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A HIGH-FLUENCE FUSION NEUTRON SOURCE

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Abstract

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A conceptual design of a D-T fusion facility for continuous production of 14-MeV neutron wall loading from 5 to 10 MW/m² at the plasma surface is presented. In this design, D-T neutrons are produced in a linear, two-component plasma formed by neutral beam irradiation of a fully ionized warm plasma target. The beam energy, which is deposited in the center, is transferred to the warm plasma mainly by electron drag and is conducted along the target plasma column to end regions where it is absorbed in neutral gas at high pressure. The target plasma is operated in a regime where electron thermal conduction along the column is the controlling energy-loss process. The loss rate is minimized by adjusting the diameter and length of the plasma column. A substantial gradient in T_e along the column results in recombination of the plasma to gas in the end regions before impact on the end walls. The resultant hot gas is cooled by contact with large-area heat exchangers. In this way, the large steady-state heat load from the injected neutral beams is diffused and removed at tolerable heat flux levels. The reacting plasma is essentially an extrapolation of the 2XIIIB high- β plasma to higher magnetic field, ion energy, and density.

1. Introduction

This report describes a conceptual design of a high-fluence source of 14-MeV D-T neutrons for accelerated testing of materials. Fusion materials development is one of the four key technical issues identified in the Magnetic Fusion Program Plan [1]. Typical fusion reactor goals, which are relevant to materials development, are a minimum integrated neutron wall loading of 10-20 MW · yr/m², a minimum first wall lifetime of 2-5 yr, and a minimum neutron wall loading of 3-6 MW/m² [2]. A radiation dose of ~100 displacements per atom (dpa) in the 14-MeV neutron environment is necessary to predict materials performance at the minimum lifetimes considered for fusion applications. The design goal (10 MW/m²) of the high-fluence neutron source was chosen to provide the required 14-MeV neutron radiation dose in approximately 1 yr. We preferred a fusion-based source for this application because (1) a relevant 14-MeV D-T neutron spectrum is obtained and (2) a fusion source

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better simulates the reactor environment where materials can be subjected to high thermal loads, energetic particle irradiation, high mechanical stresses, intense magnetic fields, and high magnetic field gradients. The small-volume, high-fluence source we propose complements the capabilities of a facility such as ITER.¹ The present ITER design goals of a 14-MeV wall load $> 1 \text{ MW/m}^2$ and 25% availability are adequate for development of nuclear technology such as blankets but are not adequate for materials development. For our source, the volume of reacting plasma and the fusion power have been minimized (0.7 MW), while maintaining an intense neutron wall loading. As a consequence, tritium consumption is modest, and the amount of tritium required is readily available. In this preliminary study, we have not specifically addressed costs for the proposed facility; however, a major cost item is certainly the power drive system, both in capital investment and operating cost. We endeavored to minimize the beam power required and calculated suitable parameter regimes at a beam power of $\sim 50 \text{ MW}$. An optimum neutral beam energy ranges from 200 to 250 keV, requiring a negative-ion based source. Alternatively, with only slight degradation in performance, beams at 150 keV based on existing positive-ion sources could be used.

2. Neutron Source Concept

In our design, D-T neutrons are produced in a linear, two-component plasma [3] formed by neutral beam irradiation of a fully ionized warm plasma target. The beam energy deposited in the center is conducted along the target plasma column to end regions where it is absorbed in neutral gas at high pressure. To minimize power loss at the ends, we rejected convection-dominated approaches in favor of a conduction-limited power flow. In the end regions, the resultant hot gas is cooled by contact with large-area heat exchangers. In this way, the large steady-state heat load from the injected neutral beams is diffused and removed at tolerable heat flux levels. A similar approach to handling large heat loads was proposed in tokamak divertor design, and an experimental simulation of a gaseous tokamak divertor was described by W. Hsu et al. [4].

The basic elements of the neutron source design are shown schematically in Fig. 1. There are three well-defined plasma regions: the central neutron production region, a power transport region, and a gas-heat exchange region. In the central region, high-energy neutral beams of deuterium atoms are

¹An international group with representatives from the EURATOM, Japan, USA, and USSR is preparing a proposal for a joint conceptual design study of a large D-T tokamak called the International Thermonuclear Experimental Reactor (ITER).

