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CONCEPTUAL DESIGN SUMMARY FOR MODIFYING DOUBLET III TO A LARGE DEE-SHAPED CONFIGURATION

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CONCEPTUAL DESIGN SUMMARY FOR MODIFYING DOUBLET III

TO A LARGE DEE-SHAPED CONFIGURATION*

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ABSTRACT

The Doublet III tokamak is to be reconfigured by replacing its indented (doublet) vacuum vessel with a larger one of a dee-shaped cross section. This change will permit significantly larger elongated plasmas than is presently possible and will allow higher plasma current (up to 5 MA) and anticipated longer confinement time. Reactor relevant values of stable beta and plasma pressure are predicted. This modification, while resulting in a significant change in capability, utilizes most of the existing coils, structure, systems and facility.

INTRODUCTION

Since its commissioning in 1978 Doublet III has been dedicated to the investigation of noncircular cross section plasmas. In that time the device has demonstrated the ability to produce and study stable plasmas of reactor regime densities in a variety of plasma shapes, including doublets, as well as circles, dees and divertor configurations in the upper lobe of the indented vacuum vessel. Experiments over the last year have concentrated on neutral beam heating of plasmas in the upper lobe and have demonstrated the importance of plasma current and cross section elongation in obtaining high beta and energy confinement time. Enhancement of the confinement time in diverted plasmas has also been observed.

In 1985 Doublet III will be partially disassembled and its indented vacuum vessel replaced with a new, larger vessel with a dee-shaped cross section termed Big Dee, increasing the plasma volume for dee-shaped and diverted plasmas by almost a factor of three (Fig. 1). The design value of the plasma current (which appears from presently observed scalings of plasma parameters to be the most important determinant of performance potential) is 5 MA. The magnetic geometry (chosen to maximize theoretical projections for beta) and very large plasma currents lead to expectations of high performance.[1] This should allow attainment of reactor relevant values of the plasma pressure in a moderate size device with modest auxiliary heating. The high degree of flexibility in the design of the new configuration will allow for experiments such as optimization of plasma beta and total pressure in terms of elongation and triangularity and a comparison of pumped limiters and poloidal divertors.

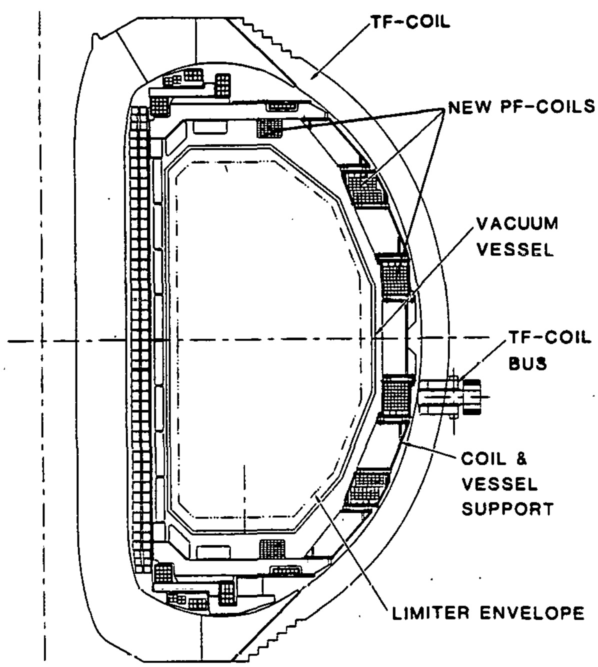

Fig. 1. Big Dee cross section

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By exploiting the inherent capabilities of the existing facility, the modification of Doublet III to a large dee-shaped tokamak will result in a device with performance capabilities needed for realistic modeling of tokamak reactor operating scenarios. In the presently conceived U.S. fusion program plan, Doublet III in the Big Dee configuration and TFTR will be the two principal tokamak research facilities in the late 1980's. Information and experience gained from the two programs will greatly influence the choices to be made for the next crucial step toward a fusion reactor.

