Advanced Tokamak Modes and ITER

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I. Introduction

In view of the accumulating evidence from tokamak experiments and theory that advanced tokamak modes of operation have achieved higher values of confinement, beta and bootstrap current fraction than had been thought plausible at the beginning of the International Thermonuclear Experimental Reactor (ITER) Engineering Design Activity (EDA) we provide comments on:

1. (a.) The range of "advanced tokamak" parameters that might be established as credible by the existing tokamak experiments over the next two to three years. (b.) The technological capability in a design that is required for accessing the plasma conditions necessary to achieve these parameters.
2. The capability of the present ITER design to access the plasma conditions identified in 1.(a.). Possible modifications of the ITER design which could significantly enhance this capability, without having a large impact on the overall design.
3. Improvement in the ITER design (performance) which would result if the "advanced tokamak" parameters identified in 1.(a.) could be used as the physics design basis.
4. The design base for DEMOs that would be provided by ITER operation in the reference pulsed ignited mode and in advanced tokamak modes.

Present day tokamaks (DIII-D, JET, JT-60U and TFTR) operate with intermediate advanced tokamak parameters of confinement enhancement - $H = \tau_E/\tau_{ITER\ 89P} \leq 3$ and normalized betas $\beta_N = \beta/(I/aB) \leq 4$ for duration about equal to the energy confinement time. At longer time scales, the order of the plasma current redistribution time, the best performance is $H \sim 2$ and $\beta_N \cong 2$. It is credible that during the next two to three years, experimental techniques will be developed to produce intermediate advanced tokamak plasmas with $H \leq 3$ and $\beta_N \leq 4$ for plasma current redistribution time scales.

The present ITER EDA Outline Design appears to be sufficiently flexible to accommodate an intermediate level of advanced tokamak performance characterized by $H \leq 3$, $\beta_N \leq 4$, the data base for which is anticipated to be established by existing experiments over the next two to three years. This intermediate level of advanced tokamak operation would enable ITER to operate in steady-state at 1500 MW fusion power with 100 MW current drive/control power.

Operation in an accessible, full-power, steady-state advanced tokamak mode would significantly enhance, relative to operation in a low-duty-factor pulsed ignition mode, the capability of ITER to accomplish its nuclear testing mission and to provide the design data base for attractive DEMOs. With steady-state operation, the significant limitations on nuclear testing which would be imposed by low-duty-factor pulsed operation would be avoided, and the likelihood of achieving neutron fluence accumulation in excess of 1 MW-a/m^2 during the Extended Performance Phase would be enhanced. The physics design base for DEMOs that would be established by ITER operating at $H \leq 3$, $\beta_N \leq 4$ would support DEMO designs which would be about 1 - 2 m smaller in major radius and about 4 - 6 MA lower in plasma current than the DEMO designs that would be supported by ITER operation in the present reference pulsed ignited mode with $H \leq 2$, $\beta_N \leq 2.5$. Operation of TPX at $H \leq 4$, $\beta_N \leq 6$ would provide the data base for further improvements in the attractiveness of a DEMO.

II. Assessment of Advanced Tokamak Status

The design of ITER is based on physics principles developed from a broad worldwide data base of inductively driven tokamak operation. Recent experimental results indicate the possibility of new improved operating modes (see Table 1) which offer the advantage of operating ITER at lower plasma currents, 12 to 16 MA, rather than 24 MA.

Table 1. Advanced Tokamak Operating Modes are Numerous and Widespread

	H-Mode*	Reversed Shear	High Internal Inductance	High Poloidal Beta	VH-Mode	Internal Transport Barrier	Super Shot	Hot Ion Mode
DIII-D	√	√	√	√	√	√		√
JET	√	√	√		√			√
JT-60U	√	√		√		√		√
C-Mod	√							
PBX-M	√					√		
TFTR	√	√	√	√			√	√
TORE SUPRA		√	√	√				

* Many other machines operate with H-mode confinement.

