RC ITER: An Opportunity to Study Burning Plasmas and Develop Fusion Technology in a Reactor Relevant Device

R. Parker
MIT Plasma Science and Fusion Center

I. Introduction

During the last year of the ITER EDA, the ITER parties recognized that the cost of ITER, although in line with that estimated at the conclusion of the CDA, was nevertheless an insurmountable barrier to entering into a construction agreement. However, the Parties also recognized that the arguments leading to the formation of the ITER collaboration in 1986 were still valid and that the goals originally envisioned for ITER were not diminished in their validity. It was thus decided to attempt a redesign of the EDA device, in which the original goals and objectives would be retained as much as possible but with a cost objective of about half that of the EDA design. A number of names and acronyms have been used to refer to the redesign; in this paper it will be called Reduced Cost (RC) ITER.

Several basic design options, corresponding to different choices of aspect ratio, have been considered, namely a high aspect-ratio machine (HAM, A~3.5), one with intermediate aspect-ratio (IAM, A~3.26) and one with relatively low aspect-ratio (LAM, A~2.76). The HAM design has been abandoned owing to relatively poor access, lower shaping capability, higher cost and limited potential for electron cyclotron heating and current drive. Although there seems to be an emerging consensus toward selection of the IAM option, no official choice between these variants has yet been made, and the design and performance of both the IAM and LAM will be described below.

2. Objectives

As implied by the title of this paper, RC ITER has both scientific and technological objectives, and these are, as much as possible, in line with the objectives established for the EDA design. Specifically, with regard to *plasma performance*, the device should:

- Achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power of at least 10 for a range of operating scenarios and with duration sufficient to achieve stationary conditions on the time scales characteristic of plasma processes;

- Aim at demonstrating steady state operation using non-inductive current drive with the ratio of fusion power to input power for current drive of at least 5.

In addition, the possibility of controlled ignition should not be precluded.

In regard to *engineering performance and testing* the device should:

- Demonstrate the availability and integration of technologies essential for a fusion reactor (such as superconducting magnets and remote maintenance);

- Test components for a reactor (such as systems to exhaust power and particles from the plasma);

- Test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high-grade heat, and electricity generation.

Note that the only significant change from the EDA objectives is the replacement of the requirement to achieve ignition with the requirement to achieve a high gain Q ~ 10 burn, although the possibility of achieving ignition is still held out as being desirable. It is this reduction in required performance that allows substantial size and therefore cost reductions to be realized.

3. IAM and LAM Designs

The main parameters of the IAM and LAM are presented in Table 1 and compared with the corresponding parameters of the EDA device as described in the Final Design Report (FDR). Note that the IAM design has higher field and lower current than LAM, and has somewhat less shaping. IAM plasma shapes are limited to single null configurations, whereas LAM can be operated either with a single null or an up-down symmetric double null equilibrium. A feature of the LAM design is that the field at the TF coils is low enough to permit use of NbTi conductor throughout the coil. NbTi conductor is used for the PF coils in both designs, except for the CS which is wound from $NbSn_3$ conductor. Cross sections of the two designs are shown in Figure 1 where it can be seen that the access in LAM is somewhat better than that in IAM.

Both designs meet the objective of lowering the construction cost by about a factor-of-two below the cost of the FDR ITER, while offering a performance level consistent with the revised objectives given above. Moreover, both IAM and LAM designs are responsive to concerns, raised particularly by members of the US physics community, regarding the performance and flexibility of the FDR device. Both designs have a segmented central solenoid which permits stronger shaping, and both designs achieve their reference performance with $n < n_{GW}$. Further, recent results obtained by the Edge Expert Group [1] have revealed a size scaling for the width of the edge pedestal that is more favorable for overall confinement projections than the poloidal gyroradius scaling proposed in [2]. And finally, more attention has been paid to the steady-state operation of RC ITER, including the incorporation of substantial current drive capability in an advanced tokamak mode, as the promise of these modes continues to be supported by results obtained by essentially every tokamak within the worldwide tokamak community.

