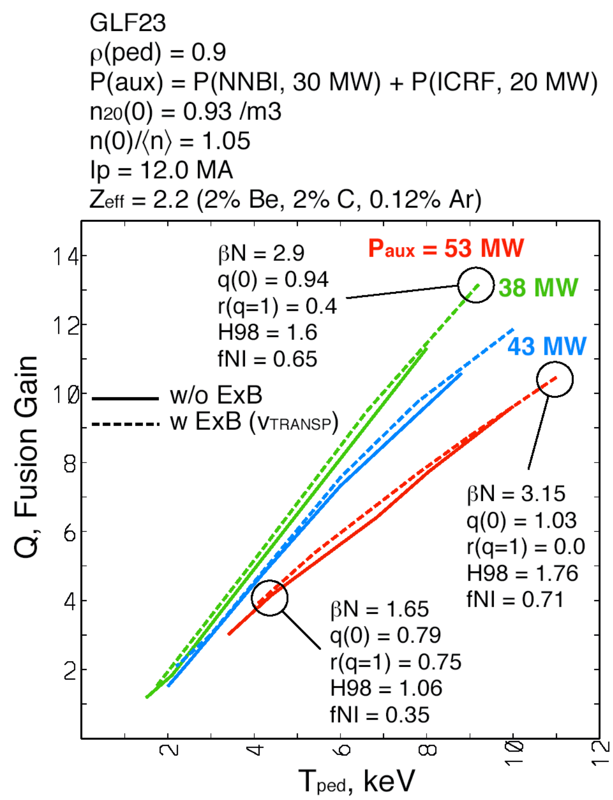

FIGURE 2. The fusion gain versus the pedestal temperature for the ITER hybrid scenario simulations, for 3 different injected auxiliary powers. The high pedestal temperatures are required to reach β_N values of 3, and the predicted plasma rotation in ITER does not improve the energy transport significantly.

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