Recent Advances in Stellarator Development+

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ABSTRACT

Large stellarator research programs are being pursued in Europe and Japan (>$1 billion in new facilities). These programs are important to the U. S. because the similarities and differences between stellarators and tokamaks can be used both to improve our understanding of toroidal confinement and to develop an improved reactor concept. Stellarators have performance similar to that of comparable tokamaks. New large stellarator experiments under construction with superconducting coils will have parameters comparable to those of present tokamaks. Significant experimental and theoretical progress has been made in understanding transport, finite-beta behavior in three-dimensional toroidal geometry, and concept improvement. Ideas to further improve the stellarator concept will be tested in new experiments. Recent studies have improved the stellarator reactor concept and shown that stellarators can be competitive with tokamaks as reactors.

I. STELLARATOR RESEARCH

Stellarators have the potential for an improved reactor concept because they require no net plasma current. The absence of a large plasma current means that stellarators are inherently steady-state devices with no plasma-terminating disruptions and no need for current drive or plasma stability control. The plasma parameters and profiles do not need to satisfy simultaneous (and often conflicting) constraints on disruption avoidance, density and beta limits, bootstrap current fraction, current drive efficiency, divertor performance, and improved confinement. Thus stellarators should be better able to operate in a true ignited steady-state fashion with time-invariant plasma profiles.

In contrast with tokamaks, new stellarator experiments are under construction around the world: the Large Helical Device (LHD) in Japan with major radius R_0 = 3.9 m [1]; Wendelstein 7–X (W 7–X) in Germany with R_0 = 5.5 m [2]; and the Flexible Heliac TJ–II in Spain with R_0 = 1.5 m [3]. Europe and Japan are investing >$1 billion in these new facilities. By contrast, the U. S. has a much smaller program with only the Helically Symmetric eXperiment (HSX) at the University of Wisconsin (R_0 = 1.2 m) under construction.

The two largest experiments, LHD and W 7–X, will have plasma volumes $V_p \approx 30$ m^3, on-axis fields B_0 = 3–4 T, and heating powers $P_h \approx 30$ MW, values similar to those of present tokamaks (D III–D, Tore Supra, TFTR). These new stellarators will have 10–20 times the plasma volume, heating power, and pulse length of present stellarators to allow extension of physics studies to more reactor-relevant regimes as well as superconducting coils for true steady-state operation at reduced heating power. To maximize understanding, the Japanese and German stellarator programs follow complementary optimization approaches with different coil geometries: LHD (Fig. 1) uses two large helical coils, while W 7–X (Fig. 2) uses nonplanar coils topologically similar to tokamak toroidal-field coils. Decades of experience in building these types of coil systems is behind construction of these devices.

