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A Modular Commercial Tokamak Reactor with Day Long Pulses

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A Modular Commercial Tokamak Reactor with Day Long Pulses

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We discuss the design features of a commercial tokamak reactor in which day long pulses are provided by the volt-second capability of the ohmic heating transformer. Illustrative parameters are a major radius of 9.7 m, a minor radius of 1.95 m, an average toroidal beta of 0.036, a magnetic field on axis of 6.1 T, a neutron wall loading of 2.3 MW/m^2 and a thermal power level of 4000 MW. The tokamak is modularized into units which consist of two toroidal field coils, blanket and shield and first wall. The removal of the two toroidal field coils associated with a module would be carried out without warming up the rest of the toroidal field coil set.

KEY WORDS: tokamak; fusion reactor, pulsed.

1. INTRODUCTION

Detrimental effects due to pulsed operation and difficulty in the execution of major maintenance operations are key concerns in tokamak reactor design. In this paper, we describe a reactor concept which could reduce these problems. It is a completely modularized device with very long pulses provided by the volt-second capability of the ohmic heating transformer. Operation with very long pulses will greatly reduce thermal and mechanical fatigue and will allow for very gradual plasma startup and shutdown scenarios. A design is considered for a device with a one day pulse length in order to illustrate the potential of the tokamak for very long pulse operation.

2. GENERAL FEATURES

The features of the design concept are strongly driven by maximization of the pulse length provided by the ohmic heating (OH) transformer. A very substantial increase in the space available for the OH transformer is obtained by the choice of a major radius which is somewhat larger ($R \approx 10$ m) than that used in current commercial reactor designs.[1] In addition, the OH transformer is designed to provide maximum flux by increasing the peak field through the use of a Nb_3Sn superconductor. Finally, relatively high temperature plasma operation is employed to reduce the plasma resistance. A relatively large aspect ratio is used to allow for operation at high wall loading without producing an unacceptably large amount of thermal power in the large major radius design.

The entire tokamak system is modularized into units consisting of first wall, blanket and shield, and two superconducting toroidal field (TF) coils. A module could be removed without warming up any of the toroidal field coils. The removal of a module could be accomplished by a relatively small number of operations. Each TF coil would be situated in a separate dewar. Contact between the low temperature region of the magnet and the casing, which is at room temperature, would be made through struts of G-10 insulation. It appears that the heat leak could be held to acceptably low values.[2]

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A relatively simple equilibrium field (EF) coil system would be used. The main EF coils would be external to the TF coils. It may not be necessary to locate any EF coils inside the TF magnet. If EF coils are located inside the TF magnet, pressure contacts would be employed to reduce the time required for assembly and disassembly. The use of an external EF system is facilitated by utilization of a TF coil design which has a reduced size relative to a D-shape or bending free design. The TF coil size is determined mainly by ripple requirements.

Table II. Magnet System Parameters for Illustrative Design

Field at TF magnet	9.4 T
Number of TF coils	20
Ripple at plasma edge	1%
Ripple at plasma center	0.1%
TF magnet stored energy	55 GJ
Peak field at OH magnet	10 T
OH stored energy (0 to 12 T swing)	22 GJ
Volt-second capability of OH transformer	1100 V s
EF stored energy (0 to full field of 0.45 T)	7.0 GJ

3. ILLUSTRATIVE DESIGN PARAMETERS

A rough illustrative design has been developed by extrapolation from a design for a large major radius tokamak reactor with helical coils[2] and from the HFCTR tokamak power reactor design.[3] An engineering scoping study has not yet been performed. Plasma and magnet parameters for the illustrative design are shown in Tables I and II. An elevation view is shown in Fig. 1. The illustrative design has a major radius of 9.7 m, an average toroidal beta of 0.036, a pulse length of 24 hr, and a neutron wall loading of 2.3 MW/m^2.

These parameters should lead to a more attractive reactor concept than the 15-m major radius design which we previously considered.[4] That design had an average beta of 0.017, a pulse length of 14 hr, and a neutron wall loading of 1.8 MW/m^2. The improved features result from a less conservative assumption for the average value of beta, the use of a higher performance OH transformer, higher temperature operation, and an improved calculation of plasma voltage requirements, which gives a prediction of longer pulses.