DESIGN CRITERIA

Doublet III was designed with an upgradable capability. The 24 dee-shaped toroidal field (TF) coils are capable of 4.0 T operation at the present vessel axis, with an ohmic heating flux swing of 10.5 V-s. The poloidal field (PF) coils were likewise designed and built for the full 10.5 V-s flux swing. Present and near term operation is limited to 2.6 T on axis and 5.6 V-s by existing power and toroidal coil prestressing equipment. While the volt-second limitation has allowed doublet plasmas with currents as high as 2.2 MA and dees to 1.1 MA (1.5 MA by 1984), the present coils and vessel were designed for 5 MA doublet plasmas.

By late 1984 three 80 keV neutral beam injectors will have been fully tested on Doublet III, each rated at 3.6 MW for 0.5-second pulses. To date 5 MW of power has been injected with two injectors, and record volume averaged betas of as high as 4.5% have been achieved.[2] A fourth injector will be available for installation on the new device, bringing the potential beam heating power to 14 MW.

The Big Dee modification of Doublet III will take advantage of these capabties as well as the other facility systems. While it is not planned to increase the current in the toroidal field coil, the energy dissipation capability of the coil will allow eventual long pulse (10-second) operation at nearly 2.0 T.

The criterion that has guided the design of the new vessel and other associated system hardware is to replace the existing vessel with one of as large a volume as reasonably allowed within the toroidal field coil along with new poloidal field coils to allow 5 MA plasma current. This current is consistent with the 10.5 V-s fully reversed flux capability as limited by stress in the ohmic heating solenoid. Design emphasis is on allowing maximum plasma shaping flexibility and diagnostics access. While Doublet III was not originally designed specifically for long-pulse operation, the device has inherent long-pulse potential due to generous thermal limits. The maximum flux swing can drive a range of plasma currents from 5 MA for 1.5-second flattops to lower values at longer pulses, limited primarily by reasonable plasma loop voltage or the availability of auxiliary current drive. Table 1 lists the parameters at the initial and ultimate design conditions.

TABLE 1
MAXIMUM BIG DEE PARAMETERS

	Initial	Design
Plasma geometry:		
Major radius	1.67 m	1.67 m
Half width	0.67 m	0.67 m
Elongation	2.0	2.0
Aspect ratio	2.5	2.5
Toroidal field (on axis)	2.2 T	2.2 T
Plasma current (1.5 s)	3.5 MA	5.0 MA*
Flux swing	7.8 V-s	10.5 V-s*
Generator stored energy	3.0 GJ	3.0 GJ
Neutral beam power (80 keV)	14 MW	14 MW
ECH power (60 GHz)	2 MW	4-6 MW*
ICRF power (55 MHz)	5 MW*	20 MW*
$\langle\beta\rangle$ @ 2.2 T	5.5%	10%
$\langle nT\rangle$ @ 2.2 T	1 atm	2 atm

*Unfunded.

In determining the design criteria for the new configuration, limits were placed on the operating parameters. Since the crucial engineering parameters are functions of the poloidal beta, β_p, the internal inductance, ℓ_i, and the plasma current, I_p, an operating space was established for these parameters with appropriate limits by specifying the design plasma pressure to be 2 atmospheres ($\langle\beta\rangle \simeq 10\%$ at B_T = 2.2 T). In addition limits of $q > 1.7$ and $\beta_p <$ the aspect ratio (2.5) are applied consistent with operating experience on large tokamaks. Specifying these parameters sufficiently constrains ℓ_i that it no longer need be considered a free variable. Figure 2 shows the operating space for several values of elongation, κ, up to the maximum, 2.0. While stable MHD equilibria with betas in excess of 10% have been predicted, operating above this value at full field is not necessary since beta limits can be evaluated at lower field. Also shown is the 5 MA plasma current limit due to a volt-second limit in the ohmic heating coil. Several applicable key engineering design parameters, especially vessel pressure due to disruption and field-shaping coil current, are determined largely by the plasma pressure. Thus, design limits for these parameters closely follow the constant pressure curves of Fig. 2. Operating above this curve (higher $\langle\beta\rangle$ at B_T = 2.2 T) would result in more stringent design requirements for the vessel, coils, structure and power systems (and therefore higher cost). Present experimental experience and theoretical scaling of beta limits indicates that the design value of plasma pressure is most readily achieved at high I_p and low β_p. This is the motivation for labeling the design operating curve in terms of the point I_p = 5 MA and β_p = 0.6