	IAM	LAM	FDR
R(m)	6.20	6.45	8.14
a(m)	1.90	2.33	2.8
Plasma Configuration	Single Null	Single or Double Null	Single Null
I_P(MA) (q_{95} = 3)	13.3	17	21
B_o (T)	5.51	4.23	5.68
Ignited/Burn Pulse Length (s)	450	450	1000
Elongation κ_{95}, κ_X	1.68, 1.83	1.74, 1.92	1.6, 1.75
Ave Triangularity δ_X	0.43	0.49	0.35
<T> (keV)	10.5	10.8	12
$\langle n_e \rangle$ (10^{20} m^{-3})	0.83	0.83	1.0
$\langle n_e \rangle / n_{GW}$	0.87	0.83	1.17
Z_{eff}	1.9	2.0	1.8
Fusion Power (MW)	505	525	1500
β, β_n (%)	2.86, 2.25	3.88, 2.25	3, 2.2
Ave Neutron Wall Load (MW/m^2)	0.6	0.5	1.0
Number of TF Coils	18	20	20

Table 1. Main parameters of IAM and LAM and comparison to the FDR design.

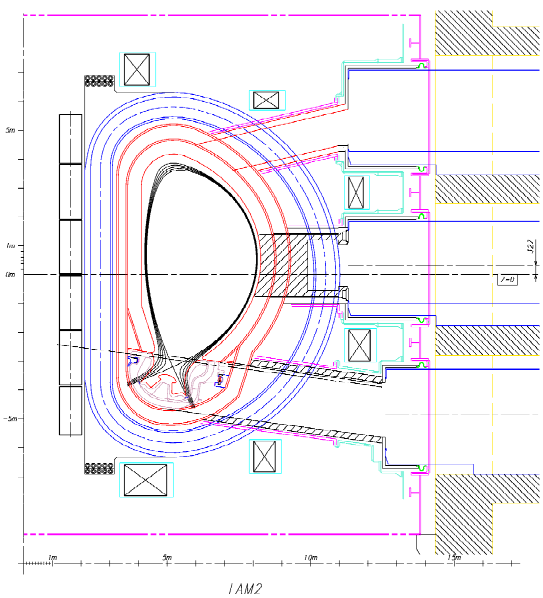

Fig 1a. Cross section of IAM design.

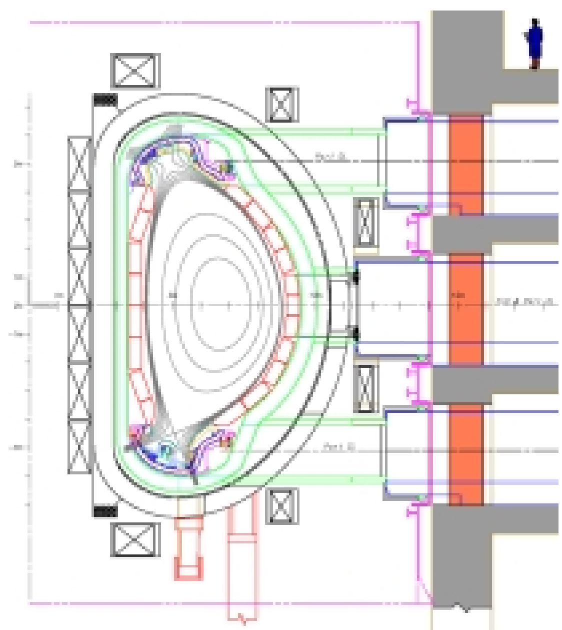

Fig 1b. Cross section of LAM design.