Table I. Plasma Parameters

Major radius	9.7 m
Minor radius	1.95 m
Aspect ratio	5.0
Plasma elongation	1.5
Average toroidal beta	0.036
Magnetic field on axis	6.1 T
Plasma current ($q(a)=2.5$)	9.7 MA
Plasma density profile	1.3×10^{14} cm^{-3} $(1-r^2/a^2)$
Electron temperature profile	35 keV $(1-r^2/a^2)$
Plasma voltage ($Z_{eff}=1$)	1.1×10^{-2} V
Pulse length ($Z_{eff}=1$)	24 hr
Neutron wall loading	2.3 MW/m^2
Thermal power (fusion power × 1.25)	4000 MW

The single loop voltage requirement for the plasma is calculated assuming that the voltage is constant across the plasma, the safety factor on axis, $q_{axis,}=1$, and $Z_{eff}=1$. The voltage is given by

$$V = 2\pi RE = \frac{2\pi R j_{axis}}{\sigma_{axis}}$$

where E is the electric field, R is the major radius, j_{axis} is the current density on axis, and σ_{axis} is the plasma conductivity on axis. Relating j_{axis} to the toroidal field on axis, the expression for voltage becomes

$$V \simeq \frac{4\pi B_t}{\mu_0 \sigma_{axis}\sqrt{\frac{b}{a}}}$$

or

$$V \sim B_t T_e^{-3/2}$$

where b/a is the plasma elongation and T_e is the electron temperature. For a central electron temperature of 35 keV, the voltage requirement is 1.1×10^{-2} V.

The ohmic heating transformer plus the EF system provide ~1150 V-s, of which approximately 900 V-s is available for driving the plasma current during burn. Thus it should be possible to drive the plasma current for about 80,000 s (approximately one day). The value of $n\tau$ predicted by INTOR scaling ($\tau \sim nab$) is about twice that which is required for ignition.

4. MAGNET SYSTEM

In order to maximize the volt-second capability for a given amount of space, the superconducting

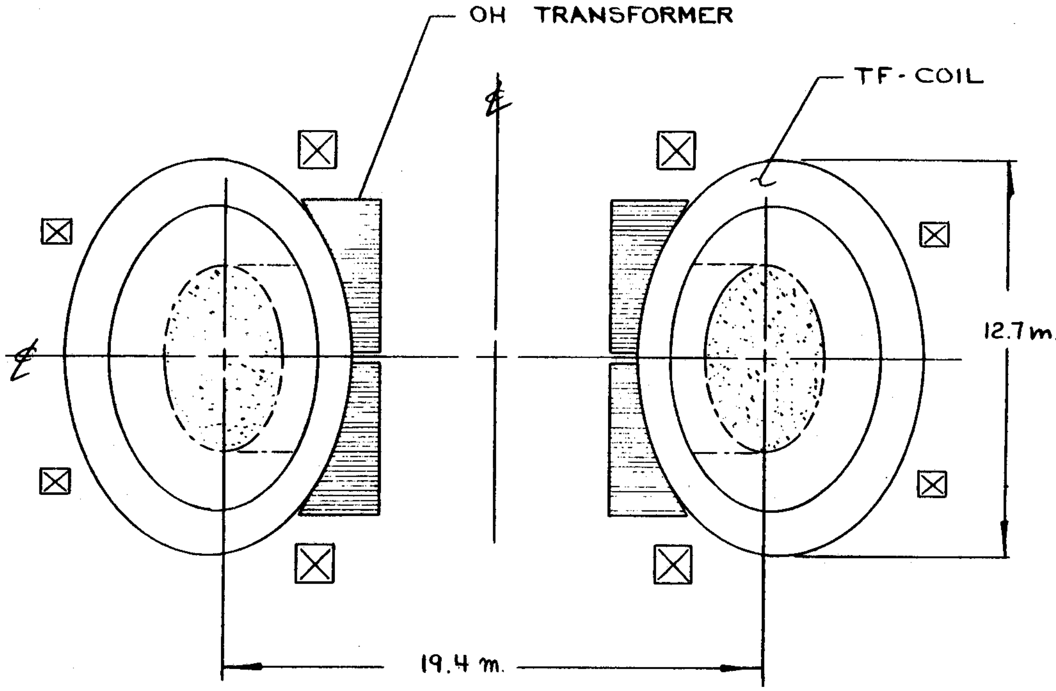

Fig. 1. An elevation view of the reactor.