Lower current operation with current profile control, at higher safety factor, is motivated by three experimental observations:

1. The energy confinement time τ_E increases proportional to plasma current; but in DIII-D only to safety factor of $q_{95} \approx 4.5$. At higher current, the confinement no longer increases with plasma current. Experiments in DIII-D indicate that the product $\beta\tau$ maximizes at safety factors ≈ 4.[(2)].
2. The plasma disruptivity decreases at lower current. The probability of plasma disruptions for $q_{95} > 4$ is low. Since higher normalized beta operation is possible at higher q_{95}, similar plasma beta can be achieved.
3. At lower current (higher q_{95}), the bootstrap current fraction is higher. Thus, steady-state operation would require non inductively driving a smaller fraction of a lower plasma current so that the effective current drive efficiency would be significantly higher.

To take advantage of lower current operation, it will be necessary to optimize the radial current profile. An attractive profile is one with reverse shear (a radial plasma current profile which peaks off axis) generated by means of off-axis RF or neutral beam current drive. The off-axis peaked current profile provides stability against ballooning modes as well as against internal MHD modes by avoiding minimum values of the safety factor which are low rational values. Concentrating the plasma current closer to the wall enhances wall stabilization effects which can provide increased MHD stability to global modes. Improved stability and/or improved confinement have been observed with such reversed shear configurations in five tokamaks (see Table 1). This operating mode utilizes bootstrap currents very efficiently. Gradients in the pressure profile naturally generate off-axis bootstrap currents so the bootstrap currents are located where they are most effective. Thus high bootstrap current fractions are allowed, reducing the need to drive large plasma currents, and thus reducing the current drive power requirements to acceptable levels. Such steady-state operation in ITER would reduce the risks associated with plasma disruptions.

The status of advanced tokamak research can be characterized by three parameters: confinement quality, $H = \tau_E/\tau_{ITER89P}$, bootstrap current fraction, $f_{BS} = I_{BS}/I_p$ and the stability factor $\beta_N = \beta/(I/aB)$. Ultimately these parameters need to be simultaneously sustained in steady-state at high values. Three levels of tokamak performance can be defined as given in Table 2:

Table 2. Characterization of Tokamak Performance[(3)]

Nominal:	$H \le 2$	$\beta_N \le 2.5$	$H\beta_N < 5$
Intermediate (Advanced):	$H \le 3$	$\beta_N \le 4$	$H\beta_N < 12$
Superior (Advanced):	$H \le 4$	$\beta_N \le 6$	$H\beta_N < 24$

The values of confinement quality, H= $\tau_E/\tau_{ITER89P}$, bootstrap current fraction, $f_{BS} = I_{BS}/I_p$ and the stability factor $\beta_N = \beta/(I/aB)$ which have been transiently achieved independently and those achieved simultaneously are shown in Table 3 along with a comparison of projected advanced tokamak performance parameters. Triangularity refers to the shape of the outer flux surface and is defined as the ratio of the radial distance from the highest point on the outer flux surface to the magnetic axis divided by one-half the width of this flux surface at the midplane.

ITER is designed on the basis of nominal tokamak performance parameters which have been achieved for about ten energy confinement times. Today's tokamak experiments have transiently achieved the *intermediate* (advanced) performance level, a level that the present ITER design might achieve in steady-state.[5] Experiments in the next three years, are expected to demonstrate *intermediate* (advanced) performance in steady-state for time scales on the order of the current profile relaxation time. Such demonstrations in ITER-like geometry would put the possibility of steady-state *intermediate* (advanced) operation of ITER on a firm footing. On a somewhat longer time scale, experiments will attempt to reach *superior* (advanced) performance levels. Proposed experiments in TPX and JT-60SU would establish steady-state operation in the *intermediate* (advanced) and *superior* (advanced) performance levels, for times long relative to both the current profile relaxation time and the plasma-wall particle equilibrium time. These results are expected well before ITER operation in 2008.

