CONF-8004100--1

GA-A15869

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STATUS AND FUTURE PLANS

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MASTER

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Presented at the 4th International Atomic Energy Agency Technical Committee Meeting on Large Tokamak Experiments, April 14-18, Tokyo, Japan.

Work supported by
Department of Energy
Contract DE-AT03-76ET51011

GENERAL ATOMIC PROJECT 3235.803.428
APRIL 1980

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DOUBLET III :
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ABSTRACT

A synopsis is presented of the experimental results from the ohmic heating phase of Doublet III, with emphasis on the production of good target plasmas for the upcoming neutral beam injection phase. The program plan for the device over the life of the U.S.-Japan cooperative program is discussed, as is the status of the preliminary investigation into replacing the present vacuum vessel by one better suited for ETF simulation.

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*Work supported by Department of Energy, Contract DE-AT03-76ET51011.

DOUBLET III : STATUS AND FUTURE PLANS

It is now generally acknowledged that operation with a noncircular plasma cross section will be a necessary feature of a tokamak fusion power reactor. The feasibility of shaping plasmas into vertically elongated configurations and the effects of this shaping on plasma stability and confinement have been investigated at General Atomic Company in a series of noncircular tokamak experiments beginning more than a decade ago that have culminated in the construction and operation of Doublet III, the world's largest operating tokamak. This device is specifically designed to study the effects of noncircularity in high-beta, auxiliary-heated plasmas, ultimately with reactor-like parameters.

The key systems of Doublet III are illustrated in the accompanying cross section (Fig. 1). The plasma chamber is elongated 3 to 1, with a width equal to that of PLT, and is indented; such a chamber allows the production of dee and circular plasmas as well as doublet ones. The toroidal field coils are demountable, a feature which accelerated the construction phase since it permitted simultaneous assembly of the centerpost and the vacuum vessel and attached coils. Furthermore, such a feature adds to the flexibility of the facility since it in principle permits the substitution of an alternate vacuum vessel at some later date, a possibility that will be explored below. The E-coil consists of a central solenoid wrapped around the B-coil centerpost and a number of distributed turns.

There are 24 F-coils, situated in an up - down symmetric fashion which, together with the requisite power supplies, provide considerable experimental flexibility in the production and control of plasmas of a variety of cross sectional shapes. The F-coils are arranged in several parallel-connected

