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A DESCRIPTION OF THE SPHERICAL TORUS MACHINE IN THE FUSION DEVELOPMENT FACILITY

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The mission of the Fusion Development Facility is to move sequentially through the major fusion objectives of advanced confinement physics, burning plasma and DT physics, blanket and other fusion nuclear technology development, tritium self-sufficiency, and perhaps a chance at net electric breakeven. A single ST that has a chance to meet all of these goals in a small, modest cost device will of necessity use the most advanced physics we can expect. To be an acceptable burning plasma device, we need operating scenarios with Q_{plasma} of five or greater. To be suitable as a blanket development facility, we need peak neutron fluxes at the first wall of over 1 MW/m^2 and the capability to run very long pulses (~1 day). To demonstrate tritium self-sufficiency, the device must have a very high duty cycle or in fact be a steady-state device. The net electric breakeven goal is self-defining.

The general logic of a fusion development roadmap based on an ST Fusion Development Facility (FDF) have been described in Ref. [1]. The computational methods for deriving the parameters of an ST suitable for carrying out the mission of the ST-FDF are given in Refs. [2] and [3]. The purpose of this note is to present the parameters of a suitable ST device and to examine the robustness of its missions against downsides in the physics outcomes.

General Aspects of the Device Design

The steady-state and high performance aspects of the device are coupled. A toroidal magnetic field in the 2–3 T range is needed to provide the necessary fusion power output. Because there is so little space in the inboard region, a toroidal coil of that capability essentially precludes an OH coil. Hence the device must be committed to full non-inductive startup, rampup, and current sustainment. The device will be intrinsically steady-state although the first phases of its operation will probably be in 60 second pulses (Ref. [1]).

The ST achieves high fusion performance in small size by discarding inner bore components. Because the centerpost is a small area at low aspect ratio, it is not necessary to breed tritium on the inner wall to have a TBR > 1. Also the fusion power intercepted on the centerpost region can be neglected in the power balance. Hence no blanket assembly need be on the inner bore side. The ARIES-ST design [4] achieved adequate tritium breeding at aspect ratio 1.6. Above some aspect ratio, the centerpost area will intercept enough neutrons that one must begin to supply at least a partial breeding blanket on the inner side or make the major machine concept bifurcation transition to a full blanket and a superconducting toroidal coil. This transition point has not been accurately worked out but lies somewhere around A ~ 2, perhaps as low as A ~ 1.8.

The TF centerpost is a water cooled copper rod. The ARIES-ST design provided 20 cm shielding to the copper TF coil, which was adequate for a Class C waste rating after two full power years. In the ST-FDF, enough fluence to damage the centerpost is not produced until very late in the project [1], perhaps 17 years into the project and after the physics, burning plasma, and blanket development phases. Hence we consider the ST-FDF to be designed with an essentially unshielded centerpost. The TF centerpost will of course be designed to be replaceable. Initial NSTX experience has shown that a machine so designed makes a surprisingly easy and frequent use of the removability of its centerpost even for maintenance. Space in the inner bore that is not devoted to making the TF field or to fusion power producing plasma has a very detrimental effect on overall device performance. We have made an allowance in the calculations herein for 10 cm space between the TF coil and the inner side of the plasma. If possible, it will be attractive to engineer the inner first wall as an integral unit with the TF centerpost and to use the TF coil as a heat sink for the inner limiter.

The fact that the toroidal coil is copper and therefore fully demountable enables many interesting approaches to maintainability. The ARIES-ST design [4]removes the interior of the machine while fixing the external structure. Other proposals envision withdrawing the outer legs of the TF coil, crane lifting out the axisymmetric PF coils and blanket assemblies and the vacuum vessel, leaving the centerpost. Hybrid schemes are possible in which sectors of TF coils and vessel segments are withdrawn, enabling access the interior. The main point is the entire machine comes apart; there are no interlinked windings that cannot be disconnected.

We have assumed a straight centerpost whose height is from X–point to X–point. We have calculated the centerpost dissipation and assumed that the outer return legs, which begin with flaring at the X–point, can be designed with sufficient cross-section that the Ohmic dissipation in them is 50% of the centerpost dissipation. Cooling the centerpost is not a constraint; the largest water temperature rise calculated in any case herein is 50°C. Stress is unlikely to be a problem since the largest field at the TF surface in any case herein is 12.7 T. In some of our alternate scenarios discussed below we make up for lower physics performance by turning up the toroidal field.

