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THE U.S. STELLARATOR POWER PLANT STUDY

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The U.S. Stellarator Power Plant Study (SPPS) was carried out by a national reactor study group that also conducts tokamak power plant studies (ARIES, PULSAR). The purpose was to assess the feasibility of a stellarator as a steady-state reactor and to identify critical issues for stellarator reactor design. The same system codes and assumptions (materials and costing approach) used in the tokamak reactor studies were used in this study of a 1-GW(e) stellarator power plant in order to develop a meaningful comparison between tokamak and stellarator reactors.

The short time scale allowed for the study and the limited resources only permitted a first look at a stellarator reactor. Nevertheless, this study indicates that the Modular Helias-like Heliac (MHH) chosen for the SPPS could lead to an attractive steady-state reactor that is competitive with the ARIES-IV second-stability tokamak reactor. Options and tradeoffs were studied for a reference MHH case and areas needing further improvement were identified. No fatal flaws were found; solutions were found in all areas, although the design was not really optimized with respect to the magnetic configuration and reactor engineering.

Concept Selection. Four 1-GW(e) stellarator reactors were compared by minimizing their cost for the *same* physics and engineering constraints: (1) an $M = 6$ torsatron with $R_0 = 12$ m, $B_0 = 4.6$ T, and 2 large continuous helical windings; (2) an $M = 12$ torsatron with $R_0 = 13$ m, $B_0 = 5.5$ T, and 24 modular coils; (3) an $M = 5$ helias with $R_0 = 23$ m, $B_0 = 4.1$ T, and 50 modular coils; and (4) an $M = 4$ helias-like heliac with $R_0 = 13$ m, $B_0 = 5.3$ T, and 32 modular coils. The size (hence cost) was determined by the space between the plasma and the coils needed for the blanket, shield, structure, assembly gaps, etc.

The MHH configuration was chosen for the study because the large helical excursion of the magnetic axis that provides most of the rotational transform ι allows the coils to be farther from the plasma and permits a 2-m distance between the plasma and the coils for blankets, shields, structure, clearance, etc. for a 14-m average major radius. Figure 1 shows the MHH coil set with the last closed flux surface and the intervening blanket and shield. The 32 modular non-planar coils (only 4 different coil designs) are closest in the corners on the inside, which limits the on-axis field to 5 T if the maximum field on the coils is restricted to 14.5 T.

The MHH Plasma Geometry. The MHH plasma geometry has some similarities to the W 7-X helias configuration, but MHH has 4 field periods instead of 5, less triangularity of the flux surfaces, and a larger helical axis excursion. Figure 2 shows the change in the flux surface geometry in half a field period and the helical rotation of the magnetic axis. As in a helias, the plasma separatrix permits a helical divertor geometry on the outboard side. MHH has excellent physics properties, similar to those of a helias with a larger plasma-coil separation, which allows a smaller reactor. Although the bootstrap current is somewhat higher than that in a helias, numerical nonlinear tests indicate that the plasma should be stable at $\langle\beta\rangle = 5\%$ and Monte Carlo orbit calculations indicate confinement properties also similar to that for Helias.

MHD Studies. The MHH configuration was analyzed with the 3-D VMEC, BETA, NSTAB and PIES codes. Equilibrium and stability limits are set by magnetic islands caused by Pfirsch-Schlüter and bootstrap currents. The increase in the plasma-coil distance in the initial MHH coil optimization did not allow eliminating these currents; recently an $M = 3$ configuration has been found that has better properties. The most serious modes are pressure-driven ballooning or interchange instabilities that enhance transport; the reversed shear in MHH stabilizes ballooning modes. VMEC calculations of local stability for ideal (Mercier) and resistive modes shows that MHH is stable at $\langle\beta\rangle = 4.8\%$.

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removed radially through the large access doors. The adjacent modules (containing 3 blanket modules on one side and 2 on the other side) can then be removed tangentially through those doors. The divertor plates can be replaced through the large doors without removing a coil.

Systems and Costing Studies. The most expensive components of the reactor (relative to the fusion power core equipment cost) are the two-component shield (43%), the magnets (28%), and the reactor vacuum systems (11%). Table 2 compares plasma and device parameters for the reference MHH and ARIES-IV reactors. Although the 3-D MHD studies support $\langle\beta\rangle = 5\%$ for MHH, viable reactors can be obtained at lower beta ($\langle\beta\rangle \geq 3\%$). Significant reductions in the COE can be obtained for larger power output since the reactor size does not need to change up to 2 GW(electric).

Table 2. Comparison of 1-GW (electric) Reference MHH and ARIES-IV Reactors

	ARIES IV	Ref. MHH		ARIES IV	Ref. MHH
Vol.-ave. density (10^{20} m^{-3})	2.9	1.5	Major radius, R_0 (m)	6.0	14.0
Density-ave. temp., $\langle T \rangle$ (keV)	10	10	Plasma volume, V_p (m^3)	500	730
Volume-average beta, $\langle\beta\rangle$ (%)	3.4	5	On-axis field, B_0 (T)	7.6	4.9
L-G conf. multiplier, H'	2.5	1.4	Stored magnetic energy (GJ)	93	78
Neutron wall load (MW/m^2)	2.7	1.2	Reactor core mass M (tonnes)	9000	17600
Thermal power (GW)	2.5	1.7	COE, (cent/kWh)*	6.8	7.4

* In constant 1992 US dollars

Summary and Conclusions. Although our studies show that the MHH configuration has the physics properties needed for an attractive steady-state fusion power plant, verification of these findings is needed from the large next-generation stellarators LHD and W 7-X (e.g., divertor physics and transport at higher plasma parameters). More work is also needed to improve the magnetic coil and divertor geometry, to reduce the coil-force support structure, to simplify the blanket and shield modules, and to improve the remote maintenance approach. An $M = 3$ MHH design (shown in Fig. 7), which further increases the space between the coil pack and the plasma and has lower bootstrap current and better MHD properties, was also developed as part of the SPPS program. However, the power plant implications of this configuration have not been developed.

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Figure 1.

Figure 2.

Figure 3.

Figure 4.

Figure 5.

Figure 6.

Figure 7.