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Heating the Compact Ignition Tokamak (CIT)

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Abstract: The proposed CIT starts operation in the late 1990's with 20 MW of rf heating power. The tokamak and facility are to be designed to accommodate 50 MW auxiliary heating. The heating methods now being considered are ion cyclotron heating (ICH) and electron cyclotron heating (ECH). Aspects of these systems are described, and the choice of power level and type is discussed.

1 Requirements

The CIT is proposed to demonstrate and study ignition prior to the development of a prototype reactor. A small major radius (2.1 m) allows the cost to remain moderate, while high field and current (11 T and 12.3 MA) produce good confinement of plasma energy.[1] Two essential questions concern the pulse length and the amount of power.

1.1 Pulse length

At minimum, the heating power must last for the duration of the maximum magnetic field, which is 5 seconds. This time is selected to be an order of magnitude greater than the energy confinement time τ_E required for the ignition of a plasma at the empirical density limit of $\approx 5 \cdot 10^{20}$ m^{-3}. A longer pulse length is desired to allow for heating during ramp-up and ramp-down of the magnetic field. Heating as the field increases adds energy, and temperature, to the plasma before the flat-top period; and heating as the field returns to zero can possibly help avoid disruptions. Another factor is that the allowable pulse length increases rapidly as the operating field is reduced from the design value. Therefore, operating below maximum field benefits from a longer heating pulse length up to the point of straining the thermal properties of the tokamak and the heating system.

In sum, a practical judgement is that 10 seconds should be the maximum duration for heating.

1.2 Heating power

Empirical scaling laws relating electron density n_e, temperature $T_{e,i}$, energy confinement time τ_E, heating power P_{aux}, and other parameters have been the subject of several studies.[1,2] A key premise is that parameters which fit existing tokamak achievements of τ_E can be applied to the future, and that an enhancement factor c_τ can be introduced on τ_E to take account of improvements to confinement seen in some discharges. The value of c_τ is in the range 1.5 - 2.5.

The following conclusions can be drawn from confinement studies:

1. Heating power of 20 MW transmitted from an rf launcher is sufficient to bring the discharge to ignition early in the flat-top period if confinement is sufficient to allow ignition.

2. Heating power of 50 MW or more might be required to reach maximum energy multiplication Q if confinement does not support ignition.

3. Operation without tritium will require power greater than 50 MW to reach the β-limit at full field because of the lack of α-heating.

It seems prudent to plan on a range of heating powers, starting with a moderate level for the first discharges and allowing for additions in the future. The initial complement of auxiliary power for CIT is 20 MW based on the first conclusion. The tokamak and supporting facilities will be designed to accommodate up to 50 MW of rf heating power for high-Q studies with tritium, and high-n_e, T, β studies without tritium.

1.3 Other requirements

The basic requirements on the heating are 20 MW, expandable to 40 or 50 MW, 10 seconds duration, as stated above. Other important qualities are:

1. Heating efficiency over a broad range of density and temperature;

2. High power density, reducing the number of ports required;

3. Low impurity generation;

4. Robustness of the wave launcher to disruptions, and fluxes of particles and radiation;

5. Usefulness over a range of operating magnetic fields. The maximum field is 11 T. Operation from 8 to 11 T is desirable for testing and scaling studies.

2 Heating methods

2.1 Possible technologies

The technologies tested for heating on tokamaks include the following, in approximate chronological order: neutral beams (40-100 keV particles), lower hybrid (0.5-5 GHz waves), ion cyclotron (30-100 MHz waves), electron cyclotron (60 GHz waves), and high-harmonic ion cyclotron heating (30-200 MHz waves). ICH and ECH have been chosen for integration with the CIT design.

Neutral beam heating (NBH) has been considered, especially in the past year.[3] However, this option remains inactive because development of sources, accelerators, neutralizers, and transport is not being supported, and because the beam lines change the conception of the facility, and possibly the tokamak. Ion Bernstein wave heating (IBWH), referred to above as high-harmonic ion cyclotron heating, has an attractive implementation based on fundamental-mode waveguides operating at 250 MHz. The waveguides fit into the port, wide edge vertical, and are fed with coaxial lines near the tokamak.[4] IBWH could possibly be considered if high power experiments achieve good results in the future. Lower hybrid waves have had good success heating electrons, and outstanding results with current drive. However, an installation for CIT would need 10 GHz, and therefore extremely fine-scale launching structures. Also, there is a fundamental limit to penetration at high temperatures (above 10 keV) from Landau damping.

2.2 Choices

2.2.1 Baseline

The choice of ICH was based on success in experiments such as PLT and on the ready availability of power in the needed frequency range (100 MHz). Recent ICH results on JET[5] and TFTR[6] have demonstrated efficient heating at temperatures relevant to ignition. The plan is to implement a 20 MW ICH system on CIT for first operation, using five ports of the tokamak, and ten transmitters adapted from the former Fusion Materials Irradiation Test (FMIT) program. The total cost of the installed power and antenna is in the range of $40M, or approximately $2 per watt. This assumes R&D on the antenna assembly and on required improvements to the transmitter, which are accepted as components of the multiple-purpose R&D effort in the Development and Technology Division (D&T) of the Office of Fusion Energy (OFE) in the Department of Energy.