The single turn centerpost requires an unusual power supply geometry in which the return legs of the TF coil are fed in parallel from separate power supplies, with the current summing in the centerpost. The maximum centerpost current in any case herein is 20 MA and the voltage drop on the centerpost is <7 Volts. Most effective scenarios are at about 14 MA. We have envisioned 12 return legs and so the device needs individual power supplies capable of perhaps as much as 1.8 MA at about 10 Volts. Such supplies are unusual but possible.

The ripple from the rather distant 12 return legs is negligible; one has the freedom in the design to remove the return legs as far as one wishes, since all the PF coils are inside the TF coil. The actual TF ripple will come from the lack of matching in time of the currents in the independently powered return legs. Current regulation of 0.1% is needed and must be provided from the ac side.

Massive leads are needed to connect the power supplies to the TF return legs. In order to keep the Ohmic dissipation in these leads sufficiently low, the leads cannot be longer than about 5 meters. Hence the power rectifiers must closely ring the machine. Neutral beam injectors and other large apparatus would have access to the machine between the rectifiers if they were placed on the midplane. The ARIES-ST design envisioned the rectifiers and leads off the midplane. This aspect dominates the overall layout of the machine. One has to take great care in regard to the error fields arising from such large currents in so few lead conductors approaching the machine.

The device is sized so that operating at its stability limit and with a peak neutron flux at the first wall of 8 MW/m^2 and in steady-state, net electric breakeven is achieved. Although the device is sized for net electric breakeven, achieving that <u>cannot be</u> the goal of the device. The reason for that is that whenever a fusion device defines as its goal any absolute performance level like ignition, Q = 10, 5, or 1, then in the many reviews and forums in which such a project must be defended, the designers will be pressed to provide assurance of that absolute performance objective. The way to provide increased assurance in the design is always to make the device bigger until the project becomes too costly. Hence although we set a target of net electric breakeven if all goes perfectly well, the goal of the machine is the softer goal of providing a suitable blanket development facility.

Plasma Physics Considerations

For the stability limit in these calculations, we used an equilibrium from the ARIES-ST study at β_N = 8.3, κ = 3.0, δ = 0.6, A = 1.6, and bootstrap fraction f_{bs} = 0.995 [5]. This equilibrium is ballooning and kink stable with a wall at 1.1 times the minor radius; a conducting wall can be placed that close. This equilibrium is not the highest performance case existing. Another case [6] has β_N = 8.8, κ = 3.4, δ = 0.6, A = 1.6, and bootstrap fraction f_{bs} = 0.995. These equilibria and all equilibria used in the ARIES-ST study had p' = 0 at the boundary, corresponding to a severe L–mode edge. A case at A = 1.4 exists with a finite p' at the boundary, H–mode edge, which was stable at β_N = 10 [7]. The higher κ = 3.4 might be achievable in the ST-FDF because we intend to use a double layer PF coil system in which normal conducting coils might be placed close to the plasma (behind 20 cm shielding) to enable the kind of shaping capability found in DIII–D while the bulk of the poloidal shaping flux is produced by superconducting PF coils outside the blanket. Hence there is some reserve in our choice of stability limiting case, although the value of $\beta_N = 8.3$ seems enormous in conventional tokamak experience.

We allow ourselves the freedom to pivot about this base case by assuming that the stability limit is represented by $\beta_N = 8.3$ and that β_T and β_P can be traded off freely according to the basic equilibrium relation

$$\beta_T \beta_P = 25 \, [(1+\kappa^2)/2] \, (\beta_N /100)^2 \quad . \tag{1}$$

Since the bootstrap fraction is proportional to β_P, at constant bootstrap fraction, Eq. (1) shows that the fusion power $\left(\alpha \beta_T^2\right)$ will be proportional to β_N^4, showing the enormous sensitivity to β_N. Pivoting about this base case allows quick exploration of parameter space, but each case will eventually have to have its own equilibrium and stability analysis performed.