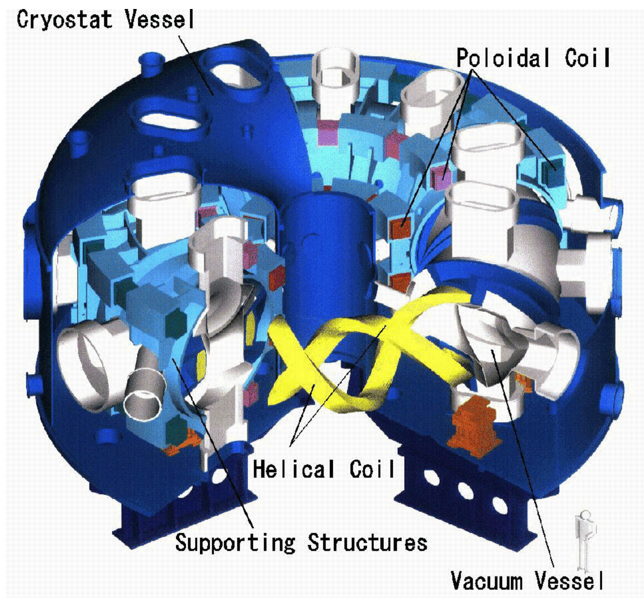

Fig. 1 A sketch of the Large Helical Device

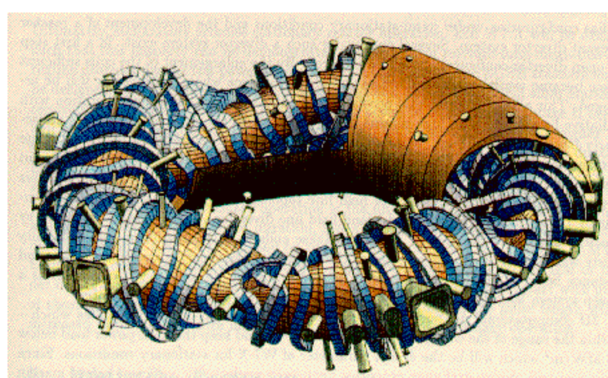

Fig. 2 A sketch of the W 7–X coils and vacuum vessel

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LHD is in the sixth year of an eight-year development and construction program; start of operation is scheduled for early 1998. The main LHD parameters are major radius R_0 = 3.9 m, average plasma radius a_p = 0.65 m, V_p = 30 m^3, and B_0 = 4 T. LHD is discussed in more detail in another paper in these proceedings [1]. W 7–X has just received final approval for construction with start of operation scheduled for 2004. The main W 7–X parameters are R_0 = 5.5 m, a_p = 0.52 m, V_p = 30 m^3, and B_0 = 3 T.

With the U. S. program emphasis on developing the science and technology required to develop fusion energy, the U.S. needs to be involved in collaborative stellarator research. The similarities and differences between stellarators and tokamaks can be used both to improve our understanding of toroidal confinement and to develop an improved reactor concept.

II. STELLARATORS

Stellarators [4] are toroidal devices with confinement properties similar to those of tokamaks. This similarity arises from both devices having toroidally-nested closed magnetic surfaces created by helical (toroidal plus poloidal) magnetic fields. The differences arise because stellarators use currents only in external coils, rather than in the plasma itself, to confine and stabilize the plasma. Because the poloidal magnetic field that produces the rotational transform È (= 1/q, where q is the tokamak safety factor) of the magnetic field is created by currents outside the plasma in a stellarator, the plasma is inherently nonaxisymmetric; the plasma cross section changes shape as it rotates around the (sometimes noncircular and nonplanar) magnetic axis. Another result is that the magnetic shear (1/r)(dq/dr) typically has the opposite sign to that in tokamaks; this "reversed shear" condition has recently been found to be important for improved confinement regimes in tokamaks. The absence of a net plasma current means that internal disruptions ("sawteeth") and current-driven tearing modes do not occur in stellarators. The bootstrap current, important in a tokamak to lessen the current drive requirements but undesirable in a stellarator, can be made zero or reversed in sign in a stellarator.

Comparisons between tokamak and stellarator results can broaden our physics understanding of fundamental processes in magnetically confined plasmas. These types of comparisons have already proven valuable in understanding bootstrap current, second stability, and other toroidal physics issues. Tokamak magnetic configurations are characterized by the plasma aspect ratio, elongation, triangularity, and type of separatrix (none, single null, or double null). The poloidal field is determined by the driven plasma current. Stellarators can access a much wider range of magnetic configurations through tailoring of the spatial Fourier components of the magnetic field to emphasize different properties such as sign and degree of shear, size and extent of the magnetic well, degree of helical axis excursion, confinement of trapped particles, shift of the magnetic axis with beta, degree of helical symmetry, etc.

In practice, there are two main types of stellarators, torsatrons (or heliotrons) with continuous helical coils and modular stellarators with a toroidal set of nonplanar coils. Another type of stellarator, the heliac, uses a helical arrangement of planar circular coils. Torsatrons generally have tokamak-like shear (but reversed in sign), a central magnetic well that increases with beta (second stability), and a nearly circular magnetic axis, while the modular stellarators and heliacs have very low shear, a global magnetic well that does not change with beta, and a relatively large helical excursion of the magnetic axis. Typically stellarators have larger plasma aspect ratio than tokamaks.

LHD, the Compact Helical System (CHS), and Heliotron E, all in Japan, are the main examples of torsatron/heliotron configurations. Wendelstein 7–AS (W 7–AS), and W 7–X (a "helias" or helical axis advanced stellarator) are the main examples of modular stellarators.

Because different coil configurations can be used to create a particular stellarator configuration, and all can be created with modular coils, the underlying magnetic configuration is the important defining feature. Figs. 3 and 4 show the last closed flux surface and its cross section at the beginning, one-quarter and half-way through a toroidal field period for a torsatron and a modular stellarator, illustrating the different degrees of triangularity and helical axis excursion that can be created in stellarators.

