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THE PRINCETON REFERENCE DESIGN
FUSION POWER PLANT

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The Princeton Reference Design Fusion Power Plant

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ABSTRACT

The Princeton Reference Design Fusion Power Plant is an electrical generating station based upon a large Tokamak confining a D-T plasma. The plasma major and minor radii are nominally 10.5 m by 3.2 m; the toroidal field at the center-line is 6 T. With a mean ion temperature of 30 keV, an average β of 0.04, and an average τ of 3.83 s a fusion power of 3440 MW is achieved during 97 minutes of the 100 minute long-pulse cycle. The plasma throughput of 2.8×10^{22} ions/s is deflected down a divertor channel where it cools by radiation and is neutralized, exhausting as hot gas to the vacuum pumps.

The total thermal power of 5305 MW is carried by a helium coolant to a conventional steam cycle producing 2030 MWe net. Tritium is bred in an 80 cm thick blanket of eutectic flibe (53.1% beryllium fluoride and 46.9% lithium fluoride) with Nimonic PE-16 (43% nickel, 39% iron, 18% chromium) as the structural material; the blanket-averaged volume fractions are 69.8% flibe, 9.4% PE-16, and 20.8% helium. The breeding ratio of 1.04 will allow doubling of the 2.57 kg inventory in 121 days. The nominal wall load of 1.76 MW of DT neutrons per m implies a total flux of 8.65×10^{18} n/m²·s at the wall and a corresponding helium production rate of 275 ppm/year.

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INTRODUCTION

During the past two years the Reactor Studies Group of the Princeton Plasma Physics Laboratory has been engaged in the conceptual design of a fusion-based electrical generating station. The study has been completed; this paper summarizes the guiding principles and the design features incorporated in the resulting power plant. A report [1] is available that describes the Reference Design in great detail.

The basic motivation for this project was a desire to determine what, other than basic plasma physics, might delay the introduction of fusion power. Consequently the ability to confine a plasma was taken for granted, and the various technological aspects of a power plant deriving its energy from fusion have been investigated. An attempt has been made to relate each part of the design to the others in a self-consistent manner so that advantages are not claimed at the expense of hidden problems. It is hoped that studies such as this one will identify those areas of technology that require special development in parallel with the plasma physics research effort.

In keeping with this, a principal guideline has been to use current technology wherever possible (where "current" allows for reasonable developments in the fission reactor program). Thus the Princeton Reference Design represents a first-generation power plant rather than an advanced, optimized one (as, e.g. in Ref. 2). It should provide a basis of comparison with other proposed fusion devices [3,4] as well as with the other advanced energy systems proposed for the turn-of-the-century.

COILS AND PLASMA

The reactor in the power plant is assumed to be a large Tokamak, because of the recent advances in plasma confinement in such devices. The nominal plasma major and minor radii are 10.5 m by 3.2 m. The toroidal field, provided by 48 large D-shaped coils, is 6 T at the centerline and reaches a maximum of 16 T at the inside of the torus. The D-coils are made of 1600 km of Nb₃Sn superconducting ribbon dynamically stabilized with copper. The use of Nb-Sn is required to achieve the 16 T peak toroidal field, but it would be justified regardless because of its lower projected cost (relative to Nb-Ti). Nb-Sn has been used for all the other coils in the Reference Design. The toroidal field coils carry 315.4 MA-turns of current and store 250 GJ of magnetic energy. A central support column bears the centering force of 50.8 Gg per coil.

The 14.6 MA plasma discharge current is maintained by changing the magnetic flux through the plasma ring. This is done by varying the current in a set of coils wrapped the "long way" around the torus outside the toroidal field coils. The vertical stabilizing field is produced primarily by

another set of coils wrapped around the torus near the plasma, inside the D-coils. The poloidal field configuration with its single-null separatrix is produced by these coils together with the plasma current and another pair of elongated coils near the divertor.