Stability studies have shown that to implement the trade-off discussed above, one should move from the full bootstrap fraction cases to lower bootstrap fraction cases by adding current at the current peak [6], which occurs well off axis and close to the outer midplane, at $\rho \sim 0.8$–0.9. In the ARIES-ST study, it was realized that only today's conventional positive ion neutral beam technology is needed to penetrate such a small distance into the plasma and drive the additional current at $\rho \sim 0.8$–0.9 [8]. The lower energy (80–120 keV) positive ion neutral beams carry in more momentum than higher energy negative ion beams and so may be able to make enough toroidal rotation for wall stabilization and to make a significant contribution to the radial electric field for transport barrier formation. We have used neutral beam current drive efficiencies at the volume average temperature in these calculations. We have assumed that at least 20 MW auxiliary power will always be applied. If the required current drive power does not exceed 20 MW, then the difference is added in as heating power.

We have assumed equal central electron and ion temperatures of 20 keV and then derived the density from the beta. No case herein exceeds more than 34% of the Greenwald limit density. The temperature assumed is actually above the peak in the fusion reactivity over T^2 but we chose such a high value since current drive efficiency is always a concern. The densities are high enough (1–5×10^{20} m^{-3}) that coupling should insure $T_e = T_i$. The quality of confinement achievable with $T_e = T_i$ is of course an generic issue for reactor scale tokamaks and a subject of current research.

We make no statement herein about whether A = 1.6 is the optimum choice of aspect ratio. Our earlier studies were at aspect ratio 1.4. But now there is a substantial body of stability calculations to draw upon from the ARIES-ST study at aspect ratio 1.6. The choice of optimum aspect ratio depends on how fast the β_N limit and achievable elongation rise as aspect ratio is reduced. If the β_N limit does not rise sufficiently fast then the optimum will run to higher aspect ratio. Very high β_N cases were found at A = 1.6.

Careful revisiting of the stability issues at A = 1.4 and 1.8 will be necessary to pin down the optimal choice.

Our early papers, which were aimed at assessing the possibility of this concept, used idealized pure plasmas leaning directly on the centerpost. In the cases herein, we have assumed 10% helium concentration and a 1% concentration of neon as a radiator. Actually, the radiator would almost surely be krypton, in which case the proton defect would not be as large as we have calculated with the neon assumption. Nevertheless, these two assumptions result in a Z_{eff} = 2.4 and a DT fraction of 70%, which gives a penalty in the fusion power of a factor of 2 compared to the pure plasma case. The 10 cm inboard space allowance further substantially reduced the fusion power from the idealized case.

The Base Net Electric Case

The base case is Column A in Tables 1, 2, and 3. The size was chosen to obtain the plant Q of 1.12 and 25 MW net electric with a peak neutron wall loading just under 8 MW/m^2.

The neutron wall loading is highly plentiful in regard to blanket testing. The machine is small, the major radius is only 1.12 m. and the minor radius 0.7 m. The total fusion power is 517 MW. We chose an operating point of 90% bootstrap current to have some control although full bootstrap equilibria exist. This choice of bootstrap fraction then gives β_T of 54% with β_N = 8.3. The plasma current is 13 MA and the current drive power for 10% of that current is modest. The plasma Q is 25; so the alpha power is five times the auxiliary heating power and the device is certainly suitable as a burning plasma experiment and is for all practical purpose ignited. The total electrical power to run the plant is a reasonable 211 MW.

Based on the ITER-89P L–mode confinement scaling, a high quality of confinement is needed, H = 4.8. In fact a characteristic of all cases herein except Case J which was deliberately created to have H ~ 2 is an H factor of around 5. The first point to make about such a high H factor is that the conventional tokamak confinement scaling laws do not make much sense when evaluated at low aspect ratio, well outside the range of their database. The H–mode scaling law predicts a lower confinement at low aspect ratio than the L–mode scaling law! But nevertheless, the H factor is some sort of guide whether the confinement time needed is high or not.

