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REFERENCES

1. W. F. Weldon and H. H. Woodson, Texas Atomic Energy Research Foundation Report No. 38 (1984).
2. B. Coppi, Comm. Plasma Phys. Cont. Fusion **3**, 47 (1977).
3. R. Carrera, E. Montalvo and M. N. Rosenbluth, Bull. Am. Phys. Soc. **31**, 1565 (1986).
4. M. N. Rosenbluth, W. F. Weldon and H. H. Weldon, Basic Design Report for the Fusion Ignition Experiment (IGNITEX). Preliminary proposal to the U.S. Department of Energy (March, 1987).

PHYSICS CONSIDERATIONS FOR THE COMPACT IGNITION TOKAMAK

R. Parker,[1] D. Post,[2] G. Bateman,[2] M. Bell,[2] P. Colestock,[2] J. Helton,[3] W. Houlberg,[4] M. Hughes,[5] D. Ignat,[2] S. Jardin,[2] J. Johnson,[2] S. Kaye,[2] C. Kieras-Phillips,[2] L-P. Ku,[2], G. Kuo-Petravic,[2] B. Lipschultz,[1] J. Manickam,[2] M. Peng,[6] M. Petravic,[2] M. Phillips,[5] M. Porkolab,[1] N. Pomphrey,[2] J. Ramos,[1] J. Schmidt,[2] D. Sigmar,[1] R. Stambaugh,[3] D. Stotler,[2] D. Strickler,[6] A. Todd,[5] N. Uckan,[4] J. Wesley,[3] K. Young[2]

Introduction. As presently conceived, the world's fusion program will enter during the next decade into the burning plasma phase, beginning with Q $\lesssim$ 1 experiments in TFTR and JET and culminating with operation of an International Tokamak Experimental Reactor (ITER) devoted to study of the engineering problems associated with a long-pulse, ignited (or near-ignited) tokamak. The Compact Ignition Tokamak (CIT), scheduled for operation near the middle of the decade, is viewed as a bridge between these initiatives and has a research program focused on resolution of alpha particle physics issues which occur on the timescale $\tau_{pulse} \sim 10\tau_E$. The early experience with ignited plasmas gained in CIT should be extremely useful to an ITER. In addition to providing early confirmation of the design assumptions, the CIT will provide a basis for detailed design decisions which can be made after the main parameters are set, for example detailed design of the divertor and the auxiliary heating and fueling subsystems. Equally important, the CIT experience should be valuable in developing ITER operating scenarios during the phases of startup, burn-initiation and control, and shutdown.

Aside from the programmatic role just described, CIT will permit study of a number of important and intriguing physics issues associated with collective phenomena which can occur in the presence of a sufficiently intense and energetic α-component. Some of the more important of these include:

- Effect of $P_\alpha >> P_{aux}$ on the plasma profiles and resultant global energy confinement, including sustainability of the H-mode and effect on χ_i through excitation of the η_i-mode;
- Effect of the α-power on the sawtooth including effect on τ_E and also on MHD stability through modification of $p(\psi)$ and $q(\psi)$ profiles /1,2/;
- Effect of the α-precession resonance on destabilization of n=1 internal kinks (fishbones) or high n ballooning modes /3,4,5/;
- Effect of super Alfvénic ($v_\alpha > v_A$) alpha particles on global or kinetic Alfvén eigenmodes and the possibility of anomalous spatial α-diffusion /6,7/;
- Effect of fast-α losses (due to ripple, sawteeth, fishbones, low mode-number tearing modes, etc.) on central impurity accumulation and ensuing fuel dilution.

The last point includes the effect of self-consistent changes in the ambipolar electric field due to all α-losses thereby classifying it as a collective phenomenon even if α-driven instabilities are not directly involved.

Machine Description. The CIT design concept is motivated by the potential of combining the good confinement properties of compact, high-field tokamaks with the additional advantages

[1] Massachusetts Institute of Technology, Cambridge, MA
[2] Princeton Plasma Physics Laboratory, Princeton, NJ
[3] GA Technologies Inc., San Diego, CA
[4] Oak Ridge National Laboratory, Oak Ridge, TN
[5] Grumman Aerospace Corporation, Princeton, NJ
[6] Fusion Engineering Design Center, Oak Ridge, TN

which accrue from shaping, *e.g.*, improvement in confinement (H-mode) in the auxiliary heating regime. A divertor has been chosen as the baseline configuration not only as a means of confinement improvement but also as being more prototypical of future devices. This, together with the desire for longer pulse length and more conservatism in the engineering approach have caused a general growth in size relative to the 1 m major radius typical of early Ignitor designs.