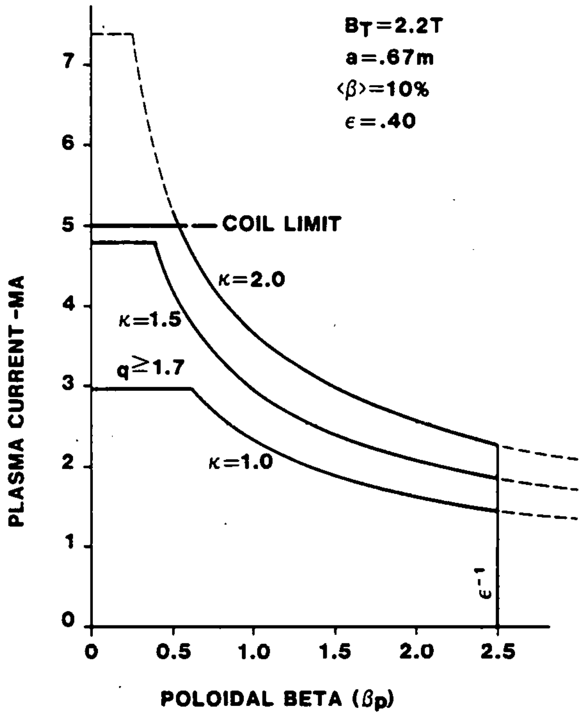

Fig. 2. New vessel operating space

MACHINE DESIGN

In order to meet the stated objectives within the established design criteria, the Big Dee has the following major design features:

1. A new dee-shaped vacuum vessel as large as allowed by the coil systems for the design plasma pressure, and capable of 20 MW, 10-second operation (by upgrade).

2. New poloidal field coils for the design plasma pressure and eventual long-pulse operation.

3. A new support system for the new vessel and coils which maximizes torus access for diagnostics, plasma heating, limiters, etc.

4. Initial internal vessel protection in the form of limiters, beam armor, and divertor armor for 1.5-second operation at 5 MA (2.5 MA for divertor experiments) and short pulse 14 MW plasma heating upgradable to higher power and longer pulses.

5. Necessary power systems additions and modifications to allow a startup plasma current of up to 3.5 MA for a 1.5-second flattop duration (7.8 V-s limit). The systems are upgradable to encompass 5 MA, 1.5-second and long pulse operation up to the energy dissipation capabilities of the coils.

Vacuum Vessel

The objectives for the new vacuum vessel are to obtain: a base pressure of 2×10^{-8} Torr or less; a toroidal resistance of 0.13 mΩ or more; a maximal plasma volume; the capability of withstanding large impulsive magnetic loads due to plasma current disruptions in addition to atmospheric loads; and the capability (with upgrade) of withstanding 20 MW of heating for 10-second operation.

In order to meet these requirements, the vessel is an all welded chamber fabricated of conical and cylindrical sections.[3] The material used is Inconel 625, which has an ultimate tensile strength of 830 MPa (120 ksi) at 250°C and the desirable combination of formability, high as-welded strength, and high resistivity. The walls of the vessel are corrugated sandwich construction (see Fig. 3) except the cylindrical section at the outer midplane which is solid owing to its large ports. The corrugated design achieves high strength and high resistivity. The corrugations are poloidally oriented and serve as passages for cooling water. The vessel's outer radius is 2.46 m, inner radius 0.91 m, and height 2.92 m. It is supported at the outer midplane in four places by guided supports held in place by the support structure. There are approximately 130 ports for diagnostics, neutral beams, rf antennas, limiters, vacuum pumping, etc.