As was pointed out in Refs. [2] and [3], such high confinement quality is to be expected from vigorous transport barrier formation in the ST. At its high beta, the ST's diamagnetic contribution to the radial electric field is by itself more than adequate to form a transport barrier. No reliance on toroidal or poloidal rotation is needed. Full 1-D

Table 1

		A	B	C	D	E
		Net Electric!	OK Blanket			
	ST FDF Cases	10 cm	No Wall Stabilization			
		Inboard	β_N Down	f_{bs} 0.5	f_{bs} 0.7	Turn up B
A	Aspect ratio	1.6	1.6	1.6	1.6	1.6
a	Plasma minor radius (m)	0.70	0.70	0.70	0.70	0.70
R_o	Plasma major radius (m)	1.12	1.12	1.12	1.12	1.12
κ	Plasma elongation	3.00	3.00	3.00	3.00	3.00
R_{hole}	Hole size (m)	0	0	0	0	0
J_c	Centerpost current density (MA/m^2)	50	50	45	50	60
f_{ramp}	Induct ramp fraction	0.00	0.00	0.00	0.00	0.00
P_f	Fusion power (MW)	516.69	99.63	211.78	164.69	206.58
P_c	Power dissipated (MW)	84.45	84.45	68.40	84.45	121.60
$P_{internal}$	Power to run plant	211.99	174.35	256.84	215.77	250.47
Q_{plant}	Gain for whole plant	1.12	0.26	0.38	0.35	0.38
Q_{plasma}	P_{fusion}/P_{aux}	25.83	4.98	4.51	8.23	10.33
$P_{netelec}$	Net electric power (MW)	25.69	–128.53	–159.42	–140.02	–155.44
P_n/A_{wall}	Neutron power at blanket (MW/m^2)	7.77	1.50	3.18	2.48	3.11
β_T	Toroidal β	0.54	0.24	0.43	0.31	0.24
β_N	Normalized β (mT/MA)	8.30	5.50	5.50	5.50	5.50
f_{bs}	Bootstrap fraction	0.90	0.90	0.50	0.70	0.90
P_{cd}	Current drive power (MW)	13.65	3.97	46.92	19.70	6.86
I_p	Plasma current (MA)	13.19	8.74	14.16	11.24	10.49
B_o	Field on axis (T)	2.876	2.87	2.59	2.87	3.45
B_c	Field at conductor (T)	10.05	10.05	9.05	10.05	12.06
$T_i(0)$	Ion temperature (keV)	20.00	20.00	20.00	20.00	20.00
$T_e(0)$	Electron temperature (keV)	20.00	20.00	20.00	20.00	20.00
n(0)	Electron density ($\times 10^{20}$ m^3)	4.62	2.03	2.96	2.61	2.92
$\bar{n}/n_{GR}$	Ratio to Greenwald limit	0.43	0.29	0.26	0.29	0.34
Z_{eff}		2.40	2.40	2.40	2.40	2.40
W	Stored energy in plasma (MJ)	82.22	36.10	52.64	46.42	51.99
P_{heat}	Total heating power (MW)	123.34	39.93	89.27	52.94	61.32
τ_E	τ_E (s)	0.67	0.90	0.59	0.88	0.85
H	H factor over 89 L–mode	4.81	5.47	3.47	4.89	5.18
VH	τ over 85 ELM-free	3.82	4.20	2.73	3.65	3.89

transport calculations with a transport barrier formation model based on E×B shear have been done for similar ST devices. If neoclassical ion transport inside the transport barrier and an electron thermal diffusivity 10 times the ion neoclassical diffusivity was used, then the required H factors were obtained. The critical factor is whether the turbulence shearing rate achievable in L–mode can be large enough to trigger transport barrier formation. We found that if the assumed growth rate of the turbulence was 60 kHz or less, then the transport barrier could form. If that required growth rate had turned out to