Figure 1: Schematic of the high-fluence neutron source.

ionized in the warm, dense tritium target plasma with the resulting hot deuterons trapped in a quadrupole-mirror field.² As the deuterons slow down by collisions with the target electrons, they interact with the tritium to generate the desired D-T fusion neutrons. At electron densities of about 10^{18} cm^{-3} , the characteristic loss time of hot-ion energy to the cold-electron population (τ_{drag}) is $\sim 100 \mu\text{s}$ and is much shorter than the ion scattering time (τ_{ii}). Consequently, the hot-ion density remains localized in the center, requiring only a low mirror-ratio quadrupole for confinement. This central minimum-B region of high $P_{\perp}$ provides the necessary MHD stability for the entire device since plasma pressures are low elsewhere. The reacting plasma is essentially an extrapolation of the 2XIIB high- β plasma to higher magnetic field, ion energy, and density. In the extrapolation, care has been taken to preserve scaling and plasma attributes contributing to equilibrium, MHD stability, and microstability in 2XIIB [5].

The trapped deuterons degrade in energy until nuclear interaction becomes improbable; they

²Although we specify a quadrupole, we do not rule out the possibility of axisymmetric stabilization, e.g., by a "divertor" coil as tested on TARA. Coil simplicity and improved access would be worthwhile changes provided performance, i.e., β -limits, is maintained.

continue to cool until they join the warm target population and eventually diffuse out the ends of the device. The heated electrons share their energy by frequent collisions with target ions, so $T_i \approx T_e$ in the warm plasma.

In the power transport region, the plasma is collisional so conduction losses from the center are reduced by the ratio of the magnetic field (hence area) at the center to that in the transport section. The maximum practical field is used in the transport region to minimize conduction. Power requirements are further reduced by lengthening this section. Radial transport of warm plasma ions due to classical collisional diffusion and mobility in the electric field is estimated to be, at most, 7.5×10^3 A, corresponding to a maximum radial power flow from the entire plasma column of 1.5 MW. Radial loss of hot ions due to scattering on the warm plasma target results in a radial power transport of ~ 0.1 MW. We expect that radial particle losses will be conducted along the vacuum chamber walls to the end chamber and recycled into the warm plasma column. Thus, we assumed one-dimensional power flow in the direction of the magnetic field. Steady-state, local values of density and temperature are assumed to remain constant. A substantial gradient in T_e develops along the column but the warm plasma pressure (nkT) is assumed to be constant along the plasma column and to be balanced by the neutral gas pressure of about 0.2 atm in the end chamber. In the end regions, the now cold, dense plasma is flared to a thin cross-sectional cusp, which increases the area for plasma-gas interaction and improves the power transfer to the gas, thus preventing plasma from reaching the wall. In turn, the energy is deposited by turbulent flow of the gas over water-cooled, heat-exchange panels. To keep power loads below 250 W/cm^2 , the heat-exchanger area must be at least 10 m^2 in each end tank. The pump and turbulent flow in the end chambers in Fig. 1 only schematically represent the heat exchanger. If tritium is used for the warm plasma target, the tritium inventory in the machine will be about 120 g, with additional tritium in the reprocessing loop. Using tritium beams and a deuterium target will reduce the tritium inventory to about 110 g with most of it in the reprocessing loop. In the latter operating mode, the neutron output will be reduced by approximately 20% for given beam power.

3. Plasma Modeling and Parameter Study

The desired neutron wall loading of $5\text{--}10 \text{ MW/m}^2$ is achieved at minimum beam power by adjusting the many parameters of the device to an optimum combination that is consistent with

technological and physics constraints. In setting the parameters, our goal is to minimize extrapolation of well-established physics and to keep the technological requirements to the level of existing capability, or at most, to the level of development needed for ITER. First, for a given magnetic field and plasma β , the maximum reaction rate (hence neutron flux) is realized in the two-component mode with one hot species and one warm species of equal density. The volume of reacting plasma is minimized by limiting the length to the size of the neutral beam, and the diameter to $\sim 6 \bar{\rho}_i$, or approximately 3 average ion gyro-orbit diameters. For practical central magnetic field intensities of ~ 4 T, this diameter restriction requires a well-focused neutral beam. A central minimum-B-type magnetic field of 4 T, mirror ratio 1.2, can be generated using superconducting circular coils with a quadrupole copper insert (Fig. 1). The copper insert need not be shielded from the neutrons but requires about 7-MW drive power. The neutron contribution to the quadrupole thermal load is a maximum of 80 W/cm^2 incident on the magnet. This energy is deposited in layers about 5 cm deep and is not a problem. To minimize the heat flow along the plasma column, the column diameter is minimized by using the maximum practical intensity for the guide field. Superconducting solenoids of suitable diameter have been tested to 12 T, which we use for our solenoidal guide field. The length of the guide field was selected as 5 m (each end) from a parameter study that showed diminishing improvement beyond this length. Total plasma density is constrained because the neutral beams must penetrate to the axis. We chose the product of density and plasma radius at the beam-injection position to be equal to $2\lambda_{trap}$, where λ_{trap}^{-1} is the effective cross section for beam trapping. This choice produces a reasonably flat radial profile and the beam is almost completely absorbed before leaving the target. The beam-trapping estimate includes a correction for multistep collision processes that increase trapping rates [6]. The hot component of the plasma essentially determines the plasma pressure. Our parameter study shows improved performance as a function of increasing β , with β of 0.4 to 0.6 required if beam power is held to ~ 50 MW. In our modeling calculations, we determine β and beam power in relation to neutron flux.