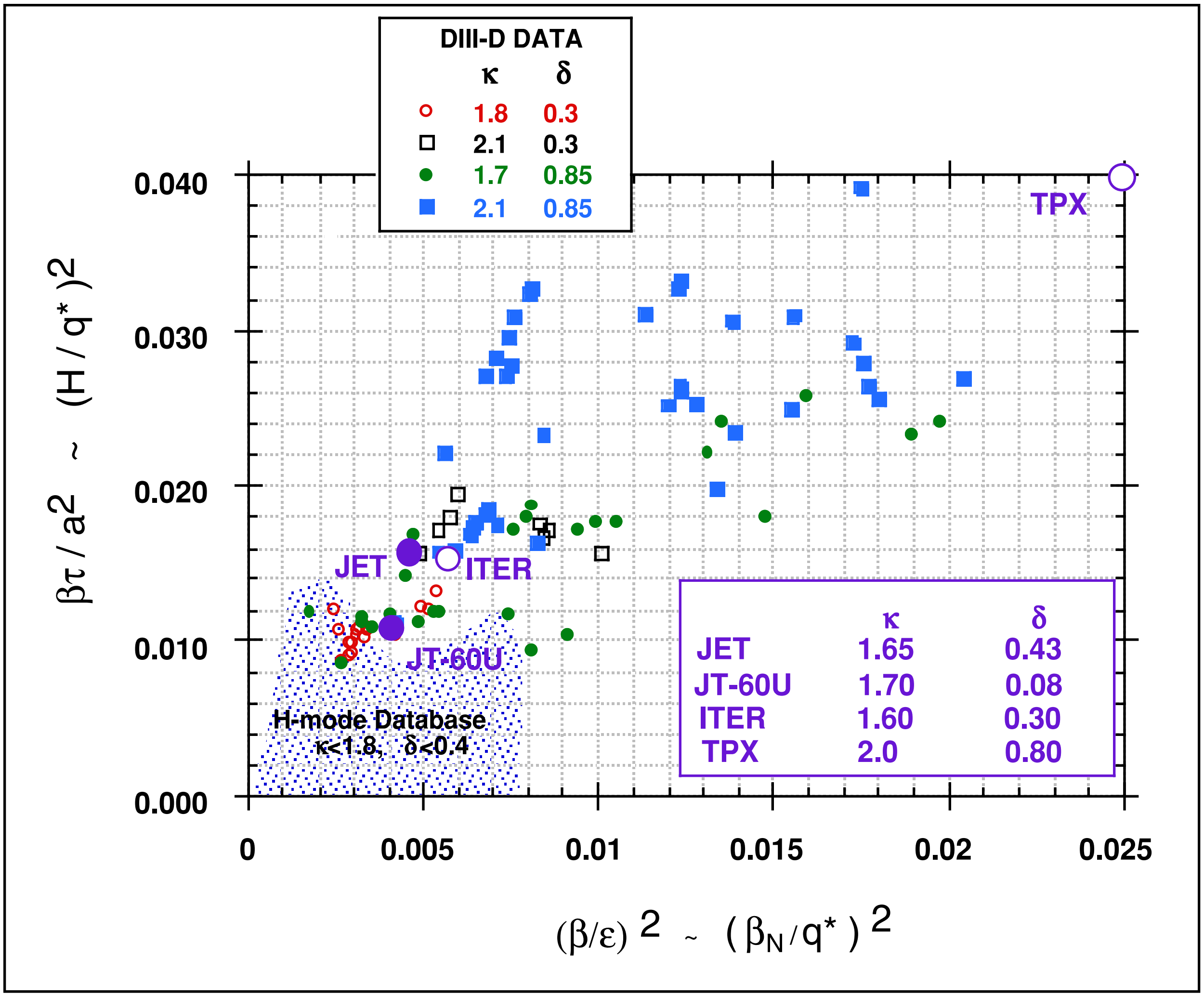
III. Technological Requirements for Advanced Tokamak Modes

While we are still learning what features of a tokamak design lead to advanced modes, certain requirements now appear necessary. Figure 1 shows normalized parameter space that characterizes advanced tokamak modes; the axis have scaled out the plasma size, current, and field so that the influence of shape, profiles, and other variables on advanced tokamak data can be extracted.[2] The vertical axis, $(H/q^*)^2$ with q^* as the cylindrical equivalent q, measures the confinement enhancement at a given q^* value while the horizontal axis, $(\sim \beta_N/q^*)^2$, measures the MHD stability. This formulation scales away the stabilizing influence of reducing the current (kink instability drive). Also, if all superconducting coils used the same peak field, the horizontal axis would also be roughly proportional to fusion power density (since εB^2 is roughly constant over normal ranges of aspect ratio). Now, one can define "penetration" into the advanced tokamak regime as occurring along the diagonal from the ITER (nominal performance) target regime to the TPX target regime, and a key parameter according to this data is the plasma triangularity.