Table 2

	ST FDF Cases	A Net Electric! 10 cm Inboard	F ~OK Blanket Low B_T	G Really High β f_{bs} = 0.5
A	Aspect ratio	1.6	1.6	1.6
a	Plasma minor radius (m)	0.70	0.70	0.70
R_o	Plasma major radius (m)	1.12	1.12	1.12
κ	Plasma elongation	3.00	3.00	3.00
R_{hole}	Hole size (m)	0	0	0
J_c	Centerpost current density (MA/m^2)	50	30	30
f_{ramp}	Induct ramp fraction	0.00	0.00	0.00
P_f	Fusion power (MW)	516.69	66.96	216.96
P_c	Power dissipated (MW)	84.45	30.40	30.40
$P_{internal}$	Power to run plant	211.99	73.80	190.69
Q_{plant}	Gain for whole plant	1.12	0.42	0.52
Q_{plasma}	P_{fusion}/P_{aux}	25.83	3.35	4.54
$P_{netelec}$	Net electric power (MW)	25.69	–43	–9.089
P_n/A_{wall}	Neutron power at blanket (MW/m^2)	7.77	1.01	3.26
β_T	Toroidal β	0.54	0.54	0.98
β_N	Normalized β (mT/MA)	8.30	8.30	8.30
f_{bs}	Bootstrap fraction	0.90	0.90	0.50
P_{cd}	Current drive power (MW)	13.65	2.95	47.78
I_p	Plasma current (MA)	13.19	7.92	14.25
B_o	Field on axis (T)	2.87	1.72	1.72
B_c	Field at conductor (T)	10.05	6.03	6.03
$T_i(0)$	Ion temperature (keV)	20.00	20.00	20.00
$T_e(0)$	Electron temperature (keV)	20.00	20.00	20.00
n(0)	Electron density ($\times 10^{20}$ m^3)	4.62	1.66	2.99
$\bar{n}/n_{GR}$	Ratio to Greenwald limit	0.43	0.26	0.26
Z_{eff}		2.40	2.40	2.40
W	Stored energy in plasma (MJ)	82.22	29.60	53.28
P_{heat}	Total heating power (MW)	123.34	33.39	91.17
τ_E	τ_E (s)	0.67	0.89	0.58
H	H factor over 89 L–mode	4.81	5.96	3.75
VH	τ over 85 ELM-free	3.82	4.90	3.09

be 100 kHz, then we could assert with near certainty that the transport barrier would form since such values of the turbulence shearing rate in DIII–D invariably form transport barriers and it is hard to imagine the turbulence growth rates exceeding 100 kHz, a value that approaches ideal instability growth rates.

Table 3

		A	H	I	J
		Net Electric!	Q = 4 Blanket	OK Blanket	Driven VNS
	ST FDF Cases	10 cm	Cases at β_N 4.15		
		Inboard	β_N Lower	f_{bs} 0.7	Get to H = 2
A	Aspect ratio	1.6	1.6	1.6	1.6
a	Plasma minor radius (m)	0.70	0.70	0.70	0.70
R_o	Plasma major radius (m)	1.12	1.12	1.12	1.12
κ	Plasma elongation	3.00	3.00	3.00	3.00
R_{hole}	Hole size (m)	0	0	0	0
J_c	Centerpost current density (MA/m^2)	50	63	60	30
f_{ramp}	Induct ramp fraction	0.00	0.00	0.00	0.00
P_f	Fusion power (MW)	516.69	81.39	110.70	84.75
P_c	Power dissipated (MW)	84.45	134.07	121.60	30.40
$P_{internal}$	Power to run plant	211.99	259.64	266.78	216.30
Q_{plant}	Gain for whole plant	1.12	0.14	0.19	0.18
Q_{plasma}	P_{fusion}/P_{aux}	25.83	4.07	5.53	1.42
$P_{netelec}$	Net electric power (MW)	25.69	–222.20	–215.86	–177.31
P_n/A_{wall}	Neutron power at blanket (MW/m^2)	7.77	1.22	1.66	1.27
β_T	Toroidal β	0.54	0.143	0.18	0.61
β_N	Normalized β (mT/MA)	8.30	4.15	4.15	4.15
f_{bs}	Bootstrap fraction	0.90	0.90	0.70	0.20
P_{cd}	Current drive power (MW)	13.65	3.41	14.63	59.72
I_p	Plasma current (MA)	13.19	8.31	10.18	17.81
B_o	Field on axis (T)	2.876	3.62	3.45	1.72
B_c	Field at conductor (T)	10.05	12.67	12.06	6.03
$T_i(0)$	Ion temperature (keV)	20.00	20.00	20.00	20.00
$T_e(0)$	Electron temperature (keV)	20.00	20.00	20.00	20.00
n(0)	Electron density ($\times 10^{20}$ m^3)	4.62	1.83	2.14	1.87
$\bar{n}/n_{GR}$	Ratio to Greenwald limit	0.43	0.27	0.26	0.13
Z_{eff}		2.40	2.40	2.40	2.40
W	Stored energy in plasma (MJ)	82.22	32.63	38.05	33.30
P_{heat}	Total heating power (MW)	123.34	36.28	42.14	76.67
τ_E	τ_E (s)	0.67	0.90	0.90	0.43
H	H factor over 89 L–mode	4.81	5.19	4.75	2.15
VH	τ over 85 ELM-free	3.82	3.89	3.47	1.72

However, the ubiquitous appearance of an H factor around 5 in the cases herein implies that seeing vigorous transport barrier formation in NSTX and MAST and understanding the physics of transport barrier formation and expansion in those machines and other conventional aspect ratio tokamaks is a critical path research item for enabling this FDF concept to move forward. We should expect important progress in this area in the 3–5 year timeframe.