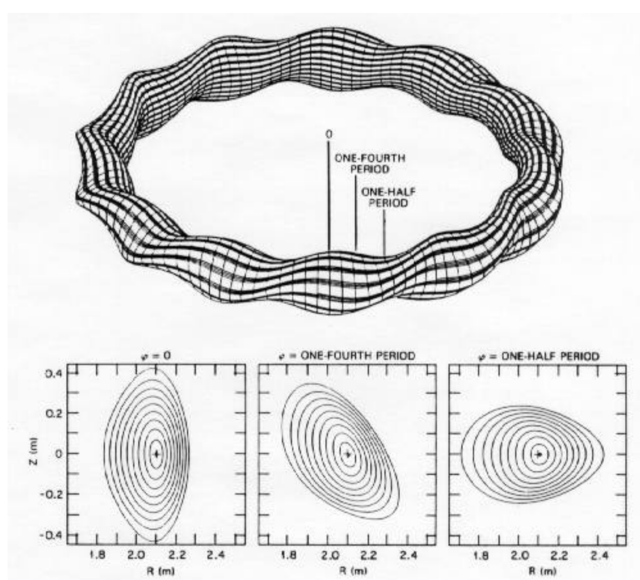

Fig. 3 Flux surfaces for a typical torsatron (the Advanced Toroidal Facility, ATF)

III. RECENT STELLARATOR PROGRESS: EXPERIMENT AND THEORY

With the advent of high-power auxiliary heating, stellarators have achieved plasma parameters similar to those in comparable tokamaks. Much of the physics in tokamaks (transport, bootstrap current) carries over to stellarators if it is phrased in terms of the magnetic configuration properties (transform, shear, magnetic well, etc.). The exceptions are those parameters directly related to the plasma current (e.g., beta limits). The issues addressed in present experiments (W 7–AS, CHS, and Heliotron E) are the same as those of interest in the tokamak program: confinement scaling at higher parameters and understanding of transport mechanisms, more effective plasma heating, finite-beta beta behavior, and concept improvement.

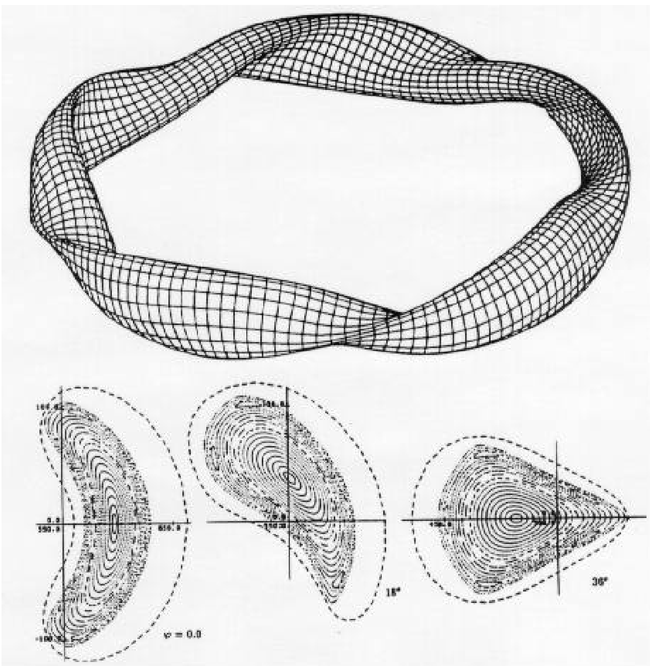

Fig. 4 Flux surfaces for the modular stellarator W 7–X

Despite widely different magnetic configurations, different types of stellarators and tokamaks follow essentially the same confinement scaling. A world stellarator data base, including a large number of data sets from ATF, CHS, Heliotron E, W-7–A, and W 7–AS, has been created to look for differences among stellarators (including scaling with transform, shear, etc.). The energy confinement times follow a gyro-Bohm-type scaling with the same numerical coefficient as for tokamaks. H-mode behavior is seen as in tokamaks, but the confinement improvements thus far are not as large.

Detailed studies of neoclassical core transport, the plasma edge and magnetic islands, the diverted flux layer, edge turbulence, the electric field shear layer, poloidal rotation, etc. have been made for a wide range of stellarator configurations. L–H transitions and the H-mode have also been studied theoretically in stellarators in order to better understand the underlying physical mechanisms in toroidal geometry. The connection with improved confinement modes in tokamaks is of particular interest: centrally reversed shear leads to improved particle and ion energy confinement in tokamaks, while long particle confinement times and ion neoclassical transport are seen in stellarators with globally reversed shear.