material for the OH transformer is Nb_3Sn, allowing higher peak fields than NbTi. In addition, it is made of a large number of pancakes that are independently driven, increasing reliability. The field in the OH transformer is swung from +10 T to −10 T. The change in stored energy in a swing from 10 T to 0 T is 22 GJ. Because of the large stored energy, the direction of the plasma current and the magnetic field from the equilibrium field magnets would be reversed after each pulse (D. L. Jassby, private communication, 1977). If this procedure is not followed, either a relatively long time would be needed to recock the transformer or very large power would be required. A second ohmic heating transformer would be employed if a relatively fast initial current start-up scenario is desired.

The toroidal field magnet system consists of 20 coils of approximately circular shape.(5) The size of the coils is determined by the requirement for acceptably low ripple. In the illustrative design the ripple at the plasma edge is 1% and the ripple on axis is 0.1%. The peak field on the TF magnet is 9.4 T. The use of Nb_3Sn would be advantageous in that the sensitivity of the conductor to pulsed fields would be reduced.

5. RF AND/OR NEUTRAL BEAM SYSTEMS

Approximately 75 MW of auxiliary heating power would be provided for start-up heating. In addition, there may be significant advantages in using RF or neutral beam current drive during the start-up phase. The use of noninductive current drive could remove the need for rapidly changing magnetic fields in the OH transformer, thus reducing requirements on the OH magnet system and the first wall design. RF or neutral beam current drive may also be important in the prevention of disruptions by current drive profile control.

6. IMPURITY CONTROL

Because of its simplicity, a pump limiter is the preferred method of impurity control.(1) However, erosion and difficulty in handling severe heat loads may preclude its use. Hence, the use of a poloidal divertor or a bundle divertor must also be considered.

7. SUMMARY

Operation with very long pulses from the OH transformer provides an alternative to commercial tokamak reactor concepts which use steady state current drive.[1,6] It appears that day long pulses from the OH transformer could be provided in a tokamak reactor design which is not greatly increased in size relative to current commercial reactor designs.[1] Key features in obtaining this capability are a moderate increase in major radius, the use of a high performance OH transformer, and operation at moderately high temperatures. Reactor designs with shorter pulses could also benefit from these features. It may also be possible to completely modularize the reactor thereby facilitating ease of assembly and disassembly for major maintenance and repair operations. An area of major concern is that of impurity control and plasma positioning if a pump limiter is not viable. An engineering study is underway to assess the viability of this overall design approach.

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REFERENCES

1. C. Baker et al., STARFIRE—commercial tokamak fusion power plant study, Argonne National Laboratory, ANL/FPP-80-1 (1980); M. A. Abdou et al., STARFIRE—a conceptual design of a commercial tokamak power plant, in *Proc. 8th Int. Conf. Plasma Physics and Controlled Fusion Research*, Brussels, Belgium, July 1980, IAEA-N-38/E-1 (1980).
2. R. E. Potok et al., Physical and engineering constraints for tokamak reactors with helical coils, MIT Plasma Fusion Center Report RR-82-10 (1982).
3. D. R. Cohn et al., High field compact tokamak reactors (HFCTR), MIT Plasma Fusion Center Report PFC/RR-80-3 (1979); J. Schultz, F. S. Malik, D. R. Cohn, and J. E. C. Williams, Advantages of high force density magnets and total modularization, MIT Plasma Fusion Center Report PFC/RR-80-3 (1979).
4. D. R. Cohn, J. E. C. Williams, and L. Bromberg, The low beta pulsed tokamak as a modular commercial reactor with very long pulses, MIT Plasma Fusion Center Report, PFC/JA-82-13 (1982).
5. E. S. Bobrov, Stresses in superconducting toroidal magnets under the action of out-of-plane loads, ASME paper 82-PVP-52 (June 1982).
6. S. Y. Yuen, J. H. Schultz, D. Kaplan, and D. Cohn, Design features of tokamak power reactors with RF-driven steady state current, MIT Plasma Fusion Center Report RR-79-22 (1979); S. Y. Yuen, D. Kaplan, and D. R. Cohn, *Nucl. Fusion* **20**:159 (1980).