The power flow from the central plasma to the ends is based on Spitzer [7] thermal conductivity:

$$P_{cond} = \frac{4\pi a_s^2}{l_c} \int K dT_e, \quad (1)$$

with $K = 9.3 \times 10^8 T_e^{5/2}$ (keV). In this expression, l_c is the full length of the plasma column and a_s is the radius in the 12-T power transport region. We use cgs units, except energy is in keV. The Spitzer

conductivity model for electron power balance only applies for $\lambda < l_e/2$, where λ is the electron-electron mean free path $\lambda = 2 \times 10^{19} T_e^2 / (n_e \ln \Lambda)$. The nominal operating point for our present neutron source lies well within the Spitzer conductivity region. Newton [8] finds good agreement between thermal conduction models with both present field-reversed configuration (FRC) experiments and past θ -pinch experiments for T_e from 40 to 900 eV. Assuming the power loss is by thermal conduction, we can equate this loss to the power input to the plasma through electron drag of injected deuterons:

$$P_{drag} = V \frac{n_h n_e \bar{E}_h e}{(\pi r)_{drag}}, \quad (2)$$

where V is the hot plasma volume; n_e is the total plasma density, which is equal to $n_h + n_w$; n_h is the hot-ion density; $\bar{E}_h$ is the average hot-ion energy; e is the electron charge; and $(\pi r)_{drag} = 1 \times 10^{13} A_i T_e^{3/2} / \ln \Lambda$, where $\ln \Lambda = 11.6$ for deuterons. Equating $P_{cond} = P_{drag}$, we obtain:

$$T_e \text{ (keV)} = 9.72 \times 10^{-9} (n_h \bar{E}_h n_e l_h l_c R)^{1/5}, \quad (3)$$

where l_h is the length of hot plasma, and R is the ratio of field in solenoid to central field. The hot-ion density is set equal to $n_e/2$ to maximize the reaction rate unless a β -value of 0.6 limits n_h to a lower density.

The neutron wall loading Γ is calculated from a flat-profile, cylindrical model and is quoted at the plasma surface of radius a .

$$\Gamma = n_w n_h <\sigma v> e E_f \frac{V}{S}, \quad (4)$$

where E_f is the neutron fusion energy released per reaction, n_w is the density of the tritium target, V is the hot-plasma volume, and S is the hot-plasma surface area.

For our parametric scaling studies, a midplane magnetic field of 4 T was chosen so that for optimum injection energy (~ 200 keV), a plasma with a radius of 3.75 cm (approximately $3 \bar{\rho}_i$) corresponds to the anticipated dimension of a well-focused neutral beam. The average gyro-radius $\bar{\rho}_i$ for the hot-deuterium ions was determined from the average hot-ion energy of ~ 60 keV at the midplane.

For our analytical model, we used plasma parameters based on square-well Fokker-Planck calculations of the hot-ion distribution function using the multispecies Hybrid II code. This code [9] was used to calculate the hot-ion distribution function and average values for $<\sigma v>_{DT}$ and the hot-ion energy, $\bar{E}_h$. In the Fokker-Planck calculations, T_e is an input parameter, so we must iterate

between the analytic expression for T_e and the code to obtain a consistent set of values for T_e , $\tilde{E}_h$, and the other related variables. We also modeled the plasma with the Z -dependent SMOKE code [10] to verify localization of hot ions. Assuming the hot-ion distribution is fixed by neutral beam injection in the central location, we also investigated the formation of self-consistent axial density and temperature profiles using the PHLOW code [11] with the addition of the axial conduction model. Our results are consistent with the analytical model presented earlier. Also, the cold ions remain Maxwellian with $T_\perp \simeq T_\parallel$ over the length of the device, and the electron temperature decreases along the axis with a more precipitous drop at the location where the tritium gas is recycled.

