D-^{3}He Tokamak Reactor

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UWFDM-908

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1. Introduction

Today, the worldwide controlled fusion program is concentrated on employing the D-T fuel cycle because it is the easiest reaction to achieve fusion in the laboratory. The fusion performance in terms of $\mathrm{n}\tau_{\mathrm{E}}\mathrm{T}$ has been increased by two orders of magnitude in the last five years. The achieved plasma conditions are closest to those needed in the International Thermonuclear Experimental Reactor (ITER).

However, there are significant disadvantages with the D-T fuel cycle stemming mainly from the fact that 80% of the reaction energy is released in the form of neutrons and there is a need to breed, control and contain a large amount of radioactive tritium. The 14.1 MeV neutron irradiation causes severe damage to structural components and induces a significant amount of radioactivity in the surrounding structures.

D-^{3}He fuel is the most likely advanced fuel candidate because it produces only charged particles which are contained by the magnetic field. The fraction of the reactor power associated with neutrons from D-D reactions is much less than it is in the D-T reactors. Consequently, a D-^{3}He reactor should have significant technological and ecological advantages compared with a DT reactor.

The goal of the present paper is the investigation of D-^{3}He tokamak reactors based on the presently achieved plasma physics characteristics and on the acceptable present day engineering solutions. A modest magnetic field D-^{3}He reactor design and a high-magnetic field design (Apollo) are studied and a comparison of issues is done.

2. Modest-Field D-^{3}He Reactor Design

Based on experimental data from the DIII-D tokamak with a poloidal divertor [1] and on the ITER scaling law for the energy confinement time, the reactor plasma parameters were optimized with respect to the major radius R, aspect ratio $A = R/a$, $\langle\beta\rangle$, elongation κ, toroidal magnetic field B_0, and the plasma density and temperature profiles.

The optimal helium-3/deuterium density ratio in the plasma is equal to $n_{He-3}/n_D \cong 0.7$ (41% ^{3}He, 59% D). The optimal temperature at the maximum Q-value is equal to about 40 keV ($Q = P_{FUS}/P_{AUX}$, where P_{FUS} is the fusion power and P_{AUX} is the power injected into the plasma).

The effect of plasma density and temperature profiles was studied. The control over the plasma profiles allows one to change the mode of the reactor operation.

The major plasma and reactor parameters for this design are given in Table 1. The value of $\langle\beta\rangle$ is chosen to be equal to 18%, proceeding from the results achieved at DIII-D [1]: $\beta_0 = 44\%$, $\langle\beta\rangle = 11\%$. Steady-state current drive by injection of the power $P_{CD} \cong P_{AUX}$ into the plasma, and the bootstrap current fraction is 0.56 I_P.

The presence of the current between the divertor plates and the chamber walls on JET [2] and on DIII-D [3] confirms the opportunity to use direct energy conversion. At the fusion power $P_{FUS} = 2600$ MW the power of cyclotron and bremsstrahlung radiation was estimated as $P_C + P_B \cong 1300$ MW, and the power carried out by electrons and ions through the separatrix surface $P_S + P_S^E + P_S^I \cong 1300$ MW. At a P_S^E/P_S^I ratio equal to 1/1 $P_E^S \cong P_S^I \cong 650$ MW. In principal the ion power fraction through the separatrix, P_S^I, can be converted to electricity with high efficiency due to a locally non-ambipolar particle transport in the divertor layer across the magnetic field [4], as was observed in DIII-D.

The power $P_S^E \cong 650$ MW will enter the divertor plates. The ratio between the power to the outside divertor plate and that to the inside one will be $P_{OUT}/P_{IN} \cong 2:1$, i.e. $P_{OUT} \cong 430$ MW, and $P_{IN} \cong 220$ MW. In operation with a gaseous divertor the power density to the divertor plates may not exceed 10 MW/m^2. This means that the area of a collecting surface should be $S_D = 2\pi R\Delta_D \cong 50\Delta_D m^2$, where $\Delta_D \cong 430$ MW/(10 MW/m^2 $\times$ 50 m) $\cong 0.9$ m, i.e. the height of the divertor chamber should be about 1 m.