Seven CIT ports have been allocated to heating. If increased power is required relative to the baseline (20 MW ICH), then two more ports can be assigned to ICH. The resulting system has 28 MW at the plasma, and 14 FMIT units. Increasing ICH power beyond 28 MW will require increased power density at the launchers. Such an increase is not planned.

2.2.2 Alternate

The choice of ECH was influenced by consistently successful heating in tokamaks, including high-c_r observations in Doublet-III-D,[7] and by extremely high power densities for transmission (100 MW m^{-2}), and by positive prospects for developing powerful sources in the range of 300-600 GHz as required for the high-field CIT. High power sources of millimeter radiation meeting CIT requirements have not been developed yet, but enthusiasm for ECH exists because the power can be launched in vacuum, some distance from the plasma, and because coupling is insensitive to plasma density.

Two technologies are receiving large R&D funding from the OFE in the ECH area: the gyrotron and the Induction accelerator Free Electron Laser (IFEL). The OFE Development and Technology Division (D&T) supports the gyrotron, and OFE Confinement Systems Division (CS) supports the IFEL. The IFEL project depends on spin-off from a weapons program. Plans on the gyrotron and the IFEL show development of ECH to a proof-of-principle in the early 1990's. The plan for CIT shows a decision in the same period of time on whether one, or even both, methods should be included in the CIT project. Until that decision, the option of installing 10 to 30 MW of ECH power is being maintained in the design process.

Preliminary cost studies show that gyrotron and IFEL implementations can be as low as $4 per watt of rf power if a building is available. R&D is not included in such an estimate. Use of an existing high-voltage dc power system, such as the TFTR neutral beam power supply, can lower implementation costs by approximately $2 per watt of rf.[8]

The CIT design goal for ECH is 15 MW per port, averaging nearly 40 MW m^{-2}, so that 30 MW is coupled through only two ports.

2.3 Strategy

CIT strategy regarding heating methods is as follows:

1. The plan for CIT shows 20 MW of ICH available at the beginning of operations.

2. Design of CIT and the ICH system is proceeding in accordance with this plan.

3. Design of the ECH systems is proceeding at the conceptual level. CIT design is proceeding with the guideline of keeping both ECH options open, based on the ECH design.

4. Four years or so prior to completion of the tokamak, CIT plans a decision on how to proceed with respect to ICH and ECH.

3 Design descriptions

3.1 ICH system

3.1.1 Physics

The heating mode chosen for ICH is minority ^3He in a DD or DT plasma, owing to the high absorptivity and successful heating in past experiments. In a heated DT plasma, tritons can receive energy directly through second harmonic resonance once the β is sufficiently high. At the CIT densities, the single-pass absorption rates are expected to be high, even if the exact location of cyclotron resonance is midway from plasma center to edge. Fast particles are not expected to be induced by the rf fields due to the short equilibration times.[1]

The frequency corresponding to 11 T is 105 MHz, including a 5% reduction from the nominal resonant formula to account for some improvement predicted for heating slightly outside the major radius. To allow heating over a range of fields, the rf source and transmission system is specified to cover 60 – 110 MHz with a tuning time measured in hours.

3.1.2 Antenna

Coupling to the ICH waves is planned through inductive loops protected from the plasma by limiters or recesses in the vessel surface, or possibly both. Current elements in two toroidal locations in each port will allow spectral control, in particular to emphasize high $k_{\|}$ most often associated with low impurity production.[9] The loops are split in the poloidal direction to reduce voltages in the existing engineering design.[10] Each loop is fed at the end, so eight coaxial transmission lines are mated to the flange on each of five ports. The resulting layout suffers from the crowding of components, so the option of two current loops separated toroidally will be studied. Recent experience with such loop couplers is consistent with the CIT goal of 4 MW through each port of 0.4 m by 1.0 m, making the net power density 10 MW m^{-2}.

Disruptions can exert large forces on the Faraday shields and current-carrying straps of the antenna, thereby leading to a difficult design problem, in large part owing to uncertainty in the behavior of the disruption. Heat loads from plasma and radiation are also somewhat uncertain, but it is clear that the goal of inertial cooling will be difficult to accomplish for a full ten seconds in a full-power, ignited experiment.

3.1.3 Source

Ten modified FMIT units, each equipped with an upgrade to the EIMAC X-2242 power tetrode, will supply the rf power. To account for transmission and coupling losses, each unit is asked to provide 2.6 MW at 105 MHz. Such an achievement has been judged to be possible with development effort. A joint effort of General Atomics, Varian, and the Japan Atomic Energy Research Institute has made progress in this direction.

