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PILOT PLANT: A SHORTENED PATH TO FUSION POWER

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ABSTRACT

Previous fusion reactor studies have focused on the characteristics of fusion reactors in a mature, commercial market, on full-scale "demonstration reactors" as commercial prototypes, and on other engineering development facilities. The projected large size and high capital cost of the development facilities presents significant practical impediments to the development of fusion as a commercial power source. In other technologies, "pilot plants" have been constructed in advance of full-scale facilities. Such plants have had the characteristics of small size, low capital cost, and a limited set of objectives, while still having the integrated performance deemed necessary to gain experience with the operating characteristics of the new technology. A range of possible tokamak fusion pilot plants is considered, having as the primary objective providing requisite fusion power experience to an electric utility prior to construction of a full-scale demonstration reactor. Two approaches are explored, having the characteristics of either net electricity production or only the production of high-grade heat. The effects of choices such as mode of plasma heating and normal vs. superconducting coils are also examined. Since tokamak scaling laws do not seem to permit simply "miniaturizing" the DEMO, fusion pilot plant designs incorporate only certain essential features of a power plant, while leaving the development of other features to complementary, specialized facilities.

1. UTILITY INTERESTS

Electric utilities are most interested in areas for which they would have direct responsibility in building, operating and maintaining a reliable, economic electric power supply system. These would include the following areas:

(a) **Production and Extraction of High-grade Heat.** A pilot plant[1] must at least be able to generate heat suitable for conversion to electricity through a proven steam cycle.

(b) **Operation and Maintenance.** This area is of greatest utility concern. The pilot plant should provide the utility with direct experience in monitoring, accessing and maintaining fusion-specific components, and provide data for the design of subsequent reactors.

(c) **Instrumentation, Control and Protection.** The utility will be especially interested in those aspects that differ significantly from their current experience.

(d) **Safety, Environment and Licensing.** Pilot plant licensing should bring out all the safety, environmental and licensing issues associated with operating a full fusion power system and allow utilities to compare these with fission power systems. Of particular interest are inherent safety characteristics of fusion, the degree of applicability of existing nuclear power regulations, occupational and public exposure, public reaction to tritium, and waste management.

(e) **Fuel Cycle, Tritium Self-Sufficiency, Waste Management and Decommissioning.** Design and operation of the pilot plant should provide experience and data for future reactors.

The pilot plant should incorporate proven applicable utility materials control, quality assurance, and operating procedures. The utilities should provide and train the operating and maintenance staff and, if possible, the pilot plant should be built at a utility power plant site.

2. TOKAMAK FOR NET ELECTRICITY PRODUCTION

A pilot plant whose first priority is net electricity production[2] should have the following features:

(a) Fusion power should be as low as possible to reduce cost.

(b) Burn pulses should be long compared with the current ramp-up and ignition phases.

(c) Large aspect ratio is desirable in order to realize long burn pulses with minimal power.

(d) Non-inductive current drive is undesirable and superconducting magnets are essential to reduce power consumption.

Using the above guidelines, a 250 MWth reactor was scoped out with a major radius of 9 m, aspect ratio of 9 and plasma current of 3 MA, producing 40-50 MW of electricity, about half of which is required to operate the reactor itself. H-mode confinement is assumed and superconducting magnets with maximum field of 15-16 T are used. About half of the 50 MW of alpha power is re-radiated by electrons, while the other half is transferred by the plasma to the walls. Only 5 kW per linear centimeter of the plasma column is transported to the walls, allowing use of a circular plasma cross section with a pump limiter instead of a divertor. It is sufficient to have 5-10 MW of auxiliary power for 15-20 s to reach ignition. The burn duration is about 1000 s using inductive current drive. Owing to high aspect ratio and high poloidal beta, a significant part of the current can be produced by the bootstrap effect. Tritium consumption is estimated at less than 4 kg/y, so that there is no need for a tritium breeding blanket.

3. OPTIMIZATION FOR MINIMUM CAPITAL COST

If the requirement for net electricity production is dropped, smaller (15-50 MWth), less costly pilot plants become possible. In this case, the first priority is given to the continuous production of high-grade heat. Superconducting magnets are no longer required, as the plant is permitted to be a net consumer of electricity. Plasma ignition is also not required.

The TETRA systems code was used to explore a range of concepts, seeking the minimum-cost steady-state tokamak that achieves a prescribed wall load and satisfies the ITER constraints on H-mode confinement and beta limits[3]. The results are given in Fig. 1, showing the minimum cost trend and associated major radius vs. the average neutron wall load for both the copper and superconducting coil systems. Neutral beam injection, assumed for maintaining a steady-state plasma current, produces a significant amount of beam-target fusion. At all wall loads, the copper coil concepts result in lower cost, smaller reactors relative to the superconducting coil concepts. This is primarily due to the lesser shield distance between the copper coil and plasma (30 cm) than in the superconducting case (77 cm). Because of the smaller size, up to 80% of the fusion power is produced by beam-target fusions for copper coil systems vs. only about 10% for the larger superconducting coil systems. For wall loads near 0.5 MW/m^2, the copper coil systems with major radius of about 2 m are estimated at about $1B in direct capital cost. The copper coil systems have lower plasma current, fusion power and injection power relative to the superconducting systems, all of which help reduce cost. However the copper coils have large resistive power losses. Enhancement over H-mode confinement does not significantly reduce capital cost in either case, but beta enhancement can significantly reduce cost in the superconducting systems. The analysis (the upper two curves in Fig. 1) looked at the conventional range of aspect ratios (2.5 - 5), optimizing at around 4-5.