The positions and relative currents in all of these poloidal field coils are indicated in Figure 1. Altogether they require 860 km of Nb-Sn ribbon, carrying 112.1 MA-turns of current. $4.55 \times 10^5 \text{ m}^3$ of helium (STP) is required as the cryogenic coolant fluid. The refrigeration load of 280 kW at 8 K is due to thermal conduction (150 kW), neutron and gamma heating (95 kW), and eddy currents from the field pulsing (35 kW).

The reactor is intended to operate on a long-pulse cycle with 97 minutes of quasi-equilibrium burn and 3 minutes for shutdown, field reversal, and re-ignition. Figure 2 shows the plasma temperature profiles during the burn phase, with corresponding density profiles in Figure 3. The mean ion density is $5 \times 10^{19}/\text{m}^3$; the mean ion temperature is 30 keV. Operation in a thermally stable regime with this low temperature is attained by admixture of up to 4.8% argon, which enhances the bremsstrahlung. The average β is 4%, with a maximum of 13%; the average β_0 is 1.83. The nominal safety factor q is 2.067.

Since the mean ion confinement time is assumed to be only 3.83 seconds (several hundred Bohm-times confinement) a continuous fuel feed is required. This is attained by the high velocity ballistic injection of solidified fuel pellets of a variety of sizes, chosen to produce a desirable fuelling profile. The fuel requirement is 1.40×10^{22} tritons/sec, an equal rate of deuterons, and 0.134×10^{22} argon atoms/sec. The DT reaction rate is $1.22 \times 10^{21}/\text{sec}$, for a burnup of 8.7% per pass.

The boundary condition for the main plasma is set by the temperature at the separatrix, around 500 eV. Ions diffusing across the separatrix are guided by the divertor field lines into the arms of the divertor channels (see Figure 4) where they radiate most of their energy to the walls. This flow eventually neutralizes and enters the exhaust chamber as a hot supersonic gas, thereby decoupling the plasma from the vacuum pumping system. Another main function of the divertor is to shield the plasma from impurities by ionizing and sweeping away any neutrals which might be knocked off the walls of the chamber.

The placement of the single null of the separatrix on the plane of symmetry is optimal from the standpoint of providing proper poloidal field curvature for MHD stability. It also allows placement of the exhaust chambers in the elongated ends of the moment-free D-shaped toroidal field coils. This configuration contributes, however, to the large

size of the reactor, because of the need to provide sufficient shielding between the divertor channels and the cryogenic coils. (Of course, the scalings of gross power, wall load, and capital cost tend to increase the size also.)

BLANKET

The choice of DT fusion entails the production of tritium by reactions of the DT neutrons with lithium. Accordingly a blanket of flibe, cooled with helium, with a "stainless steel" structure surrounds the plasma. This set of materials was felt to be a practical combination and a more conservative choice than the previously accepted refractory metal structure with molten lithium as both breeder and coolant.

The blanket is 80 cm thick and consists of a large tank of flibe, fronted by a vacuum and (plasma) radiation shield wall (at $r=360$ cm), and penetrated by tubing for the helium. The flibe composition is the eutectic (53.1% beryllium fluoride and 46.9% lithium fluoride), to allow the lowest inlet coolant temperature. The structural material is Nimonic PE-16, an austenitic alloy (43% nickel, 39% iron, 18% chromium) with good fabricability, corrosion resistance, high-temperature strength, and radiation damage resistance. The helium is pressurized to 5.2 MPa (50 atm) to improve its heat transfer properties. The blanket-averaged volume fractions are 69.8% flibe, 9.4% PE-16, and 20.8% helium, with lower fractions of flibe near the wall where greater cooling is required.

The nominal wall load of 1.76 MW/m^2 of DT neutrons establishes the source condition for calculations of the neutron flux in the blanket; spectra at several points are shown in Figure 5. The total flux at the wall, $r=360$ cm, is $8.65 \times 10^{18} \text{ n/m}^2 \cdot \text{s}$, of which $3 \times 10^{18} \text{ n/m}^2 \cdot \text{s}$ is due to neutrons with energies above 1 MeV. This intense flux will cause the production of 275 ppm/year of helium in the PE-16, constituting a very severe radiation damage problem. It will also cause a substantial transmutation of the structural material, but fortunately few of the activation products have very long halflives.