3.2 ECH physics

For ECH, the ordinary mode with $\mathbf{E}_{rf} \parallel \mathbf{B}_0$ is selected because it can be launched from the outside ports if the frequency is at or above the central cyclotron frequency. Good accessibility for fundamental harmonic waves is calculated up to a maximum density of $8 \cdot 10^{20}$ m^{-3}, and absorption is strong when the wave crosses the cyclotron-resonant layer and $T_e > 1$ keV and $n_e > 1 \cdot 10^{20}$ m^{-3}.[1]

For ECH it may be that the development of a tunable, high-average-power source is impractical or impossible for frequencies in the desired range of 300 GHz (600 GHz for second harmonic). Methods of heating the plasma over a range of fields are being investigated through modelling calculations. A very simple technique is to arrange for resonance to be outboard of center at the highest field, and inboard of center at reduced field. More flexibility can be obtained by using the Doppler shift to heat particles of a given parallel velocity at locations that are not at cyclotron resonance. This would require a launching surface that can be angled either for different discharges, or as a function of time in a given discharge.[11]

The maximum density could be raised by going to second harmonic. In that case the extraordinary mode is preferred because of the stronger interaction with the electrons than the ordinary mode. In the case of a pulsed source of microwave power, such as the IFEL, nonlinear effects may reduce absorption, or cause other problems. Calculations on these matters suggest that no practically important effects will emerge, but experimental verification is planned.[12]

3.3 Gyrotron-based implementation

The power source is taken to be a nominal 1 MW gyrotron tube operating at a frequency in the neighborhood of 300 GHz. Step-tuning is desirable over a range 200 – 310 GHz, but fixed-frequency is assumed.

The TFTR neutral beam power supply is assumed to provide prime power, up to a maximum of 30 total MW of rf from 30 tubes. The existing transformers must be rewound in this high-power case. Other assumptions are 32% tube efficiency at 90 kV and 35 A, and 90% transmission.[8]

Transmission from rf source to plasma is a major problem in the case of these high frequencies. A corrugated circular waveguide operating in the HE_{11} mode has been used at 60 GHz, and may be applicable at 300 GHz. A purely optical transmission may be required, however, at some cost in space allocation. In each case, a beam combiner resembling a Cassegrain telescope has been studied as a method to transport energy quasi-optically from the neighborhood of the gyrotrons, turning corners with spherical mirrors as the energy approaches the tokamak. Under this method, a window exists far from the tokamak at each gyrotron and another window exists at the tokamak vacuum boundary in the basement of the installation.[8,13] Whatever the transmission method, pure mode output from the gyrotron tube and extensive transmission technology development will be required.

R&D is an important element in the gyrotron program. The problem is to develop and test sources and transmission systems at the MW level, and in the frequency range of 300 GHz. The general approach is that Varian Associates will continue, from a base of 100 gyrotrons now sold, toward cw tubes at 500 kW and 1 MW at frequencies of 140 GHz and 110 GHz. The use of 110 GHz tubes on a 2 MW installation at General Atomics' Doublet-III-D tokamak will provide extra enthusiasm for concrete achievements in this area. In parallel with the work at Varian, the MIT Coherent Sources group at the Plasma Fusion Center is making progress on a megawatt-relevant, microsecond-pulsed gyrotron tube which uses a whispering gallery resonant cavity in a very high mode to minimize ohmic losses. Some recent short-pulse achievements are 925 kW at 148 GHz with 20% efficiency,[14] and step-tunability over 140 – 328 GHz in the 400 kW range.[15] Given success at Varian and MIT, it is conceivable that a MW-level, 300 GHz tube could be launched before the mid-1990's in conjunction with a major ECH effort on CIT.

3.4 IFEL implementation

An IFEL starts with an electron beam in the dozen-MeV, several kA range, which is passed through a wiggler structure at very low duty factor. The key advantages of this method are the potential tunability over $\approx$ 10% by operating within the gain curve, and over factors of two by adjusting the voltage and parameters of the different wiggler. A 10 – 40 MW implementation has been studied for CIT, concluding that the accelerators and wigglers could be placed in the TFTR 'mock-up' building.[16–18]

The GW pulses (lasting dozens of nanoseconds) preclude use of a window, so the tokamak vacuum extends through the IFEL accelerator and wiggler, relatively far from the plasma. Transport of the microwave beam will require development, as with the gyrotron. Although beam efficiency is expected to be high — in the range of 30% — the wall plug efficiency may be in the range of 10% because of various losses in power conversion.

R&D for the IFEL approach is proceeding at the LLNL in the Microwave Tokamak Experiment (MTX).[13] Plans call for single pulse experiments this year, and steady improvements leading up to 2 MW for 0.5 seconds at 250 GHz before the mid-1990's. Experimental topics are current drive and MHD control, in addition to heating. A significant achievement on MTX in the direction of controlling disruptions would certainly lead to the consideration of the IFEL for that purpose on CIT.

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