In the case of a 0.7:1 ^{3}He:D density ratio, the total power in D-D and D-T neutrons is about 4-5% and the average blanket power multiplication factor is $\sim$2. Since

Table 1. Key Parameters of D-^{3}He Tokamak Reactor

<table>
<tr><th></th><th colspan="2">Modest-Field Design</th><th colspan="2">High-Field Design (Apollo)</th></tr>
<tr><th></th><th colspan="2">Direct Conversion Included</th><th>Direct Conversion Included[1]</th><th>Thermal Conversion Only[2]</th></tr>
<tr><td>Major radius, m</td><td colspan="2">8</td><td>7.89</td><td>8.37</td></tr>
<tr><td>Aspect ratio</td><td colspan="2">2.5</td><td colspan="2">3.15</td></tr>
<tr><td>Magnetic field on axis, T</td><td colspan="2">6</td><td>10.9</td><td>11.1</td></tr>
<tr><td>Maximum field at TF coils, T</td><td colspan="2">12.6</td><td colspan="2">19.3</td></tr>
<tr><td>Plasma current, MA</td><td colspan="2">48</td><td>53.3</td><td>57.2</td></tr>
<tr><td>$\langle n_e \rangle, 10^{20} m^{-3}$</td><td colspan="2">2.1</td><td>1.87</td><td>1.94</td></tr>
<tr><td>$\langle T_i \rangle$, keV</td><td colspan="2">40</td><td colspan="2">57</td></tr>
<tr><td>$\langle \beta \rangle$, %</td><td colspan="2">18</td><td colspan="2">6.7</td></tr>
<tr><td>Elongation, κ</td><td colspan="2">2</td><td colspan="2">1.8</td></tr>
<tr><td>q_ψ</td><td colspan="2">3</td><td colspan="2">2.67</td></tr>
<tr><td>Z_{EFF}</td><td colspan="2">1.6</td><td colspan="2">1.6</td></tr>
<tr><td>τ_E, s</td><td colspan="2">9.0</td><td>15.5</td><td>17.4</td></tr>
<tr><td>α_n, α_T</td><td>0.5; 1</td><td>1;1</td><td>0.95</td><td>0.55</td></tr>
<tr><td>Fusion power, P_{FUS}</td><td>2200</td><td>2600</td><td>2144</td><td>2602</td></tr>
<tr><td>Synchrotron power, MW</td><td>200</td><td>200</td><td>1027</td><td>1249</td></tr>
<tr><td>Bremsstrahlung, MW</td><td>1100</td><td>1100</td><td>652</td><td>858</td></tr>
<tr><td>Transport power, MW</td><td>900</td><td>1300</td><td>456</td><td>497</td></tr>
<tr><td>$Q = P_{FUS}/P_{AUX}$</td><td>7</td><td>$\geq$60</td><td>$\geq$16</td><td>$\geq$17</td></tr>
<tr><td>DD-Neutron power, MW</td><td>30</td><td>36</td><td>38</td><td>40</td></tr>
<tr><td>DT-Neutron power, MW</td><td>80</td><td>94</td><td>109</td><td>116</td></tr>
<tr><td>Neutron wall loading, MW/m^2</td><td>0.05</td><td>0.06</td><td colspan="2">0.1</td></tr>
<tr><td>Direct conversion efficiency, %</td><td colspan="2">90</td><td>80</td><td>—</td></tr>
<tr><td>Thermal conversion efficiency, %</td><td colspan="2">40</td><td colspan="2">44</td></tr>
<tr><td>Net electric power, MWe</td><td>1100</td><td>1400</td><td colspan="2">1000</td></tr>
<tr><td>Net efficiency, %</td><td>50</td><td>53</td><td>43</td><td>36</td></tr>
<tr><td>COE, mills/kWh (1992$)</td><td>–</td><td>–</td><td>81</td><td>89</td></tr>
</table>

[1]Updated (1992) parameters for Apollo-L3.
[2]Thermal conversion only variation of Apollo-L3.

breeding is not required and the neutron wall loading is low enough, $\sim$0.03 MW/m^2, a high temperature helium cooled shielding blanket and plasma facing components can be used to achieve high thermal cycle efficiency. So for cyclotron and bremsstrahlung radiation and neutron components, the thermal cycle efficiency on the level of 40% could be achieved. The power of the electron component will be converted with the same efficiency. The power of the ion component can be converted into electricity with an efficiency of $\eta_I \cong 0.9$. Thus, the total efficiency of fusion energy conversion into electricity may be about 0.5, see table.

For auxiliary equipment including coils and fuel cycle, 10% on-site electrical power consumption was proposed.