To determine a length for the solenoidal power transfer region, we calculated neutron wall loading and power vs total length, l_c , of the plasma column. These estimates were done for $B_0 = 5$ T, at three injection energies. Power decreases with l_c ; the gain beyond $l_c \sim 10$ m (5 m each side of center) is small. For our purposes, we chose $l_c = 10$ m as the standard length. In selecting an optimum beam energy, we plotted beam power and β vs beam energy for a fixed value of $\Gamma = 8$ MW/m². When we chose $\beta = 0.6$ and arbitrarily limited beam power to ~ 50 MW, the maximum wall loading was obtained for beam energy from 200 to 240 keV. Studies of an intense neutron source by Ryutov and Mirnov [12] also set the optimum injection energy at 240 keV. For our purposes, we selected 200 keV as a base-case injection energy. If technological or economic considerations set the beam energy at either higher or lower values, interesting neutron fluxes could still be realized. A beam energy of about 150 keV, for instance, might allow use of positive-ion beam technology with substantial savings in beam development. Selection of $B_0 = 4$ T for the central field intensity is based on meeting practical requirements for the copper quadrupole insert, in terms of beam access and magnet drive power. Figure 2 shows improved performance in Γ as B_0 is increased although the improvement is less dramatic if Γ is specified at a fixed wall radius (6 cm). More refined trade-off studies will be required to set B_0 .

We estimated heat loads on the central-cell walls resulting from charge-exchange processes for the base-case parameters. For these calculations, the ambient gas pressure in the central volume is estimated self-consistently to be 7×10^{-5} Torr, with charge exchange as the only gas source and plasma pumping as the only gas-removal process. Including charge exchange with both beams and gas, total heat loads on a wall placed at $r = 7.5$ cm are less than 250 W/cm² and can be handled with

Figure 2: Neutron wall loading, Γ , and beam power, P_b , as a function of central field intensity, B_0 . Γ_s is measured at the plasma surface; Γ_6 , at a radius of 6 cm. $\square$ denotes nominal operating point.

conventional cooling. The distribution of power deposition (0.14 MW) by α -particles depends on the exact location of the center chamber walls. However, as α -particles spiral in the 4-T magnetic field with a maximum orbit diameter of 13.5 cm, much of this power is transported along the magnetic field until the α -particles encounter a wall. Assuming all α -particle power is deposited on radial walls, which are located in 4-T magnetic fields, the average wall load would be 124 W/cm² and is easily removed. The base-case dimensions of the reacting plasma are 7.5-cm diameter and 30-cm length. Figure 3 illustrates the access for material test probes in radial cross section. Figure 4 shows neutron flux contours in axial cross section for the base-case parameters. Allowing for neutral beam injection, about one-half of the circumference is available for samples. Maximum neutron flux will occur near the midplane of the plasma, as close to the plasma surface as possible. With a practical wall radius of 6 cm, a test volume of about 180 cm³ lies between a 6- to 7-cm radius and 5 cm either side of the midplane. Substantially larger volumes at reduced flux are available at larger radii. Thus, many small samples (few cm³) could be irradiated at high flux, while experiments such as weld or

Figure 3: Schematic of machine midplane showing access for material test samples.

thermomechanical response tests requiring $\sim 10^3 \text{ cm}^3$ could be accommodated at slightly lower flux levels.

4. Magnet Design

We completed a preliminary layout of magnets to match the requirements of our neutron source parameters. All solenoidal magnets are specified as superconducting, and sufficient clearance has been introduced to allow for neutron shielding. The quadrupole insert, made of copper, is meant to operate without shielding and to be replaced periodically.

Since we have not included recircularizing coils outside the quadrupole, the flux bundle remains elliptical along the solenoidal region. The ellipticity at the mirrors is 10:1 (major to minor axis ratio). At present, we see no objection to an elliptical flux bundle vs a cylindrical shape. The large solenoidal magnets have a mean radius of 1.5 m, and the smaller solenoids have a radius of 0.6 m. Clear bore of the smaller coils is $\sim 0.9 \text{ m}$, compared to the plasma column of $\sim 0.7 \times 7 \text{ cm}$. The quadrupole has a clear vertical bore of $\sim 28 \text{ cm}$, with extended clearance in the axial direction. Our nominal beam size is $7.5 \times 30 \text{ cm}$, so it appears feasible to inject several beams through a given quadrupole opening.

Figure 4: Contours of neutron flux, ϕ , in the axial cross section: $\phi = n/m^2 \cdot s$, and $F = MW/m^2$.

Detailed layouts have not been completed. The power required for quadrupole operation, as specified, is 6.8 MW, which seems modest and can be handled by standard water cooling (conductor packing fraction is 0.65). Trade-off studies involving beam and/or probe access and cost of operation could change the design.

The treatment of the magnetic field in the end regions also must be considered in detail. We may need to add a magnet to flair the plasma into a thin fan for better interaction with the gas; however, this should not affect the portion of the field already specified.

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