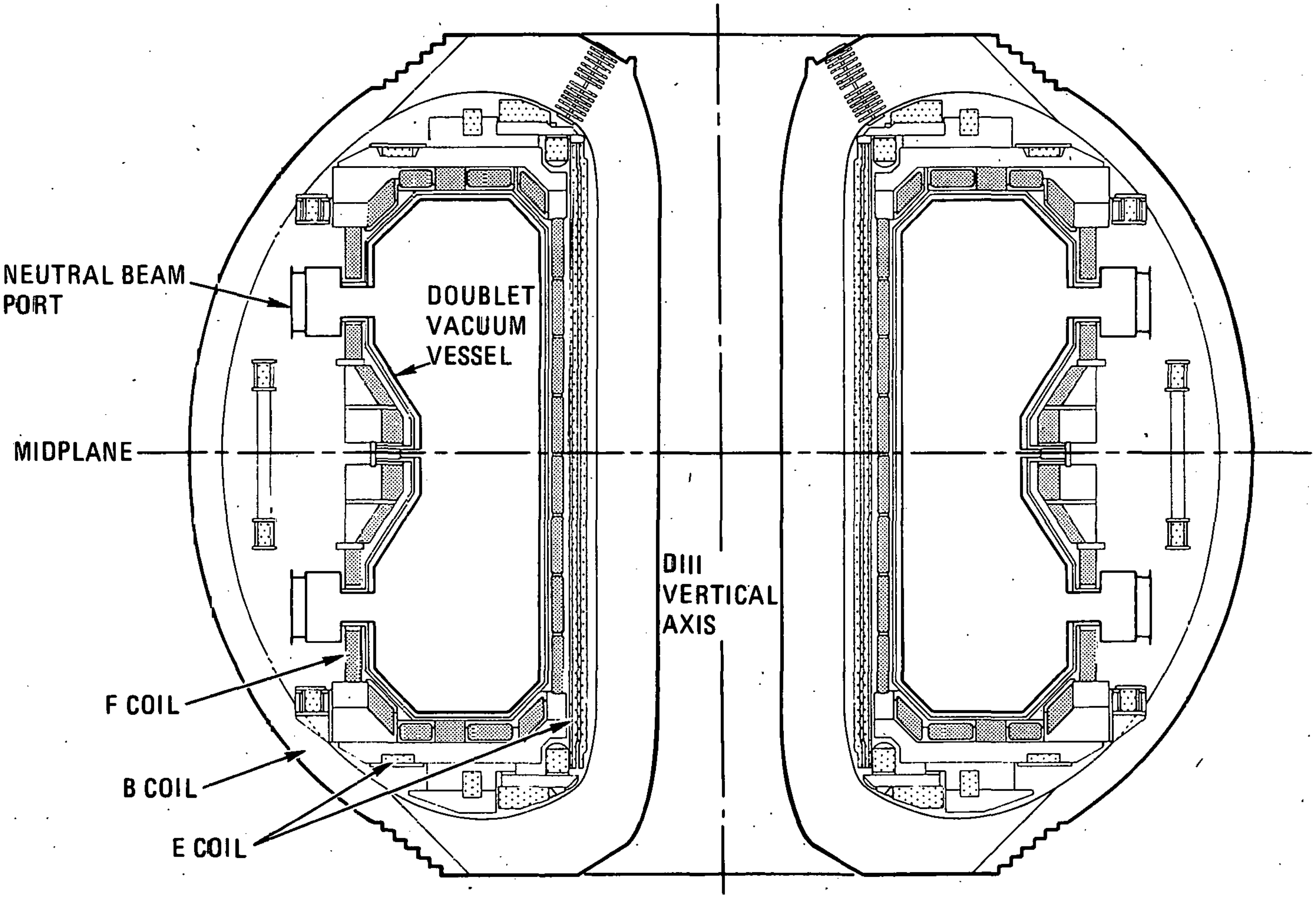


Fig. 1. Cross section of Doublet III

groups, each with a power supply in series. The voltages produced by these supplies determine the coil fluxes, and hence the plasma shape. The parallel connection of the F-coils provides passive stability against axisymmetric motion of the plasma, especially for vertical displacements. The choice of which F-coils the shaping supplies are connected to determines whether the discharge forms only in one half of the vessel (singlet) or symmetrically in both halves (droplet or doublet). The magnitude of the shaping flux applied to the outer midplane coils determines whether the discharge breaks into two independent current channels (droplets) or remains connected, with an internal separatrix (doublet).

The basic machine has been engineered for operation at toroidal fields of 40 kG, a 10 V-s flux swing, and a 5 MA plasma current. However, as presently operated, a number of the auxiliary systems, such as energy storage, power supplies, anti-torque frame, and water cooling, limit the achievable parameters to 26 kG, 5 V-s, and 2 MA. The only source of heating at present is ohmic heating but 2 beamlines, each containing two 80 kV ion sources and delivering 3.6 MW of neutral power to the torus, are scheduled for installation in 1981. There is sufficient access for a total of 6 beamlines.

The basic machine and the first 2 beamlines were funded by the U.S. Department of Energy. The Upgrade Project, which has as its mission the design, procurement and installation of that hardware needed to allow Doublet III to operate at its full engineering capabilities, is being funded under an agreement with the Japan Atomic Energy Research Institute.

In fact, Doublet III is really quite rich in upgrade potential. In addition to those elements specified in the U.S.-Japan Upgrade program, i.e. 40 kG, 7.5 V-s, and 18 MW of neutral beam injection, the demountable coils and the large, low ripple volume they enclose permit the installation of a vacuum vessel that is considerably larger than the present vessel. Furthermore, the large volt-second and energy storage capabilities allow multi-second operation, provided that long pulse ion sources become available.

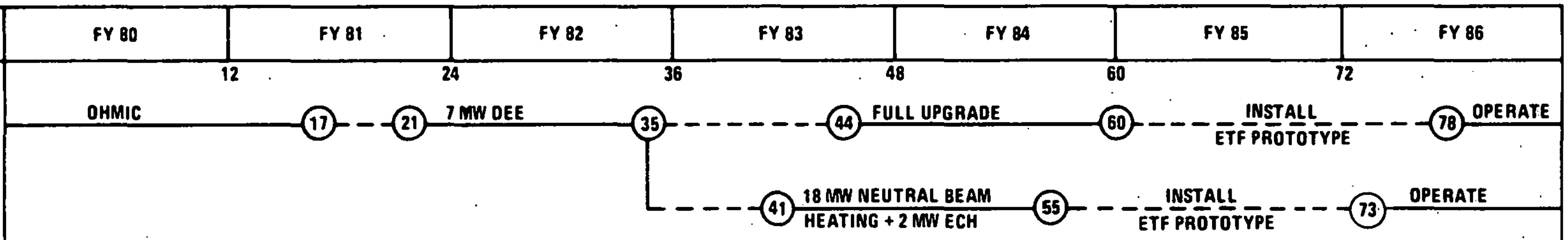
The U.S.-Japan cooperative program also calls for a JAERI scientific team to have 50% of the operating time on Doublet III over the five year term of the agreement. The intent of the visiting team of scientists is to focus attention on experiments in the upper lobe of Doublet III. This will encompass plasmas of vertical elongation ranging from 1 to 1.6; it is to be noted that the bulk of the Doublet III diagnostics is located on the top half of the device. This program, though only begun a few months ago, has already proven very fruitful. Operationally, the two teams have access to the device on alternate weeks on a two-shift per day basis. GA provides engineering and operating personnel to support the Japanese team. The teams participate jointly in the program planning and the resultant close interaction between the two groups of scientists has proven to be mutually beneficial.