Improved parameters have been obtained at modest heating power (1–2 MW): volume-average beta β = 2.1%, approximately half that required for a competitive reactor; T_e = 3.5 keV >> T_i, for studies of electron transport at low collisionality; and T_i = 1.6 keV, T_e = 1.8 keV at $n_e = 5 \times 10^{19}$ m^{-3} for studies of ion transport at low collisionality. Here T_e and T_i are the electron and ion temperatures, respectively, and n_e is the line-averaged electron density.

More effective ion cyclotron range of frequency (ICRF) heating in stellarators is important both for steady-state operation of LHD and W 7–X and for better understanding of ICRF heating in toroidal geometry. Although successfully used in tokamaks, bulk heating had not been observed in stellarators until recently. Sustained bulk heating has now been demonstrated on CHS with a single-strap antenna on the low-field side and on W 7–AS with a 1-m toroidally extended antenna on the high-field side in a very different magnetic configuration from that on CHS (and without impurity generation). The heating efficiency is similar to that obtained with electron cyclotron heating or neutral beam injection. A new two-strap antenna will be installed this year on W 7–AS for study of ion minority heating, mode conversion heating, and direct electron heating.

Density profile control is an important factor for confinement improvement in both stellarators and tokamaks. Recent pellet injection experiments on W 7–AS showed behavior different from that seen in tokamaks or in the absence of pellet injection: a rapid density redistribution without significant loss of plasma accompanied by m = 0 damped oscillations in the soft X-ray signals.

Progress is being made in the understanding of finite-beta behavior in three-dimensional (3–D) toroidal geometry. Magnetic islands have been incorporated in magnetohydrodynamic (MHD) equilibrium codes to study self-healing of magnetic surfaces (the W 7–X helias configuration). This also has important applications to tokamaks (wall modes and field errors). Different techniques have been used to improve calculations of bootstrap current, which can be applied to improved confinement tokamaks with sharp gradients. Recent calculations of the interaction of MHD modes, including shear Alfven waves, and beam particles in W 7–AS agree well with experimental observations (Global Alfven Eigenmodes); this supplements work on Toroidal Alfven Eigenmodes in the TFTR tokamak.

IV. CONCEPT IMPROVEMENT

Stellarator theory is being applied to stellarator concept improvement. New designs tailor the Fourier harmonics of the magnetic field to provide: quasi-helical symmetry (elimination of superbanana and orbit losses); reversed drift of trapped particles to stabilize trapped-particle modes; zero bootstrap current operation; and more compact modular stellarator reactor configurations (the modular "helias-like heliac", MHH).

W 7–X is an example of the sophistication of modern stellarator design techniques. The spatial Fourier components of the magnetic field were *chosen* to satisfy desired physics optimization criteria that uniquely specify the geometry of the last closed flux surface: (1) high quality of vacuum field magnetic surfaces (small islands, some shear); (2) good finite-beta equilibrium properties (small configuration change with beta, self-healing of magnetic surfaces): (3) good MHD stability properties (global magnetic well, reduced Pfirsch-Schlüter currents parallel to the magnetic field; (4) small neoclassical transport in the 1/ν regime (small effective ripple); (5) small bootstrap current in long-mean-free-path regime (<10% of that in a comparable tokamak); and (6) good collisionless alpha-particle confinement (confinement improvement with beta, <10% loss). In addition, the coil winding surface was *chosen* to produce practical modular coils: (1) sufficient distance between the last closed flux surface and the coils for an island divertor; (2) acceptable bend radius and maximum field on the NbTi superconducting coils; and (3) sufficient access for heating and diagnostics.

New medium-size experiments now under construction will address important physics issues in development of an improved stellarator concept: the HSX modular stellarator experiment at the University of Wisconsin and the TJ–II flexible heliac at the CIEMAT laboratory in Madrid, Spain. Advances in concept improvement have been facilitated by two developments: (1) the ability to calculate modular coils that produce a last closed flux surface that optimizes a set of selected physics design criteria and (2) computational tools that allow accurate calculations of the 3–D stellarator geometry and accurate fabrication of the stellarator's complex coil set and vacuum vessel. Relatively large helical excursions of the magnetic axis enter into the optimization of the newer magnetic configurations (W 7–X, MHH, HSX, and TJ–II).