The design parameters given in Table I reflect compromises which have been made between the conflicting requirements of fixed project cost and the objective of reaching ignition with substantial margin relative to both confinement and stability limits. The field of 10 T is the maximum which can be produced using power and energy from sources available at the PPPL site; an upgrade to 12 T is feasible by installing additional power and also adding an external structure which limits the vertical separating stress in the inner leg of the toroidal field (TF) magnet.

The tradeoffs involved in locating the main PF coils inside vs outside of the TF have been carefully examined. Locating the PF coils internal to the TF brings them closer to the plasma and permits a broader class of equilibria to be obtained. This configuration is also efficient in total consumption of PF energy, however the required increase in TF energy tends to offset this gain. Although somewhat enhanced TF performance is possible with internal PF coils, the engineering risks increase considerably with this choice as the PF coils are then trapped and machine assembly becomes more complex. For these reasons, an external PF system has been chosen, supplemented by several internal coils, each capable of carrying up to about 0.5 MA. These coils will be used for fine-tuning the plasma shape, control of the divertor heat loads and stabilization of the axisymmetric instability.

TABLE I. CIT Machine Parameters (5/87)

R	1.75 m	P_α	$\leq$ 60 MW
a	0.55 m	$q_\psi(a)$	3 (3.6)
κ	$\leq$ 2.0	$\beta_T = 3I/aB$	5% (4%)
δ	0.4	TF Flat Top	5s
B	10 T (12 T)	Φ	41 v-sec
I	9 MA		

Confinement. Within the context of present-day understanding of confinement and operational limits, CIT is being designed with a substantial ignition margin thus permitting a wide variation in operating parameters in order to comprehensively study those collective phenomena which arise due to the presence of the energetic α-component. Using a "neo-Alcator" scaling of $\tau_e = .07naR^2q_{cyl}$ and a Murakami-Hugill scaling for the density limit of $< n > = 1.5\ B/Rq_{cyl}$, (where $< >$ implies a volume average), the ignition margin, P_α/P_{loss}, is proportional to B^2a at fixed T and q. Many other confinement scalings, including Kaye-Goldston "L-mode" scaling /8,9/, improve with increasing B^2a, and CIT has been designed to maximize this figure of merit. The plasma current was also maximized consistent with the engineering design constraints for the poloidal field system and MHD stability considerations for the safety factor, q, at the plasma edge. A large current is necessary to allow a high beta at high field ($\beta \sim I/aB$), and to maximize energy confinement, since $\tau_e \sim I$ for most auxiliary-heated confinement scalings. The confinement properties of the CIT design have been analyzed using a number of confinement scalings. One way of characterizing the confinement properties is to plot power balance contours in density and temperature space (POPCON graphs) /10/. In this type of analysis, a power balance is computed for a range of densities and temperatures with given profiles, and contours

of constant required auxiliary heating power are drawn. Ignition is identified as the region above the zero power contour. The accessible operating space is confined to densities below a density limit (taken to be $< n > = 1.5B/Rq_{cyl}$ and to betas below the beta limit (taken to be $3I/aB$). The confinement model has the form $\tau_E^{-2} = \tau_{OH}^{-2} + \tau_{AUX}^{-2}$ where $\tau_{OH}(\tau_{AUX})$ is the ohmic (auxiliary-heated) confinement time. Generally, the neoAlcator value is used for τ_{OH} and the regression fit given by Kaye and Goldston /8/ is inserted for τ_{AUX}, where, since $\tau_{KG} \sim P^{-0.6}$, the total power $P = P_{OH} + P_{AUX} + P_\alpha$ is used. With this analysis and with relatively flat density profiles ($n(r) \sim (1-(r/a)^2)^{0.5}$), CIT does not ignite. However, ignition is easily achieved (Fig.1) with an enhanced "L-mode" scaling ($\tau_{AUX} = 2\tau_{KG}$) used as a model characterization of the "H-mode" which retains degradation with increased power. In fact, detailed analyses show that an enhancement of $\sim$ 1.3 over Kaye-Goldston allows ignition, albeit with a small operating window. For more optimistic scalings such as the ASDEX-H /11/, ignition is easily achieved. CIT has been designed with a poloidal divertor and is expected to be capable of H-mode operation.