The long range plan for the utilization of Doublet III is outlined in Fig. 2. The plan is in general agreement with the current guidance from the U.S.-Japan Doublet III Steering Committee but it does encompass some additional options, specifically ECH heating and the possibility of incorporating a large dee vessel. The overall thrust of the program reflects a balance between two principal ingredients: support of ETF/INTOR and the study of advanced tokamak concepts. The former is achieved essentially through a serial program of producing progressively more reactor-like plasmas by virtue of increases in machine and auxiliary heating capabilities. This aspect of the program is envisioned as ultimately culminating in the installation of a large dee vessel with parameters chosen to simulate ETF start-up. The advanced tokamak program will focus on devising improvements in the tokamak as a reactor concept.

Neutral beam injection experiments are slated to begin in mid-1981 with 7 MW of power into a dee plasma in the top half of the device. A key decision point in the program (at month 35 in Fig. 2) hinges on whether DIII will pursue the option of 60 GHz ECH experimentation. If that is to be the case, the toroidal field upgrade may be deferred in order to focus upon operation at the fields required for resonant ECH heating. In either case, the remaining 3 beamlines will be installed, raising the total injected power to 18 MW.

The highlights of the Doublet III program to date are the production of record plasma currents (1 MA in dee, 2.2 MA in doublet), the observation of

DOUBLET III PROGRAM PLAN



	FY 80	FY 81	FY 82	FY 83	FY 84	FY 85	FY 86
ETF ISSUES	PREPARATION FOR NBI ON DEE PLASMAS ● PLASMA SHAPING ● TARGET PLASMA ● IMPURITY CONTROL ● CURRENT PROFILE CONTROL TOKAMAK PHYSICS ● TRANSPORT ● STABILITY ETF ENGINEERING ● MA PLASMA CURRENTS ● LOW-VOLTAGE START-UP ● EXPANDING RADIUS START-UP		⊥NBI PLASMA CONTROL WITH NBI PARAMETER SCALING WITH NBI $n\tau$ AND β-OPTIMIZATION REACTOR-LEVEL OPERATION		OPTIMIZED DEE PLASMAS ECH CURRENT PROFILE CONTROL ECH ASSISTED LOW-VOLTAGE START-UP ETF LEVEL WALL LOADING		ETF PROTOTYPE OPERATION
ADVANCED TOKAMAK ISSUES	DOUBLET CONFIGURATION SIMPLIFIED DIVERTORS LOW FREQUENCY COUPLING		LOW FREQUENCY HEATING OF DOUBLETS DOUBLET TARGET PLASMA FOR NBI		ECH PHYSICS AND HEATING β-OPTIMIZATION-DOUBLET		RF HEATING ADVANCED IMPURITY CONTROL
INPUT FROM OTHER US EXPERIMENTS	HIGH β-OPERATION (ISX-B) ECH (ISX-B) NBI (PLT, PDX, ISX-B)		β-LIMIT PHYSICS (ISX-B; PDX) POLOIDAL DIVERTOR (PDX) HOT ION CONFINEMENT (PDX) ICRH, LHH (ALCATOR-C, PLT)	TFTR OPERATION PDX-M OPERATION BUNDLE DIVERTOR (ISX-B)	ISX-C OPERATION		