3. High-Field, First-Stability, D-^{3}He Reactor Design

An alternative to the modest-field, second stability design presented above is a low-beta, high field design (Apollo) operating in the first-stability regime [5,6]. This design is more strongly based on the existing experimental database developed by a large number of tokamaks and, consequently, has different characteristics. Because of the higher magnetic field, synchrotron radiation is a larger fraction of the total loss. This opens up the possibility of using synchrotron radiation to drive at least part of the plasma current and solid state technology to directly convert synchrotron radiation to electricity. In addition, the divertor power, first wall surface heating, and neutron heating of the shield is converted to electricity using a thermal cycle so that the maximum electrical power is produced.

The major plasma and reactor parameters for this design are given in the table. The on-axis magnetic field is 11 T, the plasma current is 53 MA, beta is 6.7% using a Troyon coefficient of 0.035, and the average ion temperature is 57 keV. Using ITER-89P transport scaling with an H-mode enhancement factor of 4 the confinement product, $n\tau_E$ is 3×10^{21} m^{-3}s and the triple product, $n\tau_E T_i$, is 1.6×10^{23} s-keV/m^3. The plasma current of 53 MA is composed of 23 MA of bootstrap current, 18 MA driven by synchrotron radiation, and 12 MA driven by neutral beam injection using 138 MW of absorbed neutral beam power (197 MWe).

Synchrotron radiation accounts for the dominant power loss because of the high magnetic field and low beta. A beryllium coating is used on the first wall to provide high reflectivity (98%) for synchrotron radiation. About half the synchrotron power exits the plasma chamber through waveguides and is transported to rectifying antennas (rectennas) in a remote chamber for direct conversion to electricity. The waveguides are angled in the toroidal direction to remove toroidal momentum; the reaction on the electrons drives a plasma current. Rectennas potentially allow highly efficient conversion of synchrotron radiation to electricity. They perform well for 2.45 GHz microwave energy transmission; the concept is extrapolated here to very high frequencies (3-30 THz) encountered in synchrotron radiation. High energy conversion efficiency (80%) is predicted but remains to be demonstrated, although the technology lies well within state-of-the art integrated circuit dimensions. The rectenna geometry would be a dielectric slab with the antennas facing the synchrotron radiation, with coolant channels through the dielectric, and with the remaining circuit components on the opposite side of the slab, connected by stripline techniques. Since rectennas for this application require development, a backup option is to convert the synchrotron radiation to electricity using the thermal cycle. A design using this option is also given in the table. Because of the lower efficiency (44%, using an organic coolant) for thermal conversion, this design is somewhat larger although the impact on the cost of electricity is only 10%. This backup option still uses synchrotron current drive.

In order to have a permanent first wall not requiring replacement during the lifetime of the reactor, the neutron wall loading has been constrained to be less than 0.1 MW/m^2. The neutrons come from the $^2H(d,n)^3He$ and $^3H(d,n)^4He$ reactions, where the tritium is produced by the ^{2}H(d,p)^{3}H reaction. Neutrons from the D-T reactions account for about 70% of the total neutron power; about half the tritium diffuses out of the plasma before it reacts with deuterium. This produces tritium in the exhaust stream which is recovered and stored until it decays to ^{3}He. The resulting steady-state inventory of tritium stored in a separate facility is 59 kg. The vulnerable tritium inventory in the reactor is 11 g. Neutron production in the plasma can be reduced by increasing the ^{3}He/D ratio, but this reduces the power density in the plasma. The design operation point (^{3}He/D = 0.6) is deuterium rich in order to maximize the power output and reduce the cost of electricity while meeting the constraint on the neutron wall load. Lower neutron wall loadings could be obtained with larger (more costly) reaction chambers.

In order to keep the concentration of protons and alpha particles down to a tolerable level of 5% for each, the ash particle confinement time must be about equal to the energy confinement time. Whether the plasma will provide this naturally or will require some form of active ash removal is an open question, as discussed in Section 4.

4. Comparison of Issues

For both D-^{3}He tokamak reactors presented here, the major physics issues are achieving adequate energy confinement, removing fusion ash, and generating the plasma current. For the modest-field case, a further issue is experimental verification of the second-stability regime. The main technological challenges are handling the energy deposited by the thermal quench in a disruption and the steady-state power deposited on the divertor. The thermal energy of a D-^{3}He-tokamak plasma is about five times that in a D-T tokamak, but the magnitude of the divertor heat load is similar for both fuels. The concepts presented here also invoke advanced energy conversion technology for part of the fusion power, but these are desirable features rather than necessary ones.