4. Operating Regimes and Performance Margin

4.1 Inductive performance

As shown in Figure 2a and 2b, both the IAM and LAM have reasonable margin in obtaining their baseline performance. The shaded area in the figures corresponds to the region of parameter space simultaneously below the Greenwald density and $\beta_n < 2.5$, but above the L-H transition scaling in the plane of fusion power vs. H_H, where H_H is the factor by which confinement exceeds IPB98(y,1) H-mode confinement scaling. Thus, within the nominal constraints, Q = 10 can be obtained in both machines with confinement degraded to as low as 80% of that predicted by extrapolation of the IPB98(y,1) H-mode scaling. Higher Q performance for both machines is possible, although the operating window naturally shrinks. As required by the RC ITER objectives, the possibility of ignition is not precluded but requires some enhancement over the H-Mode confinement scaling projection. In particular, as shown in Figure 3, the window shrinks to essentially a point for IAM with nominal q=3 operation, while a small domain for ignition is predicted to exist for LAM even at q=3.

4.2 Non-inductive performance

Achieving a steady-state $Q \geq 5$ requires modest improvement in confinement and normalized β. Shown in Figure 4 are Q and β_n vs. the effective current drive power with H_H as a parameter. Here, the current drive efficiency nIR/P_{CD} is assumed to scale linearly with temperature, and γ^* is the current drive efficiency at T = 10 keV. In all cases the thick lines correspond to IAM and the thin to LAM. Also, in Figure 4b, the three cases are for H_H = 1.5, 1.25 and 1, as in Figure 4a. Thus, for example, with $\gamma^* = 0.2$ and P_{CD} = 70 MW, Q ~ 5 is possible with H_H = 1.25 and $\beta_n \sim 3.5$. Note that the current drive performance is slightly better in IAM than in LAM and that in both designs, advanced tokamak operation is required to achieve the steady-state Q =5 goal.

An important parameter regarding steady-state operation is the pulse length capability normalized to the L/R time, the characteristic time for decay of the electric field in the plasma. For fully superconducting machines such as RC ITER, the pulse length can be made arbitrarily long providing there is sufficient cooling capability to cope with nuclear heating and incidental coil heating due to variations in the plasma control power. In RC ITER, steady-state pulse lengths of an hour or more are anticipated, corresponding to several L/R times. The ability to produce truly steady-state conditions reflects an important advantage that well-shielded superconducting machines enjoy over relatively short pulse and poorly shielded compact, copper burning-plasma experiments.

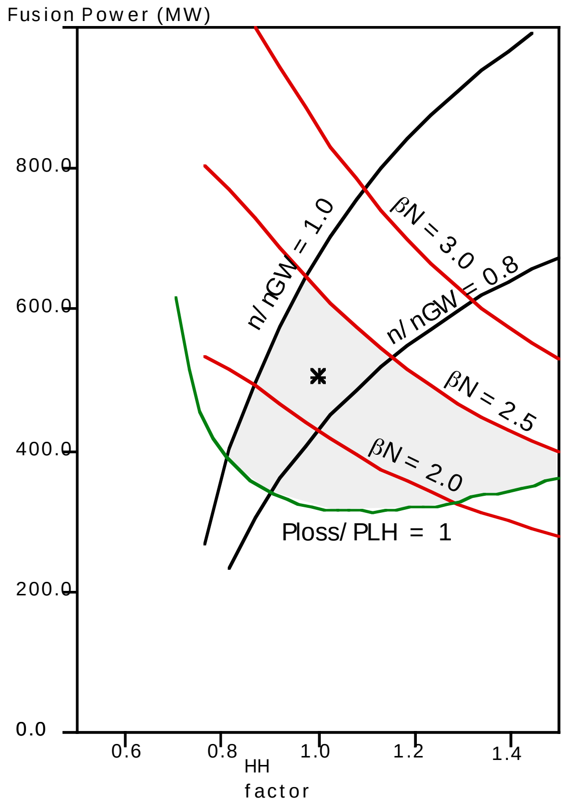

Fig 2a. IAM operating domain for Q=10.