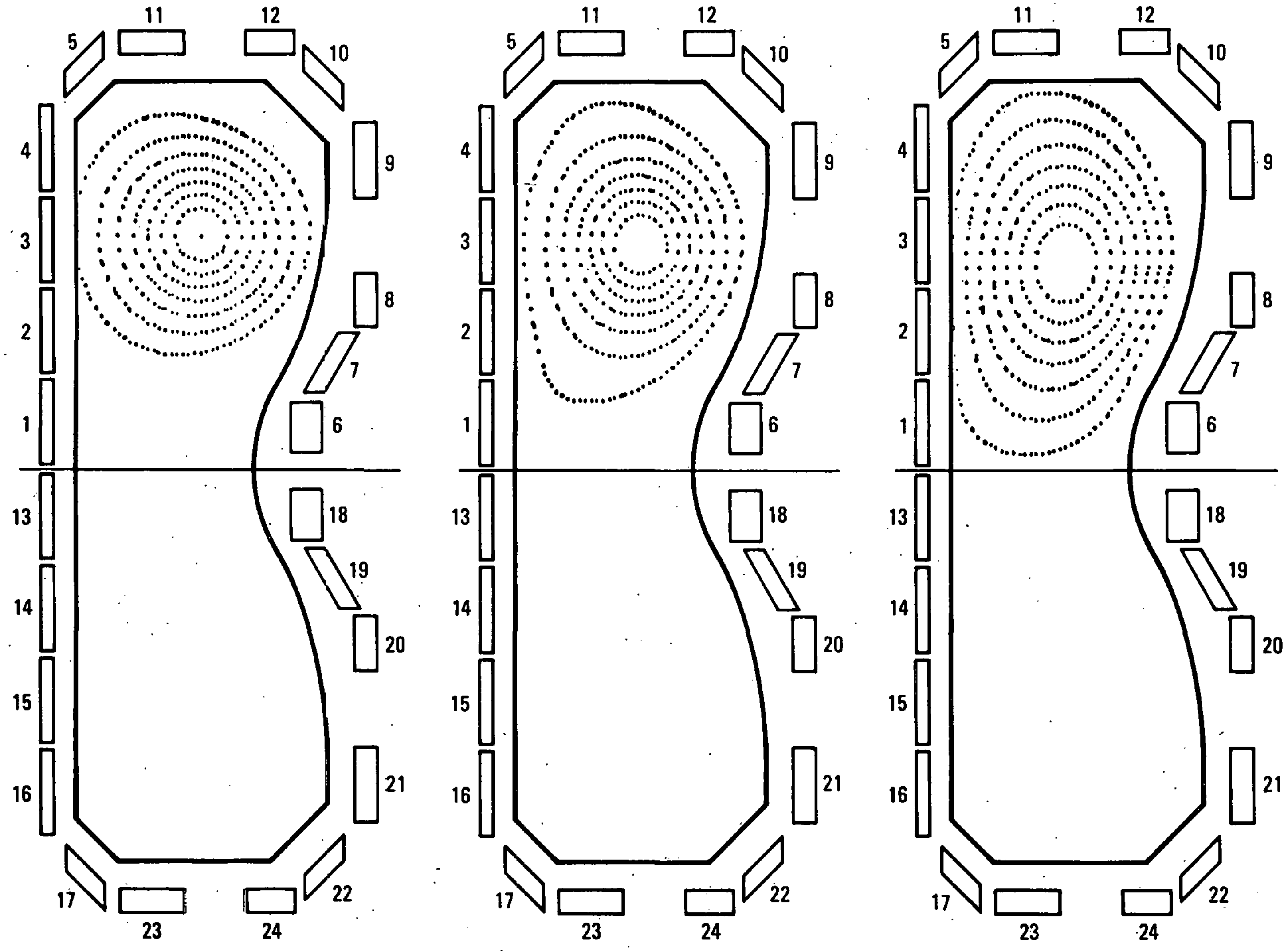
Fig. 2. Proposed Doublet III program plan. Note that FY 80 encompasses the period October, 1979 through September, 1980.

"standard" confinement, density and impurity levels in noncircular plasmas, the achievement of a diverted dee configuration, and the increased reproducibility of doublet discharges, a result that has been obtained by merging two droplet plasmas.

A major emphasis of the experiments thus far has been to produce and control plasmas of a variety of cross sectional shapes. The intricate field shaping coil system has proven to be very valuable in this regard not only for the production of such plasmas but also for tuning desired plasma characteristics, especially low metal ion content, high density, and some types of MHD activity. All configurations generated are characterized by a steady, flattop phase that persists to the 800 ms power supply limit, indicating that discharges of greater duration would be made possible by a power supply upgrade.

The JAERI team has succeeded in demonstrating dee-shaped plasmas of various elongation (Fig. 3) and also divertor operation (Fig. 4). In all cases, such discharges are long lasting and highly reproducible. The key to bringing the separatrix into the plasma to achieve divertor operation is to increase the current I_D in the inboard midplane field shaping coils. This divertor configuration has been achieved both at plasma start-up and later during the course of the discharge. In both cases, there are strong indications of reduced impurity contamination.

Doublet plasmas have been produced in Doublet III both by formation of doublets at discharge initiation, as was done in Doublet IIA, and by merging droplet plasmas after the relatively turbulent portion of the current rise phase is complete (Fig. 5). The second technique has proven more reproducible and has given preliminary indications of producing discharges with a somewhat reduced percentage of the poloidal flux inside the separatrix. In the latter context, the current profiles that characterize the ohmic heating phase have been so peaked that most of the plasma pressure is located on nearly circular flux surfaces. One measure of this peakedness is the amount of flux inside the separatrix, and this is generally in excess of 85% for the plasmas produced to date.