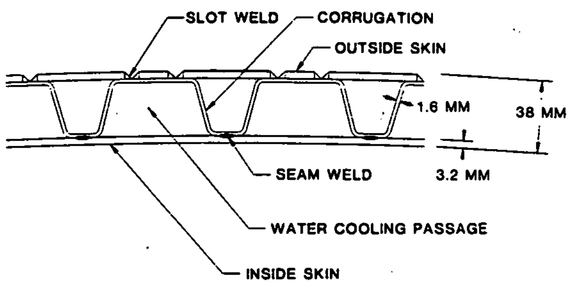

Fig. 3. Vessel wall construction

Two major decisions in the structural design of the vacuum vessel are choosing the skin thicknesses and the spacing between the two skins. Skin thickness is set primarily by the membrane stresses produced at design loads which include the static atmospheric and dead weight loads as well as the dynamic magnetic load due to a plasma disruption. Minimum skin thickness is desirable in order to reduce magnetic error fields and magnetic loads. The distance between skins of the walls determines bending rigidity and is governed primarily by maximum allowable wall bending stresses and structural stability. The vessel design calls for 38 mm overall wall thickness everywhere, except the inner wall which will be 32 mm and the solid outer midplane which will be 25.4 mm. Skin thicknesses are 3.2 mm everywhere, and the corrugations are made from 1.6 mm thick material throughout.

The two primary loadings on the vessel are static atmospheric pressure and magnetic disruption loading which act simultaneously. When the plasma disrupts, it loses its thermal energy in a time on the order of 0.1-1 ms while its current may decay over a considerably longer period. While current decay times on the order of 1 ms have been observed in low current tokamaks, in Doublet III the decay time has generally been in excess of 5 ms. For conservative vessel design, however, an assumption of an instantaneous plasma current decay was made.

The magnetic disruption load is impulsive in nature and acts to implode the vessel, acting in the same direction as the atmospheric loading, but is not uniform, owing to varying poloidal field and vessel resistive effects. The resulting peak pressures on the vessel are 9.6 and 5.6 atmospheres on the inside and outside walls, respectively, at design operating conditions. The intensity of the pressures varies with time.

The static atmospheric loading produces membrane stresses (in the plane of the shell) in the toroidal and poloidal directions. The membrane stresses are compressive everywhere except on the inboard wall where the pressure causes hoop tension. A thin shell of revolution static buckling analysis shows the critical external uniform static pressure to be approximately 30 atmospheres.

Thermal analysis of the vacuum vessel encompasses several areas. During plasma heating a large fraction of the plasma power is transmitted to the wall. At 20 MW of heating power an average heat load of 25 W/cm^2 is applied to the vessel wall, peaking at over 50 W/cm^2 near the outer midplane. This heating results in high thermal stresses due to temperature gradients across the wall. Present engineering analysis shows the vessel capable of withstanding the thermal and disruption stresses without a separate heat shield, except in local areas and the midplane solid band for longer pulses. Necessary thermal shielding will be accomplished by mounting tiles directly to the vessel wall, which will serve as an acceptable heat sink. During discharge cleaning the heat input to the inner wall is greater than that to the outer wall resulting in thermal stresses. A water-cooled thermal barrier between the vessel wall and the inner field shaping coils is planned to control the gradient and maximize the discharge cleaning duty cycle.

The corrugated wall will serve as the primary cooling system for the vessel by flowing water in the spaces behind the inner skin (see Fig. 3). Water cooling in the wall will not be required initially until higher plasma pressure and plasma heating exceeding approximately 14 MW for 1.0 second is reached. The outer surface of the solid midplane will be cooled from the beginning of operations to eliminate surface temperature ratcheting.

Coils

In the present doublet configuration, poloidal field coils for plasma positioning and shaping and for error field correction surround the doublet vessel with close spacing. As the new vessel is larger in diameter, the poloidal field coils along the outer diameter of the vessel must be replaced. Because enhanced plasma access is a goal of the design, the new coils have been located to allow maximum access and still produce high beta plasmas with ample shaping flexibility.

Four new field-shaping coils will replace the eight present outer coils. The coils are located near the vessel corners (see Fig. 1) to maximize vessel access. In addition the four existing ohmic heating coils will be replaced by similar coils located within each new field shaping coil case.

The new coils are designed for 5 MA, 1.5-second (flattop) operation and longer pulses at lower currents consistent with the ultimately available flux swing (10.5 V-s). In order to keep the entire field-shaping coil set consistent in capability, the two outermost coils above and below the vessel will also be changed, to improve their thermal capability.

The coils will be built similarly to the existing (and retained) coils: hollow, water-cooled, copper conductor, insulated with Kapton and fiberglass and vacuum impregnated with epoxy. One significant difference in the design is that the four larger coils will be potted in and remain in stainless steel cases for added coil strength. This allows the coils to be utilized in an integral space-saving support for the coils and vessel. The cases will have toroidal breaks and be isolated from their structural supports.