HSX will test some of the basic physics (such as drift orbit optimization) underlying the W 7–X experiment and the MHH reactor. It will provide the first test of a stellarator with quasi-helical symmetry in which the particle orbits and neoclassical transport are similar to, but somewhat better than, those in a tokamak. The toroidal curvature component of the magnetic field is reduced to that of a stellarator with an effective plasma aspect ratio »300 (although HSX itself has a physical aspect ratio of 8: R_0 = 1.2 m, a_p = 0.15 m, B_0 = 1 T), resulting in a virtual elimination of all superbanana and direct loss orbits.

The TJ–II flexible heliac (R_0 = 1.5 m, a_p = 0.2 m, B_0 = 1 T) will test the physics of stellarators with a very large helical axis excursion with bean-shaped flux surfaces that rotate about the helical axis. The TJ–II coil set allows exploration of a wide range of magnetic configurations for study of transport and beta limits.

Development of an effective divertor is critical for steady-state operation of LHD and W 7–X and for the viability of both the tokamak and stellarator concepts. The CHS/LHD local island divertor (LID) concept and the W 7–AS/W 7–X island divertor both use magnetic islands, but in different ways; there are also similarities with the ergodic divertor scheme on Tore Supra. The LID scheme shown in Fig. 5 uses an externally produced m = 1, n = 1 island, which avoids the leading edge problem by channeling the particle flux into a pumped limiter. The Wendelstein island divertor (shown in Fig. 6) makes use of the naturally occurring islands at the plasma edge. Experiments are now underway on CHS to test the LID concept. Both magnetic mirrors and the electric fields in the outer regions could strongly affect particle orbit trajectories and affect the

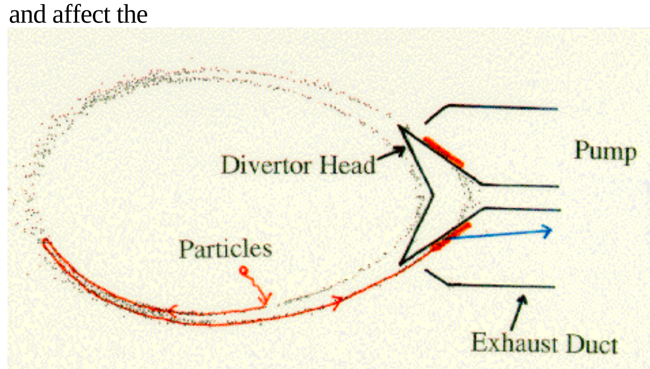

Fig. 5 The local island divertor concept

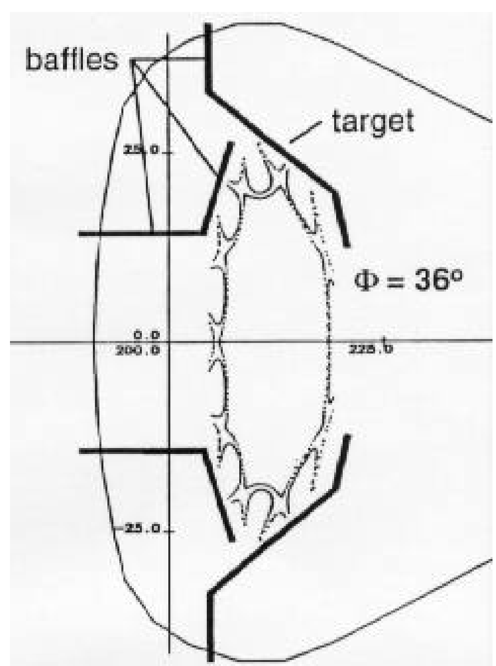

Fig. 6 The W 7–AS/W 7–X island divertor concept

deposition of power on divertor plates. The goal is to develop a model of island-based divertors with ergodic field regions and electric fields.

V. REACTOR DEVELOPMENT

Recent studies have shown that stellarators could be attractive as fusion power plants [5]. Figs. 7 and 8 show two new stellarator configurations: a modular torsatron (MATF) that removes the objection of continuous helical coils, and a modular-coil "helias-like heliac" (MHH) that reduces the size of modular stellarators and leads to a very attractive reactor. The MATF allows a relatively large distance between plasma and the coils for blankets and shielding and field lines can exit between coils to an exterior divertor chamber. Even a large helical-ripple loss region has relatively little effect on the reactor economics through either the direct loss of energetic alpha particles (compensated by avoidance of helium ash accumulation) or ripple-induced thermal losses (reduced by the resulting ambipolar radial electric field).

