Possibilities for Long Pulse Ignited Tokamak Experiments Using Resistive Magnets

E. A. Chaniotakis
L. Bromberg
D. R. Cohn

April 25, 1991

Plasma Fusion Center
Massachusetts Institute of Technology
Cambridge, Massachusetts 02139 USA

Possibilities for Long Pulse Ignited Tokamak Experiments Using Resistive Magnets

E. A. Chaniotakis
L. Bromberg
D. R. Cohn

MIT Plasma Fusion Center

ABSTRACT

Tokamak designs based on copper and copper alloy magnets could be used for the investigation of the physics issues associated with long pulse (> 40 sec.) ignited operation during the next phase of fusion research. The engineering characteristics of designs with magnets that use copper or beryllium copper alloys are presented. Active cooling of the magnets with either liquid nitrogen or water is considered. Inertial cooling is also discussed. The physics performance of the designs is calculated and compared to the performance of the "Physics Phase" ITER design, and the BPX tokamak.

1 Introductory remarks

The objectives of the fusion program prior to the development of an electricity producing pilot plant or demonstration reactor are to first investigate the fundamental physics issues of long pulse, ignited operation and second to demonstrate and integrate the required technologies. Several tokamak designs have been proposed for achieving these goals. The International Thermonuclear Experimental Reactor (ITER), a superconducting device, is being designed to achieve both of the above mentioned goals in a single machine at an estimated cost of 8 billion US dollars.[1] In addition, the United States burning plasma experiment (BPX) as well as the IGNITOR

design developed in Italy, are compact high performance resistive magnet machines designed to demonstrate alpha dominated operation in relatively short (≤ 5 sec.) pulses. In this paper resistive magnet tokamak designs which would be capable of addressing the physics issues associated with long pulse ignited operation are discussed.

ITER is designed so that its configuration can be changed in order to investigate both the physics and the technological issues.[1] Initially, in the "Physics Phase" the physics issues related to ignition and long pulse operation will be investigated. Next, in the "Technology Phase" the issues related to technological questions will be addressed.

Relying on a single experiment for addressing all of the questions associated with fusion development may not be the optimum research path. The gap between the present state of fusion development and the level of development required for the present ITER design to be successful is very wide. Information is required about issues such as confinement characteristics, effectiveness of alpha particle heating, and a variety of issues associated with long pulse operation.

The issues related to plasma transport characteristics and the development of reliable scaling relations for the global energy confinement time are of fundamental importance in fusion energy research. The uncertainty of the plasma transport processes and alpha heating effectiveness,[2] at the required thermonuclear conditions, and the coupling between the confinement scaling and the characteristics of the machine design raise strong questions of program flexibility especially for a project as costly as ITER. It is therefore important that some of these questions be addressed prior to the design of a technology oriented fusion experiment.

It is possible that the physics issues associated with ignition and long pulse operation could be achieved in machines that are characterized by significantly lower cost than the present ITER design and which could be built with less reliance on advances in fusion technology. In this way the physics machine does not have to depend on extensive engineering extrapolations and R&D.

2

We have evaluated tokamak designs with resistive magnets that are larger than the present BPX device and have the capability for active cooling of the magnets. The fundamental requirement for these designs is to be capable of providing high gain long pulses ($\sim 40 - 100$ seconds) at peak performance. This approach extends the LITE [3] design concept and uses copper and copper alloy plate magnets in machines with sizes in the 3.5 - 4.5 meter major radius range. These concepts could be an attractive candidate for international collaboration.

2   Illustrative Designs

Illustrative designs have been developed for machines whose toroidal field magnets are composed of copper and high strength Be copper alloys in a wedged design. The parameters for the machines under investigation are given on Table 1. The names C-1, C-2, C-3 refer to the copper designs and BeC-1 refers to a beryllium copper machine. These designs are capable of providing burn pulses in the range of $\sim 40 - 100$ seconds. The pulse length is limited by the OH transformer. The aspect ratio of the designs considered here is similar to that of BPX for ease of comparison. However, higher aspect ratio designs may have advantages for reactor development.

The toroidal field (TF) magnets are made of plates and enable high performance operation by virtue of their strength and structural design characteristics. These magnets for the machines could be actively cooled with liquid nitrogen or water. For operation at peak performance the beryllium copper design requires liquid nitrogen cooling while the copper designs can be cooled by either liquid nitrogen or water. For the beryllium copper design water cooling is possible at reduced toroidal magnetic fields. In both designs the coolant flows in channels grooved on the magnet plates. A high performance wound OH transformer is used with the wedged TF magnet design.