We now turn to examining how robust are the various missions of the FDF against physics outcomes less than the base case. Of course none of these cases will be able to deliver net electric performance.

Cases Without Wall Stabilization

The use of wall stabilization to achieve high β_N in tokamaks is a current subject of research and the outcome is highly uncertain at this early stage of that research. Would the basic blanket development mission of the FDF survive if wall stabilization did not work? First, we point out that wall stabilization for axisymmetric modes can be relied upon to obtain high elongation since we can probably configure the device with a close conducting wall and suitable axisymmetric control coils close behind that wall. Such configurations are known to work in present machines. The issue is wall stabilization of non-axisymmetric modes for high β_N. The machine can be designed with wall stabilization systems. The issue is whether the physics works out. We should have important results on this issue in the 3–5 year time frame.

In Table 1, we show various cases assuming performance at the high end of what might be achieved without wall stabilization. The PPPL group have calculated a stable case with $\beta_N = 5.8$ at this elongation and aspect ratio without wall stabilization [5] We also note that the START machine has achieved $\beta_N \sim 5.8$ experimentally in a case where wall stabilization should not have been a factor. We have taken $\beta_N = 5.5$ in these cases.

Case B shows the effect of just reducing β_N to 5.5. The fusion power is cut by a factor of five! Nevertheless the neutron power at the blanket remains above 1 MW/m^2 and the device is run steady-state since the bootstrap fraction is maintained at 90%. β_T has dropped to 24% in this case. This case seems suitable for the blanket development mission, although one would like to be able to test blanket designs at higher wall loading.

Case C attempts to recover some of the lost fusion power and wall loading by turning down the bootstrap fraction to 0.5 and thus turning up β_T to 43%. The fusion power doubles back up to 212 MW and the wall loading is a healthy 3 MW/m^2. The price paid is now an auxiliary current drive power of 47 MW which raises the electrical power needed to run the plant to 256 MW. This cost of this power is probably still bearable for a machine that can run steady-state for blanket development and some net tritium production.

Case D is an intermediate case with a bootstrap fraction of 0.7; less required current drive power; but lower β_T, fusion power output, and wall loading. But at a wall loading of 2.5 MW/m^2 and only 215 MW needed to run the plant, this case looks very adequate for the blanket development and tritium missions.

Case E retains the high bootstrap fraction of Cases A and B but seeks to recover the lost fusion power output and wall loading by just raising the toroidal field from 2.87 T to 3.45 T. Then 206 MW fusion power output and the 3 MW/m^2 neutron wall loading is recovered. The price paid is increased TF coil dissipation, but the total power needed to run the facility remains in an acceptable range at 250 MW.

From these cases, we conclude that the missions of the FDF up through blanket development and tritium self-sufficiency could survive with a downside outcome in the physics of wall stabilization. We also note that the plasma Q has remained above 4.5 in all these cases so that all these cases are suitable burning plasma experiments.

Cases With Some OH Startup Assist

The machine should be designed for day-one operation without an OH coil in order to progress most rapidly on a commitment to full steady-state. The physics basis for full non-inductive operation may be available in the 3–5 year time frame. As a contingency plan against either the inadequacy of the database obtained in that 3–5 year period or against initial problems with startup in the FDF, a substitute centerpost should be fabricated during the device construction and available for installation day-one. In order to make room on that centerpost for an OH coil capable of startup and rampup to some significant current, the toroidal coil portion of that centerpost would be reduced in size and only capable of about half-field operation.

We have in Table 2 placed two cases at a reduced field of 1.72 T. **Case F** is just the base case with reduced field. Wall stabilization is assumed to work and β_N is 8.3. The fusion power has fallen to 67 MW and the wall loading is a barely acceptable 1 MW/m^2. The plasma Q at 3.4 is slipping below what would be adequate for a burning plasma experiment. In **Case G**, we have recovered some of the lost fusion power by turning down the bootstrap fraction to 0.5, which increases β_T (to 98%!) and requires 48 MW of auxiliary current drive. The plasma Q is recovered to 4.5, adequate for a burning plasma experiment. The marginality of these cases indicates that only some of the first phase objectives in burning plasma and DT physics might be achieved with this reduced field centerpost. But such a centerpost is only for temporary usage anyway, until the machine operators learn in the FDF machine how to run without the OH transformer assist and then to reinstall the centerpost for full field operation.