κ = 1.06

κ = 1.37

κ = 1.62

Fig. 3. Dee plasmas of various elongation produced in Doublet III

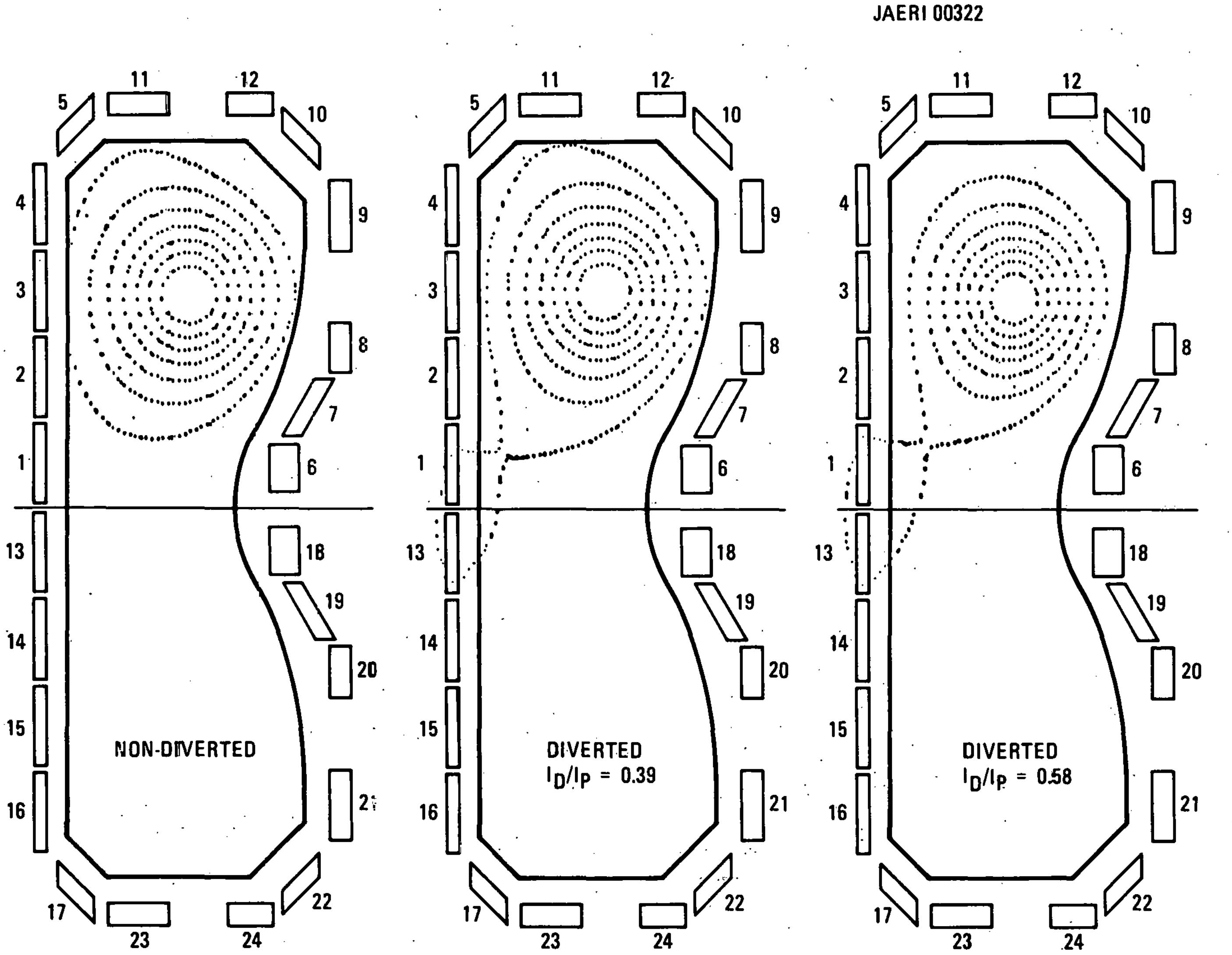


Fig. 4. Divertor configuration in Doublet III. I_D represents the sum of the currents in the two inboard midplane field shaping coils.