The resistive power in the TF magnets of the various designs is given on Table 2 along with other design features. Also, included on Table 2 is

Table 1: Parameters for resistive magnet tokamak designs

Parameter		Design name			
		C-1	C-2	C-3	BeC-1
Major Radius	R (m)	3.5	4.0	4.5	3.5
Minor Radius	a (m)	1.1	1.2	1.36	1.1
Aspect Ratio	A	3.2	3.3	3.3	3.2
Elongation	κ	2.0	2.0	2.0	2.0
Toroidal Field	B (T)	7.8	7.4	7.4	8.7
Plasma Current	I (MA)	14	13.5	15.4	15.6
Effective Charge	Z_{eff}	1.65	1.65	1.65	1.65

the TF magnet flat top time with liquid nitrogen inertial cooling at peak performance. The maximum temperature of the TF coils at the end of the pulse is approximately 290° Centigrade. Inertial cooling is thus sufficient for the inductively driven copper designs. The actively cooled TF design alternative could be advantageous in studying hydrogen or D-T plasmas with current drive. Also, they could enhance the duty factor of the machines.

An alternative design for the TF magnets could be based on the AL-CATOR C - MOD tokamak whose magnets are characterized by sliding joints with the forces been taken by a superstructure. This design could be advantageous with respect to assembly and maintenance. In addition, with a C-MOD like design it is possible to increase the pulse length since more volt seconds can be provided by the ohmic transformer.

Design C-1 has no magnet shielding and is capable of providing 10,000 burn seconds of D-T operation. In designs C-2, C-3, and BeC-1 a shielding thickness of 20 cm is provided for protection of the magnet insulation from the fusion neutrons. This shielding reduces the neutron flux by a factor of $\sim$ 30 and allows for 500,000 burn seconds of D-T operation.

3   Physics Performance

The physics performance of the designs presented on Table 1 is evaluated by considering ITER P-89 [4] energy confinement scaling. Necessary in this

Table 2: Design characteristics for the illustrative designs whose parameters are given on Table 1.

	C-1	C-2	C-3	BeC-1
TF magnet power for steady state operation in MW:				
for LN2 coolant (MW)	84	90	86	620
for H_2O coolant (MW)	470	530	530	—
TF magnet flat-top for inertial cooling (sec)	130	200	300	20
TF weight (tonnes$\times 10^3$)	2.4	3.3	4.7	1.8
TF magnet stored energy (GJ)	9.4	11.4	15.6	12.4
Shielding thickness (m)	—	0.2	0.2	0.2
Pulse length limit (sec)	40	100	100	60
*Fusion power (MW)	300	375	480	300
*D-T Burn seconds ($\times 10^5$)	0.1	5	5	5
*Total tritium consumed (gr)	8	500	630	400

*Evaluated at a wall loading of $1MW/m^2$

evaluation is the characterization of the improvement in confinement over that predicted by the energy confinement scaling. This confinement enhancement is represented by the H-mode factor which is a number greater than one. Usually for the performance to be increased to an acceptable level the H-mode factor must be approximately 2 (i.e. a two fold enhancement in confinement over the value given by the scaling relation). This anticipated increase in confinement is not completely arbitrary; there have been experimental results which indicate that in certain regimes of operation the confinement is indeed increased, even doubled, over the expected value in the so called L-mode of confinement.

The actual magnitude of confinement enhancement is uncertain. It is however possible to calculate the value that is required for achieving a certain performance level. For example, by requiring that a certain value of the thermonuclear Q (Q = Fusion Power / Auxiliary Power) be achieved and that the machine should operate below a certain neutron wall loading limit it is possible to calculate the minimum H-mode factor (H_m) for these conditions to be satisfied. On Table 3 the minimum enhancement factors are given for the designs under consideration and for the BPX and ITER designs, for various values of the thermonuclear Q. Note that the required confinement enhancement factor for machines C-1, C-2, C-3, and BeC-1 is 20-30 % lower than the enhancement factor required for BPX. Also note that the performance of ITER, as represented by the required H-mode enhancement factor, is only 5-15 % higher than the performance of the resistive magnet designs.

The performance of the 3.5 meter copper design C-1 is comparable to the performance of the 4 meter copper design C-2. This is because the larger design, C-2, has 20 cm of neutron shielding. Also note that the performance of the copper design C-3 is comparable to the smaller Be copper design BeC-1 since C-3 is limited by stress and not by cooling requirements.

The various plasma phenomena that need to be investigated evolve with different characteristic times. For example, at high values of Q (Q >10),

6

Table 3: The required minimum H-mode enhancement factor (H_m) for various designs and for different combinations of $P_w = 1$ MW/m^2 and Q (the ratio of Fusion power to the Auxiliary power). Perfect coupling of the fusion alpha particles is assumed.