Support Structure

A new structural system will support the vessel and poloidal field coils. The structure must react coil magnetic loads, vessel dead weight and fault loads, and seismic loads, while providing adequate room for diagnostics, plasma heating equipment, etc.

The design adopted as shown in Figs. 1 and 4 consists of 12 vertical plane post structures located inboard of alternating toroidal field coils (except at the uppermost and lowermost locations where 24 posts are required). The posts attach to the new coils and to the

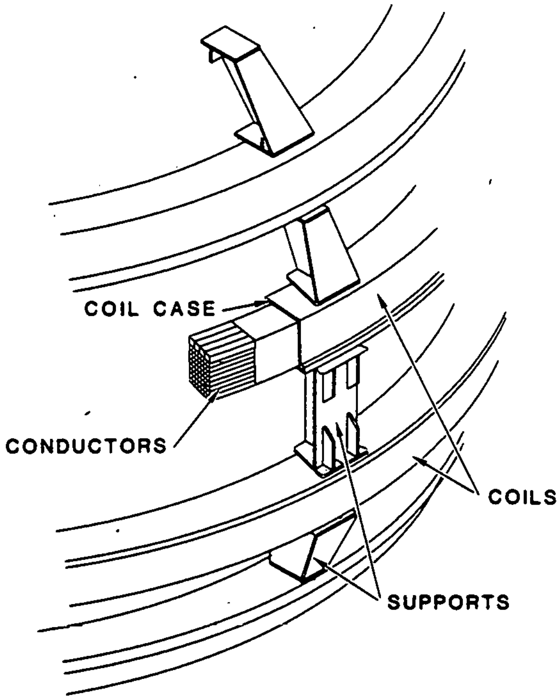

Fig. 4. Coil support structure

existing upper and lower coil support cartwheels. The posts are stainless steel weldments bolted through insulated joints to bosses on the coil jackets. Insulated breaks are provided at joints between coil cases and beams to prevent induced current loops in the structure. Four of the posts at the midplane contain horizonal journals for support of the vessel. The journals allow the vessel to grow thermally without constraint.

Limiters and Armor

A system of limiters and armor will protect the vessel during normal operation and disruptions. Five categories of protection are provided: primary limiters, disruption limiters, neutral beam armor, divertor armor, and thermal armor (described earlier). The systems will be consistent with short-pulse operation at 5 MA and 14 MW of auxiliary heating. Upgrades for higher power and/or longer pulses will be by modular addition. The design for each system is an extension of Doublet III experience.

Two primary limiters will be located at the midplane, 180° apart. Each will consist of a 0.32 m^2 surface of graphite tiles coated with TiC or SiC attached to a water-cooled Inconel blade supported by an adjustable shaft. The limiter will be 0.4 m × 0.8 m with the long axis oriented toroidally. The limiters will be capable of adjustment over a 50 cm range, and will be pumped through the large duct in which the support shaft is located.

Disruption or backup limiters will consist of conduction-cooled, discrete Inconel bars or tiles located poloidally around the vessel at several toroidal locations. These units must absorb the full plasma power for a very short time (milliseconds).

Armor must be installed on both the inner and outer walls of the vessel to absorb neutral beam shinethrough and inadvertent full beam incidence at plasma disruption or for callorimetry (see Fig. 5). In two locations on the outer wall the armor will also be attached to the inner surface of a neutral beam duct. As with the primary limiter, the material will be coated graphite tiles on water-cooled backing plates. The tiles will be able to withstand the shinethrough for up to 1.5 seconds with a subsequent interlock-controlled full beam incidence for up to 10 ms.