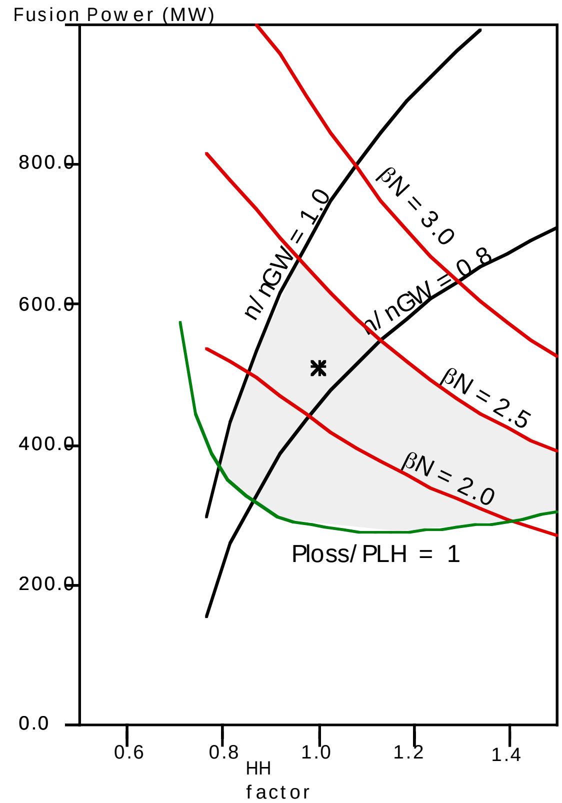

Fig 2b. LAM operating domain for Q=10.