Ignition can be obtained with "L-mode" scalings if the density profile is allowed to be more peaked than $(1-(r/a)^2)^y$, with $y \geq 1.5$ /12,13/. Such peaked profiles would be obtained by a combination of pellet fueling, off-axis ICRF heating, and careful tailoring of the current profile during ramp-up to postpone the onset of strong sawtooth oscillations. Time dependent analyses of these scenarios have been carried out by 1 1/2 D transport code simulations using BALDUR and WHIST.

In summary, CIT ignites with a modest enhancement of a factor of 1.3 times "L-mode", and is able to ignite with "L-mode" scaling if the density profile can be sufficiently peaked by pellet injection. It thus has margin for parametrically investigating the energy confinement questions associated with intense alpha particle heating which are crucial to the present ETR designs, all of which have less confinement margin than CIT.

MHD Stability. The ideal MHD stability and β-limit depend strongly on the pressure and current profiles. Optimized profiles yield stable operating regimes with β well above that required for ignition. More realistic profiles such as those which would result from sawtoothing activity have broad, nearly shearless regions in which high-n ballooning and low-n infernal modes limit the pressure gradient and reduce the β-limit. Profiles which are optimized for $P(\psi)$ but which also obey an ohmic constraint yield β-values close to the Troyon limit ($\sim 3I/aB$) except for isolated, nearly rational values of q_a corresponding to $\beta = 0$, current-driven kinks /14,15/.

Impurity Control. Impurity control and power handling capacity are key issues for high power, ignited experiments. The principal impurity control system for CIT is a double null poloidal divertor which also has the additional advantage of enhancing the energy confinement with H-mode operation. The high density, compact approach to ignition leads to very high average heat loads on the wall, on the order of 1.5–2 MW/m^2 for CIT. In addition, the first wall hardware must be able to withstand the damage of disruptions which will cause very high peak loads, on the order of 100's to 1000's of MW/m^2. Because graphite can withstand very high peak loads without melting, it will be used to protect the vacuum vessel, and as the material for the divertor plates. The present divertor plates are inclined with respect to the flux surfaces to reduce the peak heat fluxes of $\sim$ 100 MW/m^2 normal to the flux surface to the $\sim$ 5 MW/m^2 required to keep the surface temperature of the graphite below the sublimation point. The divertor configuration is somewhat open, but is also designed to isolate the recycling neutrals away from the main plasma as much as possible. The highly shaped divertor plates that are required to reduce the peak heat fluxes impose severe requirements on the control of the X-point and plasma configuration, reduce the flexibility of plasma operation, and are sensitive to the

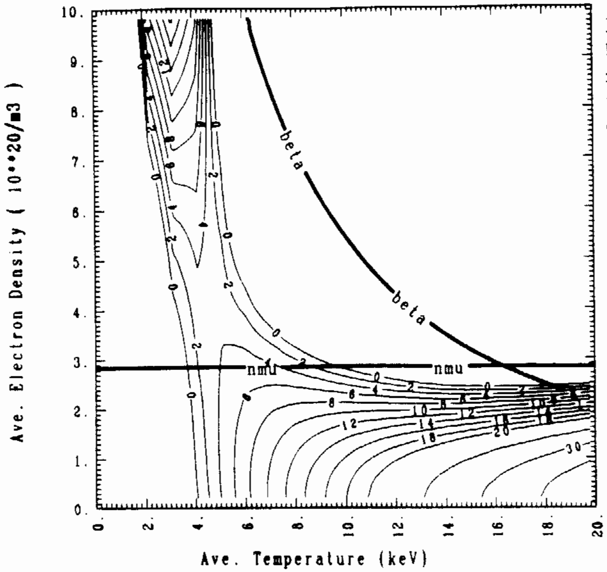

FIG.1 Equilibrium power balance contours in an n-T plane for a CIT device with the parameters of Table I. Each contour corresponds to a fixed auxiliary-heating power and ignition is given by the contour $P_{AUX} = 0$. The Hugill-Murakami density and Troyon-β limits discussed in the text are also shown.

profile of the plasma edge parameters. Studies of alternative designs with increased flexibility are presently underway.

References

/1/ "Effect of Sawtooth Oscillations and Ballooning Modes on Ignition in CIT," G. Bateman, *et al.*, Bull. Am. Phys. Soc. **31**, (Nov. 1986, Baltimore), 7Q7.