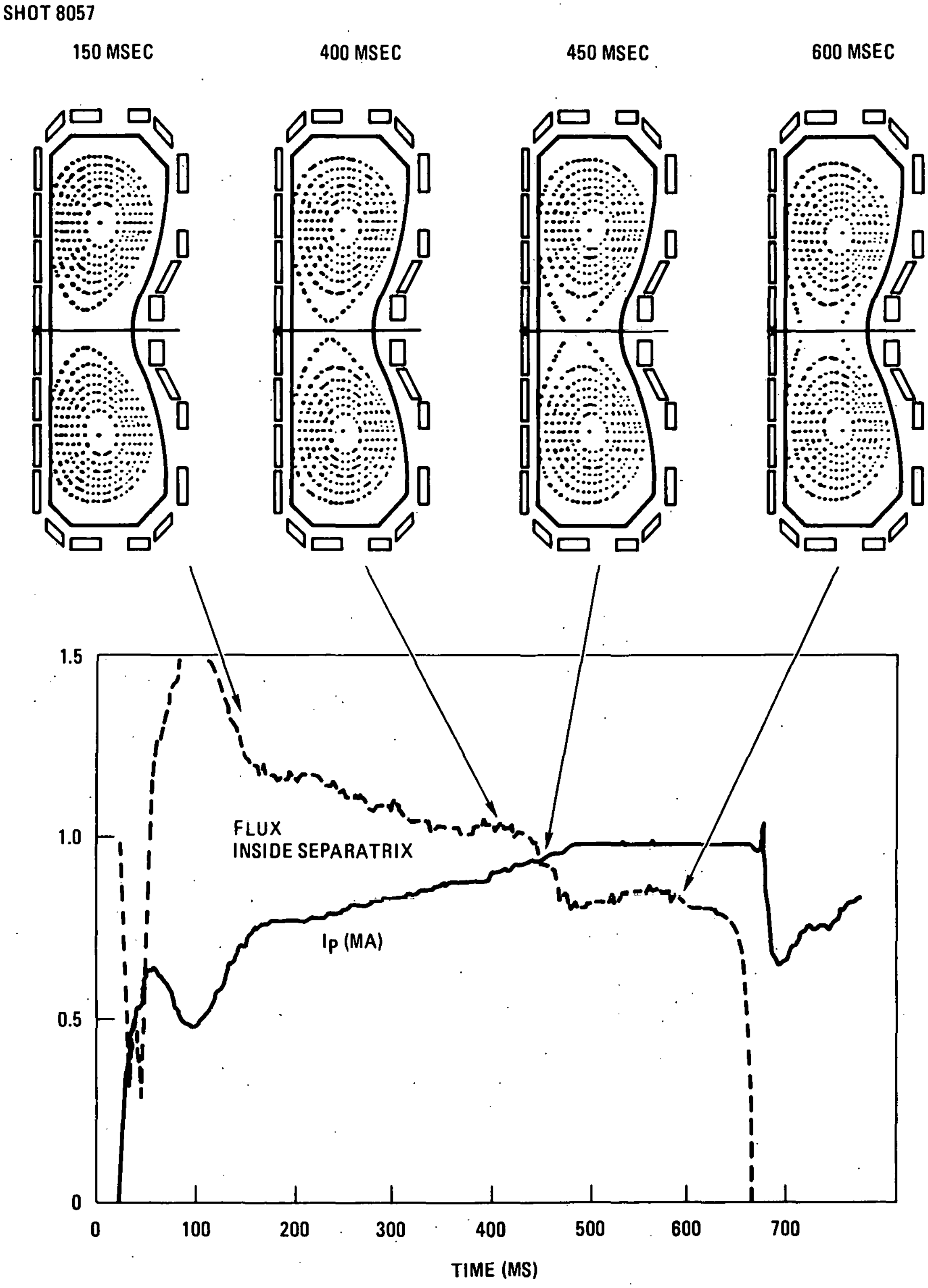


Fig. 5. The use of merging plasmas to produce doublets

Peaked current profiles make it very difficult to quantify the effects of elongation on plasma behavior. In particular, one would expect transport phenomena and pressure driven MHD modes to be dominated by the region of large plasma pressure and the geometry in this region differs little from that of a circular plasma for the case of a peaked current profile. A number of techniques are under study to attempt to broaden the current profile: carefully controlled current ramping, low q operation, noble gas injection to try to produce a hollow current profile, and low frequency heating to couple energy to the separatrix region.

The principle impurity contaminants in Doublet III discharges are oxygen and two constituents of the vessel, nickel and chromium. The techniques that have been found most useful in reducing impurities are gas puffing, the details of the current rise waveform, the triggering of MHD oscillations, and titanium gettering. With these techniques, discharges have been achieved that are characterized by $Z_{eff} \leq 2$ and a very low central radiated power density, i.e. less than 0.02 W/cm^3.

The maximum density achieved has gradually increased with greater operating experience. The highest line averaged densities, nearly 10^{14} cm^{-3}, have been achieved in titanium-gettered, moderately elongated dee plasmas. This corresponds to a Murakami scaling constant, defined as $n(10^{13}\ \mathrm{cm}^{-3})\ R(m)/B_T(T)$, equal to nearly 6, which is unsurpassed in tokamak experiments. Reproducible plasmas of this density will form the ideal target for the 80 keV neutral beam injectors.