Another important feature is the control and specific shaping of the current profile, which influences both stability and confinement through magnetic shear and perhaps other phenomena. For many current profiles, the stability analyses (at higher β_N) require a nearby conducting wall capable of sustaining wall currents in the pattern corresponding to the unstable modes. Stability on times longer than the wall diffusion time requires plasma rotation. The role of shear in the toroidal rotation may be important as well. The interdependence of the first wall distance, rotation speed, and blanket/shield thickness will require further optimization.

It is important to note that we do not yet have a unique "prescription" for designing a device for a given penetration into the advanced regimes. Nonetheless, we do know from theory and experiment that the above key features are important for advanced tokamak operation. These include: an equilibrium system with sufficient flexibility to allow appropriate elongation and triangularity; means for controlling shaped plasmas (vertical stability control); conducting walls that allow helical current patterns; means for inducing toroidal rotation; and current drive systems that allow both central and off-axis current drive.

Figure 1. Effect of Triangularity on Advanced Tokamak Performance


DIII-D DATA
κ δ
1.8 0.3
2.1 0.3
1.7 0.85
2.1 0.85
TPX
JET
ITER
JT-60U
H-mode Database
κ<1.8, δ<0.4
JET 1.65 0.43
JT-60U 1.70 0.08
ITER 1.60 0.30
TPX 2.0 0.80
0.040
0.030
0.020
0.010
0.000
0
0.005
0.01
0.015
0.02
0.025
$\beta\tau / a^2 \sim (H / q^*)^2$
$(\beta/\varepsilon)^2 \sim (\beta_N / q^*)^2$

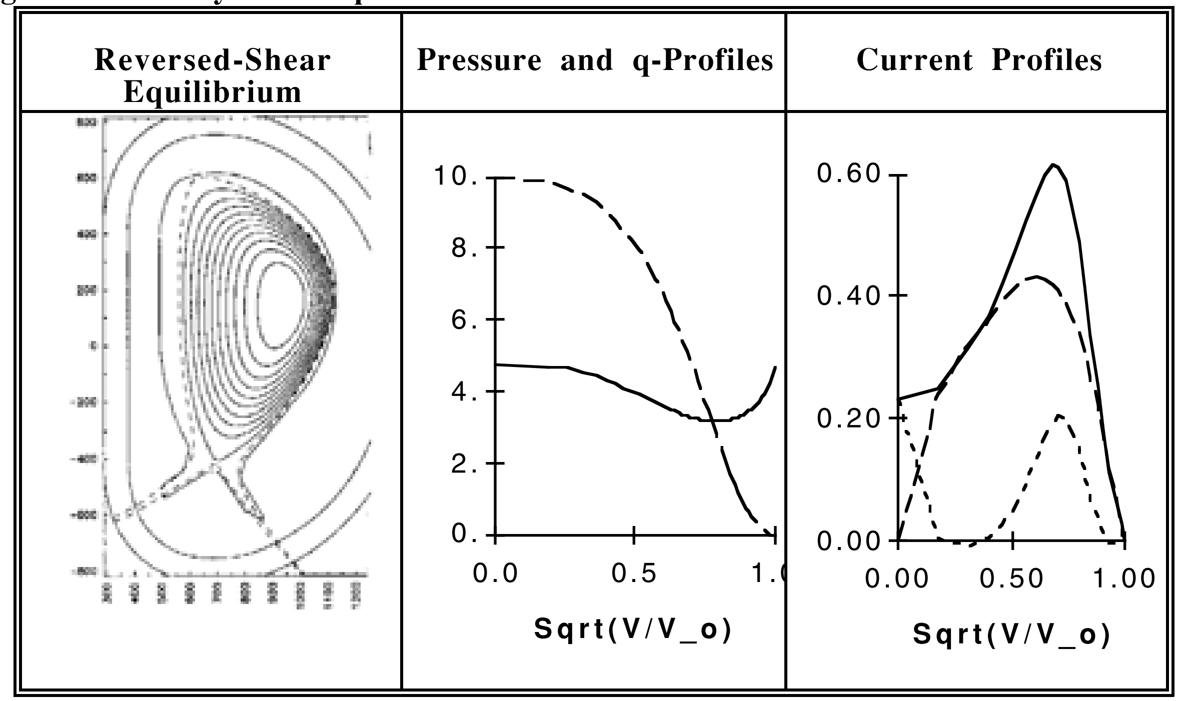
In Summary the technological requirements for accessing advanced tokamak operation include:

- high triangularity, $\delta_{95} > 0.4$, with improvement expected for higher values,
- plasma control adequate for plasma position and shape,
- conducting first walls that allow helical response currents,
- error correction coils to prevent locked modes, and
- plasma rotation control, a few per cent of the Alfvén speed,
- plasma current profile control using off- and on-axis current drive,
- divertor geometry sufficiently flexible to allow high triangularity, and
- steady-state engineering capability (e.g., true steady-state coils and cooling systems)

IV. Capability of ITER to Access Advanced Tokamak Modes

Steady-state operation of ITER at interesting fusion powers ($P_{fusion} \geq 1$ GW) using the initial auxiliary heating and current drive system ($P_{aux} \leq 100$ MW) is only feasible in scenarios with high bootstrap current fractions. MHD stability and current profile alignment favor advanced tokamak scenarios in which the magnetic shear reverses between the magnetic axis and the plasma surface. These scenarios are shown to be compatible with the ITER poloidal field and auxiliary heating and current drive systems(5) could sustain steady-state operation while producing 1500 MW of fusion power (assuming a 12.5% helium-ash fraction) with 100 MW of auxiliary power. The equilibrium configuration, and radial profiles of pressure, safety factor, and plasma current are shown in Fig. 2.

Figure 2. Steady-State Operation in ITER



This operating scenario has been examined for MHD stability with the PEST[19] code, and found to be stable for $\beta_N \leq 4$ in the presence of a conducting wall at 1.25 times the plasma radius. The profiles of safety factor (solid curve), plasma pressure (dashed curve shows $10P/P_o$), total plasma current density in MA/m^2 (solid curve), and bootstrap current density (long-dash curve) and driven current (short-dash curve) are shown in Fig. 2. We see that the bootstrap current density is reasonably well aligned with the total current density with a high bootstrap current fraction ($I_{BS}/I_p = 0.80$). The energy confinement time required at this operating point is 2.8 s (for an energy enhancement H = 2.23 relative to ITER-89 confinement scaling—greater than required for ITER ignited operation, but less than what has been produced transiently in JET PEP modes[20]).

The free-boundary MHD code TEQ[21] has been used to study the compatibility of this advanced operating mode with the ITER poloidal field system and divertor. This magnetic configuration is compatible with the ITER divertor, poloidal field, and vertical position control systems. While the advanced tokamak performance capability of ITER could be further improved with design changes, *intermediate* (advanced) tokamak operating modes utilizing reversed shear are possible within the present ITER design with the addition of current profile control and the assurance of adequate vertical position control. These configurations with a total plasma current of 14 MA and bootstrap current fractions of > 80% require $\beta_N = 3$ and H = 2.3 which are well within the range of parameters that have been achieved transiently and are anticipated to be achieved over longer time scales within the next three years.

V. Improvement in ITER Performance Using the Advanced Tokamak Basis

In this discussion, we have assumed that the toroidal field, first wall and other engineering systems can operate in steady-state at the parameters specified in this study. In addition to the ultimate goal of demonstrating steady-state operation, advanced steady-state operating modes could be employed in ITER to contribute to the development of the physics data-base for a steady-state demonstration reactor and as a high-fluence steady-state source of neutrons for blanket testing in the Extended Performance Phase.