/2/ "CIT MHD Stability," M. Hughes *et al.*, ibid, 7Q11.

/3/ "Alpha Particle Loss Due to Ripple and MHD Modes," S. Zweben, *et al.*, Sherwood Theory Conf. (April, 1987, San Diego) 3A25.

/4/ "Stability of Shaped Tokamaks with Energetic Alpha Populations", D. Spong, *et al.*, ibid, 2C23.

/5/ D. Spong, *et al.*, Phys. Fluids **28** (1985) 2494.

/6/ K. Tsang, *et al.*, Phys. Fluids **24** (1981) 1508.

/7/ Y. M. Li, *et al.*, Phys. Fluids **30** (1987), in press.

/8/ S. M. Kaye and R. J. Goldston, Nucl. Fusion **25** (1985) 65.

/9/ S. M. Kaye, Phys. Fluids **28** (1985) 2327.

/10/ W. A. Houlberg, S. E. Attenberger, L. M. Hively, Nucl. Fusion **22** (1982) 935.

/11/ O. Gruber, Int. Conf. Plasma Physics (Lausanne, 1984) vol.1, 67-96.

/12/ "Delaying Sawtooth Oscillations in the Compact Ignition Tokamak," G. Bateman, PPPL-2373 (1986).

/13/ "Simulation of a Compact Ignition Tokamak Discharge (CIT-2L)," D. F. Stotler and G. Bateman, PPPL-2415 (Feb. 1987).

/14/ "MHD Stability Analysis of the Compact Ignition Tokamak (CIT)," M. W. Phillips, *et al.*, Sherwood Theory Conf. (April, 1987, San Diego) 1D10.

/15/ "MHD Equilibrium and Stability Considerations for CIT," J. Manickam, *et al.*, ibid, 3A15.

EFFECT OF CURRENT RISE ON ICRF HEATED PLASMAS IN THE JIPP T-IIU TOKAMAK

K. TOI, T. WATARI, R. AKIYAMA, R. ANDO, Y. HAMADA, S. HIROKURA, K. IDA, E. KAKO, K. KAWAHATA, Y. KAWASUMI, S. KITAGAWA, K. MASAI, K. MATSUOKA, A. MOHRI, S. MORITA, N. NODA, I. OGAWA, Y. OGAWA, K. OHKUBO, M. SAKAMOTO, M. SASAO, K. SATO, K.N. SATO, S. TANAHASHI, Y. TANIGUCHI, K. YAMAZAKI

Institute of Plasma Physics, Nagoya University,
Nagoya 464, JAPAN

Abstract

In limiter tokamak JIPP T-IIU, improvement of edge plasma transport is observed during high power ICRF heating, at the same time current density (j_ϕ) profile is *actively* controlled by an additional current rise(CR). D_α emission is reduced by ~ 40 %, compared with no CR discharge. Fall-off length of electron density in the scrape-off layer is remarkably reduced. Edge electron temperature derived from the intensity ratio of OVI lines is by about 2 times higher. Soft X-ray emission along the peripheral chord ($r/a_L \sim 0.7$) is enhanced by a factor of 2. Plasma stored energy is higher by ~ 30%.

1. Introduction

The confinement of tokamak plasmas is usually degraded by high power additional heating(L-regime). In the regime shapes of electron temperature(T_e) profile are approximately same for various deposition profiles of heating power, that is, T_e profile follows the so-called profile consistency. The reason seems to be due to the *consistency* in a current density(j_ϕ) profile, which may be tailored by the plasma itself through island formation due to low m ($\geq$ 2) tearing modes. On stable ohmic heating plasma, local MHD pressure balance will be well-established by strong coupling of ohmic heating power and current density through Ohm's law. However, intense additional heating disturbs the pressure balance considerably, because in all of present heating experiments only the gradient of plasma kinetic pressure is changed without controlling j_ϕ-profile actively. In other words, j_ϕ-profile is modified indirectly through tokamak plasma transport or magnetic island formation. The violation in MHD pressure balance may enhance the plasma transport so that pressure difference should be reduced. For H-mode observed in divertor configuration [1] large pressure change is supported by high global shear region near plasma boundary untill j_ϕ-profile is favourably tailored by transport process. It is predicted from the above consideration that the transport enhancement during high power additional heating may be avoided by controlling j_ϕ-profile actively at the same time of the heating. One of the most promissing j_ϕ-profile is that with a