The net tritium breeding ratio for this blanket is 1.04, which is more than adequate. The total inventory of tritium is only 2.57 kg (of which 2.02 kg is in the fuel feed reserve), so the projected doubling time is 121 days. The parasitic loss of neutrons to the structural material and the fluorine is compensated for by neutron multiplication in the beryllium; excessive breeding is traded-off for extra energy production in the form of capture gammas.

In the Reference Design the gross thermal power is 5305 MW, of which 688 MW derives from the fusion alphas and is radiated to the plasma chamber walls. The remainder is nuclear heating in the blanket; there is a 68% multiplication of the 14 MeV kinetic energy of the fusion neutrons.

BALANCE OF PLANT

There are two principal loops out of the reactor blanket. The first passes the flibe through gas disengagers (spray nozzles into evacuated tanks) at a circulation rate of about one turnover every two hours. The bred tritium is removed in this cycle as tritium fluoride, which is electrolyzed, purified, and sent to the fuel reserve system. The slow flow of the flibe enhances heat transfer to the helium coolant, and also feeds a "salt-pot" thermal flywheel which sustains the reactor output during the 3 minute plasma shutdown.

The helium flows through the blanket at a rate of 3.24 Mg/s, exiting at 910 K (1180°F), the operating limit of the structural material. An oxygen pressure of 5×10^{-3} atm is maintained in the coolant helium to promote the oxidation of any tritium which permeates the coolant tubes. The vapor pressure of the resulting water vapor is held below 10^{-5} atm by passing a drag stream through dessicants, thereby keeping the tritium activity in the helium very low. By this scheme a chemical barrier greatly reduces the loss of tritium from the reactor by permeation through the steam generators; such losses should be less than 6 Ci/day.

A conventional steam cycle is used to produce electricity, although the option of using direct-cycle gas turbines would seem very attractive. Primary steam is developed at 25.4 MPa (3690 psi) and 810 K (1000°F) in an HTGR-type steam generator. Two reheat stages regain the same temperature, but at lower pressures. The steam drives a cross-compound 3600 rev/min turbine generator combination with one very high pressure, one high pressure, two intermediate pressure, and four low pressure turbines on two 60 meter shafts. The gross electrical conversion efficiency is 43.4%. However, the station internal requirements are 375 MW, of which 251 MW are for the helium circulators, leading to a net plant efficiency of 38.3% and net output of 2030 MWe.

The total capital cost of the Reference Design (in 1974 dollars) is estimated to be \$1 215 230 000, or roughly 599\$/kWe. Of this, only \$403 050 000 is due to the nuclear island. The cost of busbar energy is estimated as 15.5 mills/kW·hr, based on a 15% return on investment, 85% plant availability, 3.1 mills/kW·hr for operation and maintenance, and negligible cost of fuel.

CONCLUSIONS

A principal purpose of this study was to identify research areas which would be most important for the early introduction of fusion power. To this end, current technology was exploited wherever possible, so that essential extensions would stand out. These include the following: that large size high-field Nb-Sn superconducting coils can be built; that a long-pulse tokamak can be continuously fueled; that the divertor will screen out impurities and channel out the exhaust; that the radiation damage resistance of austenitic materials can be improved by an order of magnitude; and that fail-safe sealed systems for processing large amounts of tritium can be developed.

Fortunately, none of these problems or the others encountered seem unsolvable on a realistic development timescale (with a prototype plant in the mid 1990's).

ACKNOWLEDGMENTS

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Fig. 1. Magnetic flux surfaces and poloidal field coils.

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Fig. 2. Temperature profiles

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Fig. 3. Density profiles

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Fig. 4. Cross section of the nuclear island.

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Fig. 5. Neutron flux spectra.

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