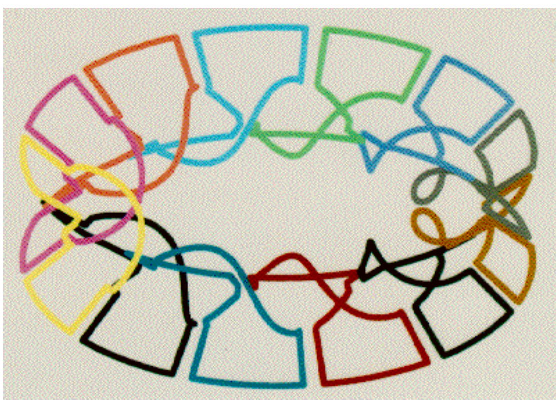

Fig. 7 The MATF modular torsatron concept

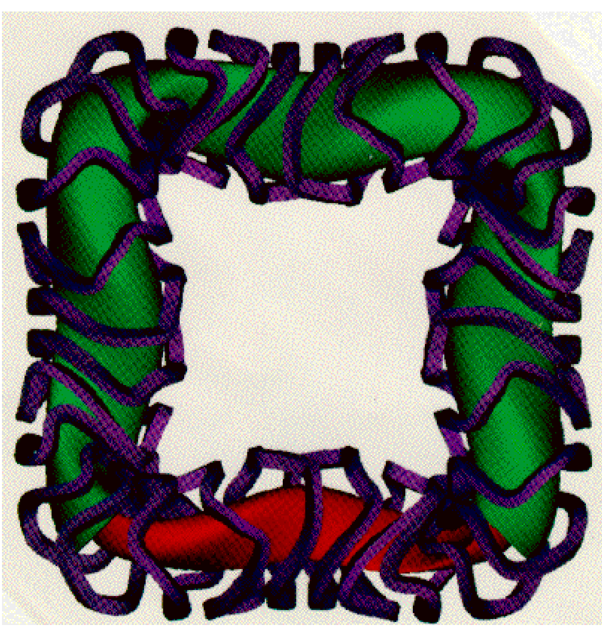

Fig. 8 A top view of MHH, showing the modular coils, the dark blanket and shield under the coils, and the light plasma surface (at the bottom)

A large helical excursion of the magnetic axis provides most of the rotational transform in MHH and allows the modular coils to be farther from the plasma, which provides more space for the blanket, shield, structure, etc. and consequently a smaller major radius (and smaller mass) in a reactor. A plasma-coil gap of 2 m can be obtained in a 14–m stellarator reactor. MHH has excellent physics properties, similar to those of the W 7–X helias. The bootstrap current is somewhat larger than in the W 7–X helias configuration. Numerical nonlinear tests indicates that the plasma stable should be at $\beta = 5\%$. Monte Carlo orbit calculations indicate confinement properties are also similar to that for a helias.

The MHH offers excellent transport and ignition at moderate values of beta ($\beta \leq 5\%$). It was analyzed in the U. S. Stellarator Power Plant study [6] by the same team that produced the recent U.S. ARIES and PULSAR tokamak reactor studies. For a wide range of assumptions, the MATF and MHH stellarator configurations lead to power plants that are economically competitive with the second-stability ARIES–IV tokamak for the same assumptions on materials, costing, and confinement. Table I compares the main parameters for 1–GW(electric) ARIES–IV and MHH power plants. There is a relatively small penalty for lower beta and less confinement improvement [7].

Table I
Comparative Plasma and Device Parameters for1–GW(electric) MHH and ARIES–IV Tokamak Reactors

	ARIES–IV	MHH
Volume-average density (10^{20} m^{-3})	2.9	1.5
Density-average temperature (keV)	10	10
Volume-average beta β (%)	3.4	5
Confinement multiplier H´	2.5	1.4
Neutron wall loading (MW/m^2)	2.7	1.2
Major radius R_0 (m)	6.0	14.0
Plasma volume V_p (m^3)	500	730
On-axis field B_0 (T)	7.6	4.9
Cost of electricity COE (cent/kWh)*	6.8	7.2

* In constant 1992 US dollars

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