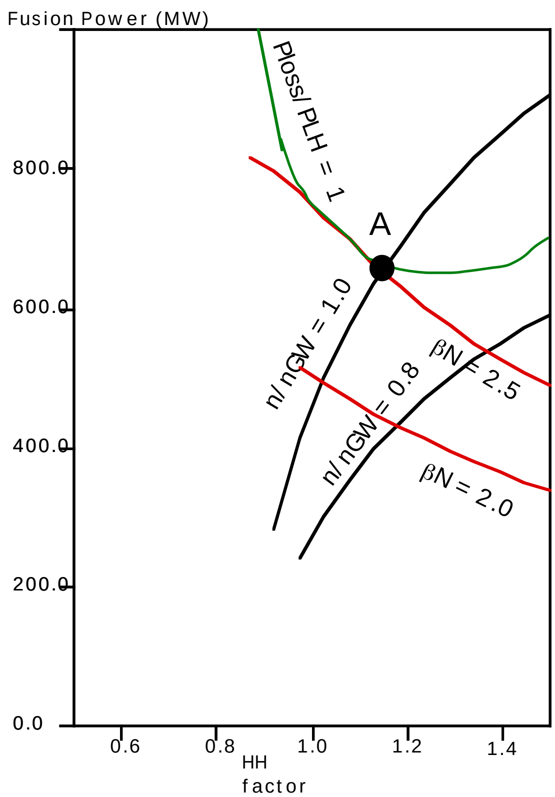

Fig. 3a. IAM ignited domain

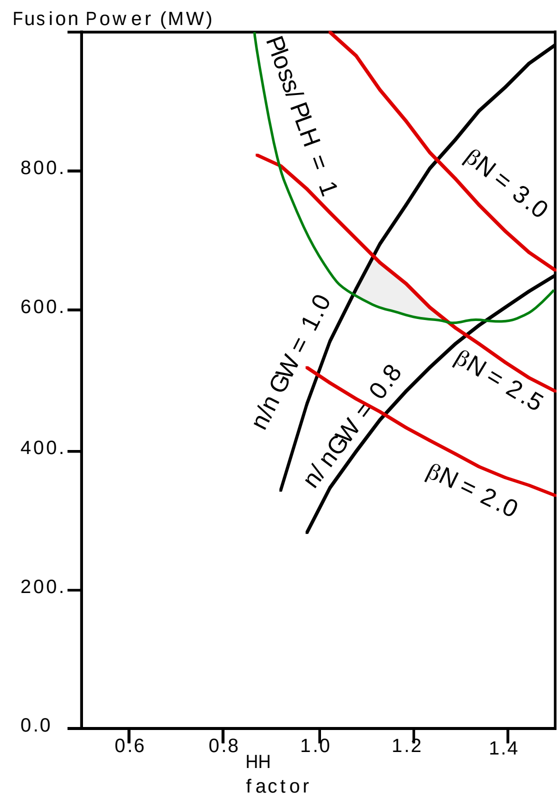

Fig. 3b. LAM ignited domain

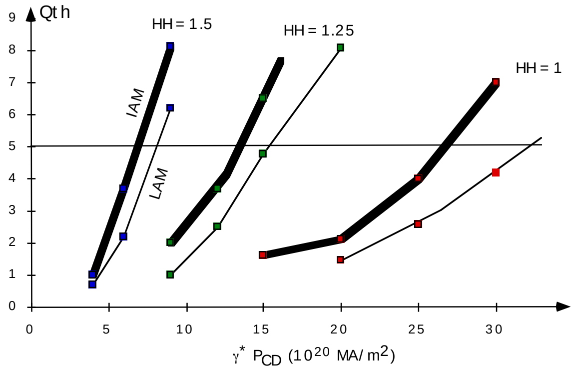

Fig4a. Q vs. effective current drive power with H-mode multiplier as parameter.

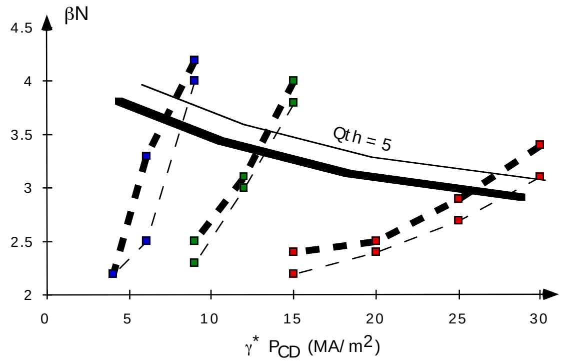

Fig4b. β_n vs. effective current drive power with H-mode multiplier as parameter. The three sets of dashed curves correspond from left to right to the H_H values in Fig 4a.

5. Access and Diagnostics

While not as impressive as the access in the FDR ITER design, the access in both RC ITER design variants is exceptional by standards of today's large tokamaks. For example, the 18 equatorial ports in IAM have cross-sectional dimensions of 1.74 x 2.2 m^2, while the 20 equatorial ports in LAM measure 1.5 x 2.2 m^2. Such generous access is required by the demands of auxiliary heating, diagnostics and blanket module testing.

An initial installation of about 75 MW of auxiliary power is planned, with 33 MW coming from negative ion neutral beams and 40 MW from RF H&CD. The latter will be injected through two ports and can be made up of 40 MW of a single H&CD band chosen from ICRF, ECRF or LHRF, or two different 20 MW systems chosen from these three bands. Port allocation allows an additional 40 MW to be added; in addition, some upgrade of the NBI power may be possible. Thus, as an experiment of this magnitude demands, there is a high degree of flexibility in both the choice of H&CD schemes and the total H&CD power.

As important as adequate H&CD is to a burning plasma experiment is implementation of a comprehensive set of state-of-the-art diagnostics. Extensive planning for the diagnostics has been done for RC ITER and a list of the diagnostics presently foreseen is presented in Table 2. Ports have been allocated for each of these diagnostics and detailed design work has been done for many of them at a fairly detailed level, including the machine interface. It should be emphasized that RC ITER is, above all, a physics experiment and, as with any experiment, its value in providing physics understanding is strongly dependent on the scope and depth of the diagnostic coverage. This point should be borne in mind when comparing a machine in the RC ITER class with lower cost, compact ignition experiments using copper coil technology.