Table 4. ITER Advanced Tokamak Operating Modes[5]

Parameter	Steady-State Physics Mode	DEMO Physics Mode
P_{fusion} / P_{CD}(MW)	1500/100	2000/50
Q_{CD}	15	40
R_o / a (m)	8.56/2.56	8.46/2.5
$\kappa_{95\%}$ / $\delta_{95\%}$	1.79/0.42	1.84/0.46
B_o (T)	5.4	4.34
q_ψ	4.46	3.7
I_p (MA)	13.9	12.2
f_{BS}	80%	0.91
f_{He}	12.5%	12%
β_N / β_N^*	3.0/4.0	5.0/6.7
$\langle n_e \rangle$ / n_{eo}(10^{20}m^{-3})	0.96/1.52	1.25/2.0
$\langle T_e \rangle_n$ / T_{eo} (keV)	12.3/21.0	10.2/18.4
τ_E (seconds)	2.84	2.64
τ_E / τ_{ITER-P}	2.23	2.59

A critical issue for a steady-state demonstration reactor is the beta-limit as measured by β_N. The column labeled "DEMO Physics Mode" in Table 4 shows that a high β_N operating mode can be achieved in ITER without greatly increasing the fusion power by operating at decreased toroidal field.[5]

Thus we conclude, the *intermediate* (advanced) tokamak capability, which should be well documented by 2000, would allow steady-state operation of ITER at 1500 MW and Q ~ 15.

The steady-state mode of operation allowed by the *intermediate* (advanced) tokamak mode has definite advantages for nuclear testing. The limitations associated with low duty cycle operation with 1000 second pulses would be eliminated. The fluence accumulation capability is also expected to be enhanced by both the higher duty cycle operation and the potential for improved reliability with steady-state operation. Machine operation in the steady-state mode with 10% machine availability would enable accumulation of neutron fluence somewhat in excess of 1 MWa/m^2, thereby satisfying the minimum fluence objectives.[(22)]

VI. Impact on DEMO Design if ITER Is Operated in Advanced Tokamak Mode

The DEMOs to follow ITER will be designed based on a conservative extrapolation of the design data base which exists at the time. That DEMO design data base will be provided in large part by ITER and its supporting R&D program. The design base provided by ITER will depend upon its physics mode of operation.

The characteristics of DEMOs which would extrapolate from both the existing and the emerging physics data bases have been calculated Physics design bases for DEMO corresponding to the three different cases defined in Table 2 were considered. The calculations were performed for a 316SS structural material and for an "advanced" structural material with the thermophysical properties superior to those of 316SS (the properties of the vanadium alloy V-4Ti-4Cr were used to characterize the advanced structural material). The calculation model consisted of an iteration on the various physics and engineering constraints, leading to the minimum major radius device that will satisfy these constraints as described in Refs. [(16, 23)]. ITER physics and engineering design constraints [(24, 25)] are employed insofar as possible.

For the purposes of these calculations, the DEMO was assumed to have neutron wall load and lifetime fluence objectives of 2-4 MW/m^2 and 10 MWa/m^2, respectively, a fusion power level of 1500 MW, a requirement for net electrical power production ($Q \geq 15$), and a requirement for tritium self-sufficiency (tritium breeding ratio $\geq$ 1.15 in a 1D model). The results of the calculations are given in Table 5. In some cases, it was not possible to achieve a self-consistent solution with the maximum physics parameters indicated in Table 2 because of other constraints. With 316SS, the first-wall heat flux limit excluded steady-state (ss) solutions with optimal ($\cong$ 4) values of q_{95}. When it was not possible to achieve steady-state, the pulse lengths were determined to minimize major radius, from a trade-off between crack growth/fatigue reduction in allowable stress with increasing number of pulses and flux core increase to provide volt-seconds for longer pulses.