Cases With Much Poorer Physics Performance

Here we look at what mission elements remain if the stability performance were so poor that only half the base case β_N could be achieved (β_N = 4.15). It is highly unlikely that the performance would turn out to be so poor. This value of β_N is below the no-wall limit and well below the value of 5.8 achieved in START already. The toroidal beta values in Cases H–J are only 14%–18%. START has achieved 40%. NSTX and MAST expect to achieve such beta values with Ohmic heating alone.

Case H is the base case with just β_N lowered to 4.15, but with the toroidal field turned up to 3.62 T to compensate an otherwise unacceptable loss of fusion power. The fusion power has dropped to 81 MW and the wall loading is barely acceptable at 1.2 MW/m^2. The power to run the plant has risen to 260 MW. This case remains barely adequate for the blanket development mission.

Case I seeks to recover some of the lost fusion power by turning down the bootstrap fraction to 0.7 and correspondingly increasing the toroidal beta to 18%. The fusion power rises to 111 MW and the wall loading becomes a more healthy 1.7 MW/m^2. The current drive power required is a modest 15 MW and the total power required to run the plant is 267 MW. The plasma Q value is 5.5, adequate for a burning plasma experiment.

We conclude the blanket development mission could survive even as poor a stability physics outcome as β_N = 4.15.

A Case With Confinement H-Factor = 2

However, we note again that all the preceding cases had a confinement H factor around 5. What sort of mission or machine would result if the confinement H factor could only be 2, ordinary H–mode (but once again measured by a conventional tokamak yardstick) and also only have the very low value β_N = 4.15? Essentially, if the H factor is only 2, one has to dramatically increase the plasma current in order to have a high enough confinement time to have enough fusion power output. For **Case J**, we had to drop the bootstrap fraction all the way to 0.2. The value of β_T rises to 61%. The plasma current then becomes 18 MA, compared to the 8–14 MA range of all the other cases. The current drive power required is 60 MW and accounts for most of the 216 MW needed to run the plant. The toroidal field had to be turned down to 1.72 T to reduce the dissipation in the TF coil in order to keep the total plant power reasonable. The fusion power is 85 MW and the wall loading an acceptable 1.3 MW/m^2. The H factor is 2.15.

This machine is in a very different parameter space than all the other cases. This machine is predicated on conventional tokamak first stable regime and ordinary H–mode scaling. To gain high performance, high current is needed. (This machine may have a problem with the edge safety factor.) High beta is achieve through high current and low toroidal field. This is an entirely driven machine with a plasma Q not much larger than one. It is not suitable as a burning plasma experiment. This type of machine is often what has been discussed as a volume neutron source, since it relies on such conservative physics.

Summary

We have described an ST device suitable for a Fusion Development Facility. It is a small device sized to have a chance at net electric breakeven with positive outcomes in all its physics aspects. We have described the general design features of such a machine. Challenging new areas of design are the single turn centerpost and the limited space on the inboard side, the unique maintainability opportunities, the unusual toroidal field power supplies (their configuration and the high current leads), and a double layer PF system for shape and non-axisymmetric mode control. Some areas of physics challenge are the high normalized beta values needed with wall stabilization and high bootstrap fractions (approaching 100%), neoclassical tearing modes, Alfvén eigenmodes, fully non-inductive startup and rampup, current drive methodologies, transport barrier formation, and high power density.

The missions of this FDF for burning plasma studies, blanket development, and tritium self-sufficiency seem secure against unfavorable outcomes of wall stabilization research and poor stability results. A high confinement quality is required and is expected from transport barrier physics. The mission of some significant burning plasma studies also seems secure against unfavorable research outcomes in fully non-inductive operation. These analyses persuade us that it would be reasonable to move directly to this FDF project after positive results from NSTX and MAST. Those positive results are mainly needed in transport barrier physics and fully non-inductive operation and secondarily in stability physics. Research results on these critical issues can be expected in a 3–5 year time frame.

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