Magnetic Diagnostics	Optical/IR Systems (Cont'd)	**Microwave Diagnostics**
Vessel wall sensors	Divertor Thomson Scattering	Electron Cyclotron Emission (ECE)
Divertor Magnetics	Toroidal Interferometric/ Polarimetric System	Main Plasma Reflectometer
Continuous Rogowski Coils	Polarimetric System (Poloidal Field Measurement)	Plasma Position Reflectometer
Diamagnetic Loop	Collective Scattering System	Divertor Reflectometer & Divertor ECA
Neutron Diagnostics	**Bolometric Systems**	Main Plasma Microwave Scattering
Radial Neutron Camera	Array For Main Plasma & Array For Divertor	Fast Wave Reflectometry
Vertical Neutron Camera	Spectroscopic and NPA Systems	**Plasma-Facing Components and Operational Diagnostics**
Micro-fission Chambers (In-Vessel)	Charge eXchange Recombination Spectroscopy (CXRS) based on DNB	IR/Visible Cameras
Neutron Flux Monitors (Ex-Vessel)	Motional Stark Effect (MSE): based on heating beam	Thermocouples
Radial Neutron & Gamma-Ray Spectrometer	H Alpha Spectroscopy	Pressure Gauges
Activation System (In-Vessel)	Main Plasma & Divertor Impurity Monitors	Residual Gas Analyzers
Lost Alpha Detectors	X-Ray Crystal Spectrometers	Hard X-Ray Monitor
Knock-on Tail Neutron Spectrometer	Visible Continuum Array	IR Thermography (Divertor)
Optical/IR Systems	Soft X-Ray Array	Langmuir Probes
Core Thomson Scattering	Neutral Particle Analyzers	**Diagnostic Neutral Beam**
Edge Thomson Scattering	Two Photon Ly-Alpha Fluorescence	
X-Point Thomson Scattering	Laser Induced Fluorescence	

Table 2. Diagnostic systems planned for RC ITER

6. Concluding remarks

A machine in the RC ITER class is the optimum way in which burning plasma physics can be fully and relevantly studied:

4 Simultaneous scaling of the dimensionless parameters ν^*, ρ^* and β to reactor-level values can most closely be achieved in a device of the RC ITER scale;

5 Burning plasma phenomena pertaining to steady-state AT physics can best be investigated in machines with essentially steady-state capability (and can only be adequately investigated in machines with T_{pulse} > L/R);

6 Sufficient access for diagnostics, auxiliary heating and current drive, and remote removal and installation of in-vessel components, is essential in order to carry out a meaningful burning plasma experimental program. Such access can be fully realized only in a device of the RC ITER scale.

The cost of either RC ITER option is presently foreseen to be about 55% of that of the FDR, or ~ 3 B$ (1989). Approximately 10% of the cost could be deferred until after the start of operations. A 15% partnership in constructing RC ITER in Europe or Japan would cost the US ~ 60 M$ per year (1999 dollars) over a 10 year period. Thus, participating in an RC ITER project as a partner would be by far the most cost effective way for the US to significantly advance its program in fusion science and technology in the area of burning plasmas.

The price paid for a substantial cost reduction in ITER is a lowering of the baseline performance from ignition to a gain of about 10. However, achieving a moderate Q, steady-state plasma is actually more useful for advancing the tokamak concept than achieving pulsed ignition or even very high Q. Since tokamaks require auxiliary power to drive a steady-state current, there is no possibility to realize an ignited, steady-state tokamak reactor. The best that can be envisioned, at present, is a steady-state tokamak with a Q of perhaps10-20 and, in that sense, RC ITER has precisely the right goal. Further, nearer term applications of fusion neutrons, such as burning weapons-grade Pu or spent fuel from fission reactors, do not require very high Q's to be competitive with other approaches, for example spallation neutron sources. In the relatively near term the fusion program would be better served by emphasizing applications requiring moderate Q and steady-state, rather than pure ignition which is not required, nor possible, for steady-state tokamak reactors. The RC ITER step is fully consistent with this paradigm shift and the US should vigorously support its construction as a participating partner at the earliest opportunity.

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References

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