Table 5. DEMO Parameters Extrapolated from Different Physics and Materials Design Bases[23]

	Nominal/Ignition		Intermediate Physics/Driven				Superior Physics	
Material	316SS	Advanced	316SS	Advanced	316SS	Advanced	316SS	Advanced
Pulse Length	dt=1h	dt=1h	dt=3h	dt=3h	SS	SS	SS	SS
R(m)	7.6	7.7	7.4	6.3	6.5	6.5	6.1	5.3
I(MA)	18.5	18.5	16.0	12.9	14.1	14.0	9.7	11.7
β_N	2.5	2.5	2.8	3.9	4.0	4.0	5.8	4.6
H	2.0	2.0	1.9	3.0	2.6	2.6	4.0	4.0
q_{95}	3.0	3.0	5.0	4.0	6.3	5.5	9.2	4.2
Bootstrap f_{BS}	0.34	0.34	0.46	0.64	0.76	0.70	0.80	0.72
q_{FW} (MW/m^2)	0.35	0.34	0.45	0.70	0.45	0.52	0.45	0.96
Γ_N(MW/m^2)	1.2	1.2	1.2	1.8	1.3	1.3	1.4	2.5

The DEMOs which would extrapolate from the reference, ignited ITER physics design base (nominal case in Table 2) would be only somewhat smaller (R ≈ 7.5 m) and lower current (I ≈ 18 MA) than ITER. These designs are smaller than ITER primarily because of credit taken for bootstrap current and 50% startup assist in determining the required volt-seconds. Confirmation of the emerging *intermediate* (advanced) tokamak physics design base of Table 2 by *intermediate* (advanced) tokamak operation in ITER would allow more compact (R ≈ 6-7 m) DEMO designs at lower plasma current (I ≈ 12-14 MA) than would be possible with the reference ignited ITER physics design base. DEMOs which would be designed on the basis of the *superior* (advanced) tokamak physics design base of Table 2 would be yet smaller (R = 5-6 m) and have yet smaller plasma current (I ≈ 10-12 MA).

The primary structural material for ITER, 316SS, has a relatively low first-wall heat flux limit, which probably would preclude full utilization of the advanced physics data bases in the DEMOs. An advanced structural material would allow full utilization of the advanced physics design bases.

The present ITER EDA Outline design [1] would appear to be capable of confirming in DT an *intermediate* physics design basis for DEMO, but, if triangularity is a critical design parameter, not the *superior* physics design base that would be established by TPX and JT-60SU. Since the *superior* (advanced) physics design base would lead to DEMOs that are about 1 m more compact and that need about 2 MA less plasma current than would DEMOs based on the *intermediate* physics design base, there would be an incentive to use the *superior* physics design base for the DEMO. The credibility of confirming the *superior* physics design base for DEMO by a combination of ITER operating in DT in an *intermediate* physics mode and TPX and JT-60SU operating in DD in a *superior* physics mode needs to be evaluated.

In any event the DEMO physics design base that would be provided by the present ITER design operating in a steady-state advanced tokamak mode would enable the DEMO design to be smaller and lower current than would be possible with the DEMO physics design base that would be provided by the present ITER design operating in the reference pulsed ignited mode.

System studies of the impact of advanced tokamak design features on the cost of electricity produced by a tokamak fusion reactor have been carried out.(26) For example, the cost of electricity of a 1000 MWe steady-state tokamak reactor is reduced by a factor of two by improving performance from H ~ 2 and β_N ~ 2.5 to H ~ 2.5 - 3 and β_N ~ 5 - 6.

VII. Conclusion and Recommendations

The physics data base supporting an *intermediate* (advanced) steady-state mode of ITER operation at full power is approaching the physics data base supporting the ITER reference pulsed ignited mode of operation, albeit only transiently (times the order of the energy confinement time). It is anticipated that the *intermediate* (advanced) data base will be supported on time scales comparable to the plasma current relaxation time by planned experiments on existing tokamaks within the next two to three years. The present ITER design can accommodate an *intermediate* (advanced) tokamak mode of steady-state operation at full power which would provide a much greater nuclear testing capability and a physics design base for subsequent DEMOs which would allow smaller and more attractive DEMOs to be designed following ITER. The planned TPX and JT-60U experiments would confirm the *intermediate* (advanced) tokamak mode of operation on essentially steady-state time scales and would establish the *superior* (advanced) tokamak mode of operation, which could lead to even smaller DEMOs.

Therefore we recommend: 1.) The ITER Design Requirements for the Basic Performance Phase should include a full-power, steady-state operating mode based on an *intermediate* (advanced) tokamak mode of operation. 2.) Operation in the steady-state full-power mode over most of the machine lifetime should be incorporated into the ITER operation plan.

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