One particular embodiment has a major radius of 2 m, aspect ratio of 4, with 2.62 MA plasma current (bootstrap fraction 32%), and 4T field on axis. The magnets are made of hard copper with demountable sliding joints and designed for steady state operation with water cooling. The plant requires 15 MW of 500 keV neutral beams, 13 MW of ICRF and produces an average wall load of 0.13 MW/m^2 (0.21 MW/m^2 peak). The power input for the toroidal coils is 125 MW.

The lower curve in Figure 1b shows the effect of small aspect ratio (ST), smaller major radius, concepts[4]. Preliminary analysis, assuming physics consistent with initial results from the START experiments[5], suggests a system with major radius of about 0.9 m and aspect ratio 1.6 - 2, with wall loads of 0.25 - 1 MW/m^2. The small size is made possible by eliminating shielding and insulation of the center leg of the copper toroidal field coils. This is expected to lead to large reductions in the fusion core cost, but the reduction of the balance of plant cost is expected to be less significant.

4. VARIATIONS

Consideration has been given to using ICRF rather than neutral beams for heating and current drive in order to increase the "reactor relevance" of the pilot plant technology. The main consequence to the plasma is the loss of beam-plasma fusion. Therefore, for the same device, switching to neutral beams roughly doubles the neutron wall load. For the same wall load, optimizing the device with neutral beams does not significantly affect capital cost, but can reduce the power requirements by perhaps thirty percent. Assuming fast wave current drive according to ITER correlations (but with limited database), steady state operation is achieved.

For neutral beam driven, copper coil systems, the desired wall load is attained primarily via beam-target fusion. This has a strong impact on reducing the required reactor size for a given wall load. If the copper coil shielding could be reduced, this advantage could be further accentuated.

Although analysis of tritium supply shows that there should be a sufficient stockpile after the year 2000 for the needs of both ITER and a pilot plant for a reasonable number of years, incorporation of a tritium breeding blanket or test module in the pilot plant may be desirable for mission reasons.

5. CONCLUSIONS

Fusion pilot plants could provide significant, relevant experience to utility personnel, and useful data for later demonstration and commercial fusion power plants. Analysis suggests such a plant could be constructed at less expense than the costs of other planned fusion development facilities.

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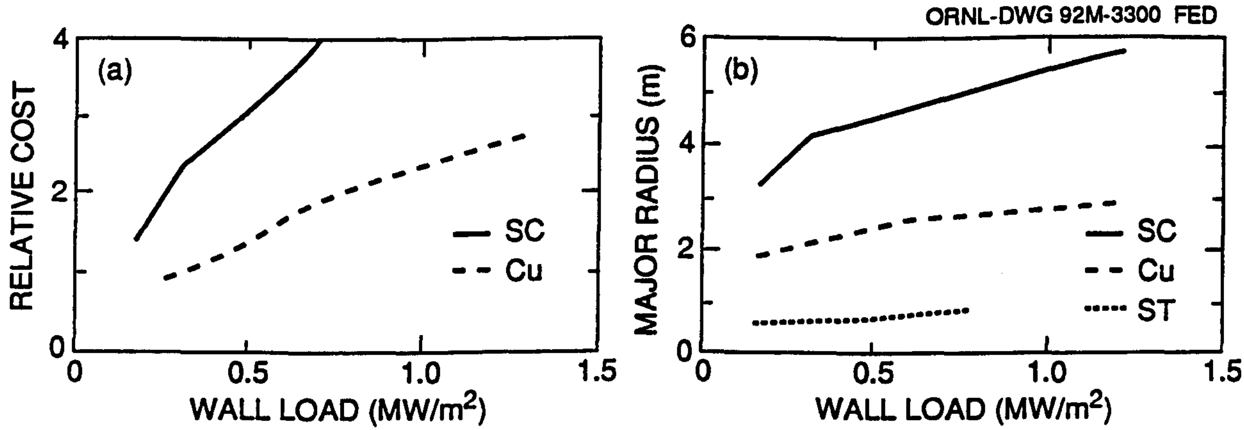

Fig. 1. (a) Minimum cost of Pilot Plants vs. wall load, and (b) the associated major radius. Cases are shown for copper coil (CU), superconducting coil (SC) and low aspect ratio small tokamak (ST), devices.