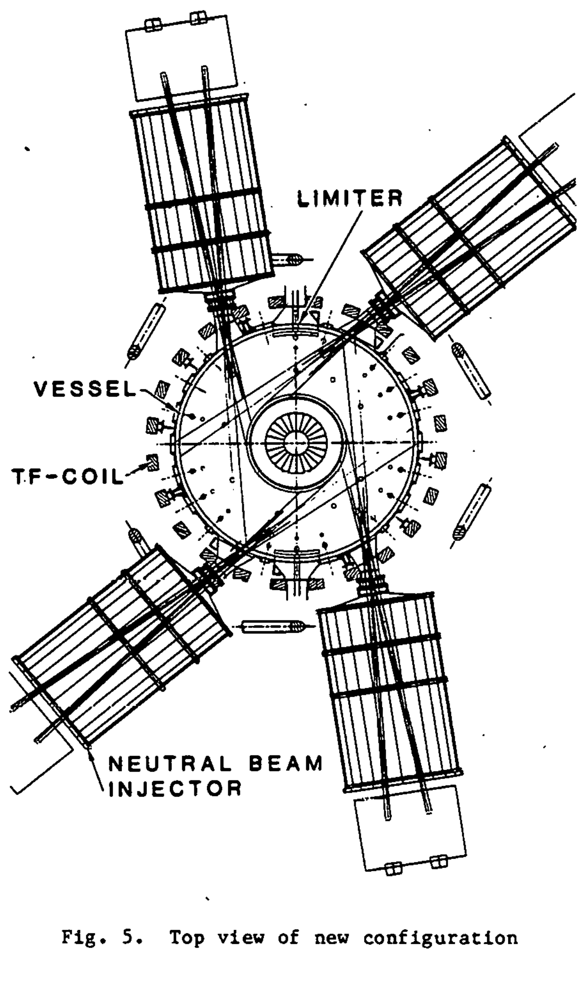

Fig. 5. Top view of new configuration

For divertor operation, armor tiles will be installed in the lower portion of the vessel. Initially these tiles will be conductively cooled Inconel plates. As plasma heating power and pulse lengths increase, the armor will be appropriately upgraded to add graphite front surfaces if necessary.

Coil Power Systems

The existing Doublet III coil power systems provide sufficient coil current and voltage for 2.6 T, 1.0 MA, 1.0-second operation in the present device. Existing motor generator power and stored energy of 815 MVA and 3 GJ, respectively, will be adequate for envisioned upgrades.

The coil power systems for the present device are being upgraded to a 1.5 MA, 1.0-second capability. Further upgrades will be accomplished to support 3.5 MA in the new vessel, including increasing the flux swing to 7.8 V-s and adding more field-shaping coil power supplies. The power supply configurations and plasma control concepts shown to be effective on the present device will be extended to the Big Dee device. The required 1.5-second plasma current flattop at 3.5 MA adds further power system requirements as the experimental time is longer due to added plasma current ramp-up time.

Systems Modifications

Because of the change in vessel geometry and the addition of new coils and power supplies, associated interfacing systems must be modified including neutral beams, diagnostics, fluids, instrumentation and control, and data acquisition.

The neutral beams will be modified for relocation on the new vessel and rotated about their longitudinal axes for optimum beam optics. They will all be installed in the same toroidal direction and oriented for injection parallel to the plasma current at an average 38° from radial measured at the plasma centerline (see Fig. 5). The capability has been maintained in the vessel design to allow two of the injectors to be turned to the counter injection position should this prove attractive at a future date.

Most of the existing diagnostics are usable on the new device but require varying degrees of modification for proper interface and function. The new configuration will permit substantially improved diagnostic access particularly for two-dimensional arrays and scanning.

The new vessel, coils and power supplies, and the relocation of the neutral beam injectors will require interface modifications to the facility vacuum, water, gas, and cryogenic systems. The relocated diagnostics and neutral beams and the new power system components plus the longer experimental time will require appropriate additions to the instrumentation, control and data acquisition systems.

CONCLUSIONS

The modification of Doublet III to a large dee-shaped configuration will provide a powerful and flexible facility for fusion research in the late 1980's. The scope of the changeover is minimized by the inherent capabilities of the existing device, and the design is an extension of the experience gained to date on Doublet III.

ACKNOWLEDGEMENTS

The Doublet III program at GA is a cooperative effort by the U.S. Department of Energy and the Japan Atomic Energy Research Institute. DOE has provided the basic device, support systems and operating funds while JAERI has provided funds to upgrade the initial hardware and to support an additional shift of research operations. It is anticipated that this successful joint effort will be extended into the Big Dee phase of the program.

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