	BPX (8.1 T)	C-1	C-2	C-3	BeC-1	ITER
$Q\Downarrow$	$P_w = 1$	$P_w = 1$	$P_w = 1$	$P_w = 1$	$P_w = 1$	$P_w = 1$
2	1.25	1.01	1.03	0.93	0.90	0.87
5	1.74	1.39	1.40	1.26	1.23	1.17
10	2.14	1.65	1.66	1.49	1.47	1.38
15	2.36	1.78	1.78	1.60	1.58	1.48
20	2.49	1.86	1.86	1.67	1.65	1.54
25	2.59	1.92	1.91	1.71	1.70	1.57
30	2.66	1.95	1.95	1.74	1.74	1.60

the temperature during thermal runaway evolves with a characteristic time which is 2-3 times the energy confinement time. On the other hand the study of current diffusion requires observation over a time that is comparable to the plasma L/R characteristic time which for classical processes is very long. Values for the various characteristic times of interest for the machines under consideration are given on Table 4. Pulses greater than 30 seconds are required for exploring, with sufficient resolution, issues such as evolution of self heated profiles, burn control (including both thermal stability and power level control), and alpha ash accumulation and removal. A variety of other "long pulse" issues including power handling, and impurity control could also be investigated.

Another important consideration in assessing the performance of the designs is to evaluate the operating characteristics of the divertors. In general, operation at high magnetic field and low plasma current facilitates the design of the divertor. In divertor design it is important to reduce both the temperature and the heat flux of the incoming particles at the divertor plates. The temperature at the divertor plates scales as $\sim 1/n^3$ while the

Table 4: Characteristic times for the illustrative designs whose parameters are given on Table 1. All the designs are assumed to operate with Q=20 and with 1 MW/m^2 neutron wall loading.

	C-1	C-2	C-3	BeC-1	BPX (8.1 T)	ITER
τ_E L-mode (sec)	0.68	0.7	0.83	0.76	0.48	1.1
τ_E H-mode (sec)	1.25	1.28	1.38	1.23	1.14	1.68
Classical L/R (sec)	1200	1400	1900	1200	700	4700
Alpha ash buildup to 10% fraction (sec)	10	10	11	10	6	14
Available pulse (sec)	40*	60*	100*	100*	3	200

* OH Transformer limited

peak power density at the divertor target scales as $\sim 1/n$ [5] (here n is the plasma density). The designs presented here operate at relatively high densities and thus may provide advantages for divertor design.

4 Concluding remarks

When the performance of tokamak design options are evaluated a variety of issues need to be addressed. These include: the expected physics performance characteristics, engineering reliability and flexibility, cost effectiveness, and time frame for construction. However, not only the quantitative issues outlined above need to be addressed but the influence of the experiment on the overall fusion program must be evaluated.

The resistive magnet designs presented in this paper could be quite advantageous when evaluated by the above mentioned criteria. Their performance, in terms of the required enhancement factor (Table 3), is substantially higher than the performance of the BPX design, and it is comparable to the performance of the present ITER design. However, the estimated cost of the resistive magnet tokamak options could be substantially lower than the cost of the present ITER design, while the increase in cost over

the present BPX design may not be prohibitively high. Also, in designing and building the resistive magnet machines there is less need for making strong engineering extrapolations or for performing extensive R&D in order to verify the required technologies.

The issues of increased cost effectiveness, improved engineering performance and reliability, greater robustness, and faster schedule without loss of physics performance could make the resistive magnet tokamak designs attractive for a major next step in fusion research. A successful long pulse ignition experiment would pave the way for the development of engineering oriented reactors within the framework of international collaboration, and could facilitate development of an electricity producing pilot plant.[6]

Acknowledgments

References

[1] ITER Conceptual Design: Interim Report. Technical Report 7, IAEA, Vienna, 1990. ITER documentation series.

[2] D. R. Cohn. High Gain Requirements and High Field Tokamak Experiments. Technical Report PFC/JA-90-6, Massachusetts Institute of Technology, 1990.

[3] L. Bromberg, D.R. Cohn, J.E.C Williams, and D.L. Jassby. A Long Pulse Ignited Test Experiment (LITE). Technical Report PFC/RR-82-18, Massachusetts Institute of Technology, 1982.

[4] ITER Physics Design Guidlines. Technical Report 10, IAEA, Vienna, 1990. ITER documentation series.

[5] P. J. Harbour. Thickness of the Scrape-Off Layer of a Large Tokamak with a Large Poloidal Divertor. *Nuclear Fusion*, 24(9):1211, 1984.

[6] S. Dean, C. Baker, and D.R. Cohn. To be published in the Journal of Fusion Energy.