The considerable imprecision in the ion temperature data makes it difficult to confidently quote a result for the global energy confinement time. The data is consistent in magnitude with an expression of the form

$$\tau_E = 5\text{-}6 \times 10^{-19}\ \bar{n}_e\ a^2 \quad ,$$

although the systematic dependence on density has not been quantified. This expression is virtually identical to that of the fit to the PLT ohmic heating data.

Turning now to the long range plans for how best to make use of the resources that constitute Doublet III, it is valuable to take note of the recent R&D assessments for INTOR and ETF in that they have identified the major hurdles to be faced in proceeding with the next generation of tokamaks. Of principal concern in these assessments are the myriad questions surrounding reactor start-up, broadly defined here as the full evolution from a cold plasma to a self-sustaining condition. It is entirely possible that this set of questions could dominate the machine time for the first several years of such a device in order to improve plasma performance to the point needed for it to carry out its basic mission. Such an eventuality would represent a serious misuse of resources and would also result in a slippage along the critical path to fusion power.

The program in place will certainly make inroads on these questions, but there is much to be gained from dedicating specific devices to simulating individual facets of reactor operation. Provided this is accomplished in a time frame that is not so remote as to preclude impacting design issues, such devices will, as a matter of course, contribute to the solution of design optimization questions as well as the operational optimization questions alluded to above.

With the additional hardware to be supplied under the U.S.-Japan Cooperative Program, Doublet III, as modified by replacing the existing vacuum vessel, provides an ideal test bed for the simulation of the presently envisioned ETF/INTOR start-up scenarios. Thus, efforts are currently directed toward exploring possible modified versions of Doublet III as one option for the long range plans of the facility. But instead of first exhibiting a design and then indicating what features are particularly well adapted to such simulation, it is a useful exercise to turn the question around and ask what are the desirable characteristics of an ETF/INTOR simulator. Demanding that such a device accomplish its objective at a reasonable cost on a time scale relevant to ETF/INTOR needs dictates a hydrogen tokamak constructed with conventional technology. Table 1 outlines a chain of reasoning built

TABLE 1

DESIRABLE CHARACTERISTICS OF AN ETF SIMULATOR

<table>
<tr><th>PROPERTY</th><th>BASIS FOR QUANTIFYING</th><th>DEDUCED VALUE</th></tr>
<tr><td>plasma size</td><td>provide size scaling for dee plasmas
<table>
<tr><th>device</th><th>ISX-B</th><th>PDX, DIII</th><th>Simulator</th><th>ETF</th></tr>
<tr><td>minor radius</td><td>26</td><td>45</td><td>75-80</td><td>130</td></tr>
</table></td><td>75-80 cm</td></tr>
<tr><td>toroidal field</td><td>desire for low voltage (ECH-assisted) startup and limitation of gyrotron technology to 60 GHz</td><td>~ 20 kG</td></tr>
<tr><td>major radius</td><td>simulate ETF density rise scenario by matching value of B_T/R</td><td>~ 1.9 m</td></tr>
<tr><td>volt-second capability</td><td>explore current buildup to several megamps</td><td>~ 10 V-sec</td></tr>
<tr><td>auxiliary heating power</td><td>neutral beam power sufficient to provide wall loading comparable to that of ETF at ignition</td><td>~ 20 MW</td></tr>
<tr><td>beam energy</td><td>heating of plasmas with $n\tau$ in ETF range requires penetration capability similar to that of ETF</td><td>75-85 keV/nucleon</td></tr>
<tr><td>energy storage</td><td>demonstration of control of high beta plasmas requires ~ 10 sec operation</td><td>2500-3000 MJ</td></tr>
</table>

around our current concept of how these first generation reactors will operate to deduce the machine parameters of such a simulator.

With the 5 beamlines, the volt-second upgrade and the new motor-generator set prescribed by the current guidance of the U.S.-Japan Doublet III Steering Committee, the vessel shown in Fig. 6 closely replicates the device characteristics deduced as desirable for such an ETF/INTOR simulator. Specifically, the machine characteristics are

$a = 80$ cm	$B_T = 21$ kG
$\kappa = 1.8$	$I = 3$ MA
$R = 178$ cm	$P_b = 18$ MW

The device employs the present Doublet III toroidal field coils and centerpost in order to optimize cost and schedule considerations. The timetable for completing this project is consistent with that shown in Fig. 2, although it is conceivable that this schedule could be accelerated to some degree.

This device exhibits a marked improvement in performance predictions compared to those of existing devices. The small aspect ratio and the increased vertical elongation lead to the expectation of very good MHD stability properties. In fact, equilibria with beta exceeding 14% have been found which are stable to all ideal MHD modes. Such a regime of beta will be accessible with 18 MW of power provided empirical scaling is valid. The reason for this accessibility is that the low aspect ratio allows trapping of the beam at low fields, a conclusion deduced from Murakami scaling for the plasma density. The large plasma size results in improvement in confinement but does carry with it the disadvantage that multi-second operation will be needed to achieve peak parameters.

Such a device represents a logical extension of the Doublet III program, an accurate simulation of ETF/INTOR start-up scenarios, and it allows access to new regimes of operation characteristic of long pulse, reactor-grade plasmas. It will also provide valuable guidance to the ETF and INTOR programs through its ability to influence design decisions and, by virtue of shortening the learning process involved, in achieving full device utilization.

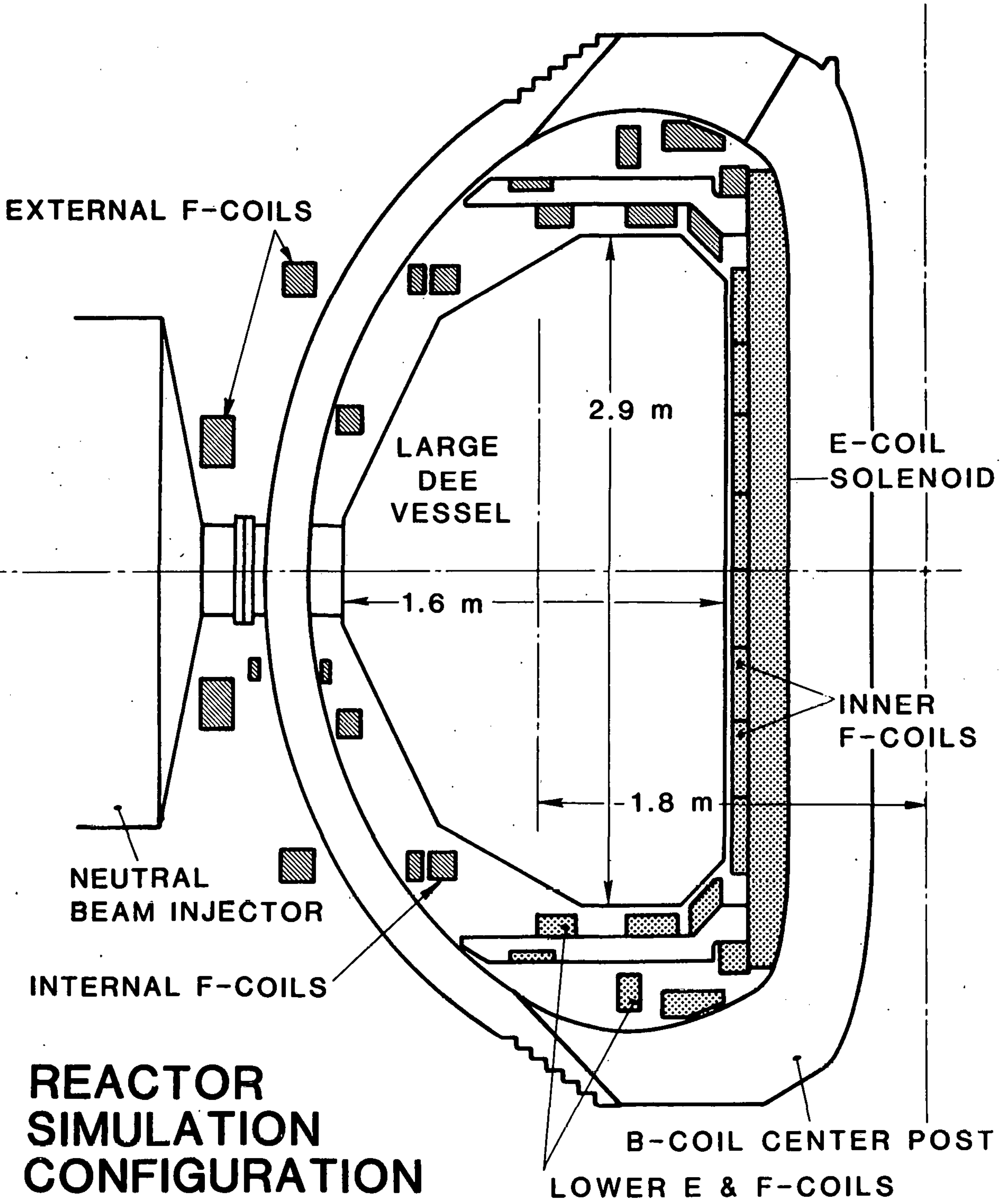


Fig. 6. Large dee vacuum vessel using present Doublet III toroidal field coils and centerpost.

ACKNOWLEDGMENTS

This report summarizes the work of the entire Doublet III team. Special thanks are owed to Drs. J. Gilleland, A. Kitsunezaki, F. Marcus and J. Wesley for reviewing the manuscript.