The energy confinement time for the first-stability reactor must be about four times better than that given by the ITER-89P L-mode scaling relation [7]. This value relies on future progress in reducing transport and is justified partly by recent VH-mode results achieved in the DIII-D experiments [8]. The plasma current must also be high, $\sim$50 MA, to increase both energy confinement and β. In order to avoid choking the plasma with fusion ash, the ratio of the ash particle confinement time to bulk energy confinement time must be relatively small, $\tau_{\mathrm{p}}^{\mathrm{ASH}}/\tau_{\mathrm{E}}^{BULK} \cong 1-2$. This regime appears to be present for L-mode operation in TEXTOR [9] and DIII-D [10], but it must be demonstrated in higher confinement operation or an active means of enhancing fusion ash transport will be necessary.

Because of the high plasma current, even with a relatively good match of the bootstrap-current radial profile to the desired plasma-current radial profile, a large current must be driven. For the first-stability case, the synchrotron radiation power drives about one-third of the required current, using a modification to the original Dawson and Kaw concept [11], in which the waveguide that transports the synchrotron radiation to the rectenna chamber serves the function of the original absorbing side of a fish-scale wall. The efficiency of either neutral-beam or fast-wave current drive is low, requiring >100 MW of injected power. More efficient current-drive techniques would not only reduce this input power, but would also allow a larger parameter space operating region.

5. Conclusion

The two reactor regimes studied here, with modest and high toroidal magnetic fields, reveal the principal options for a D-^{3}He tokamak-reactor based on the existing data base in plasma physics. The required reactor parameters do not differ much from those which are adopted for the ITER design. An essential effect of the plasma density and temperature profiles on the reactor reactivity Q has been emphasized. The control over the plasma profile can be one of the main tools for the transition from the low efficiency mode of operation to that with high efficiency.

The opportunity to use the divertor-layer plasma flow and cyclotron radiation direct energy conversion allows one to increase the net efficiency of the reactor by 10-15% in comparison with a D-T reactor.

The engineering problems of the D-^{3}He reactor design with magnetic field $\leq$6 T are similar to those in the ITER design. Therefore the ITER can serve as an experimental base for the next step development, an experimental reactor with D-^{3}He fuel.

Acknowlegements

We would like to acknowledge partial support for this work from the Grainger Foundation, the Bechtel Corporation, Grumman Aerospace Corporation, and the Wisconsin Electric Utilities Research Foundation.

References

1. T.C. Simonen, "ITER Phys./Design Work Session," San Diego, CA, July 10-18, 1991.

2. JET Team, in *Plasma Phys. and Contr. Nucl. Fusion Research 1988*, Proc. 12th Int. Conf., Nice, 1989, Vol. 1, IAEA, Vienna, 159 (1989).

3. M.J. Schaffer, B.J. Leikind, *Nucl. Fusion*, **31**, No. 9, 1750 (1991).

4. S.V. Bazenkov, S.G. Bespoludennov, M.V. Osipenko, V.I. Pistunovich, O.P. Pogutse, "On the Energy Conversion in D-^{3}He Reactor Tokamak," Preprint IAE, Moscow, 1992.

5. G.L. Kulcinski, et al., "Summary of Apollo, A D-^{3}He Tokamak Reactor Design," *Fusion Technol.*, **21**, 2292 (1992).

6. G.A. Emmert, et al., "An Improved First Stability Advanced Fuel Tokamak, Apollo-L3," Proc. 14th Symp. Fusion Engineering, Vol. 2, p. 1106 (IEEE, NY, 1992).

7. N.A. Uckan, et al., "ITER Physics Design Guidelines: 1989," ITER Documentation Series, No. 10 (IAEA, Vienna, 1990).

8. G.L. Jackson, et al., "Regime of Very High Confinement in the Boronized DIII-D Tokamak," *Phys. Rev. Lett.*, **67**, 3098 (1991).

9. D.L. Hillis, et al., "Helium Exhaust and Transport Studies with the ALT-II Pump Limiter in the TEXTOR Tokamak," *Phys. Rev. Lett.*, **65**, 2382 (1990).

10. E.J. Synakowski, et al., "Measurements of Radial Profiles of He^{2+} Transport Coefficients on the TFTR Tokamak," *Phys. Rev. Lett.*, **65**, 3502 (1990).

11. J.M. Dawson and P.K. Kaw, "Current Maintenance in Tokamaks by Use of Synchrotron Radiation," *Phys. Rev. Lett.*, **48**, 1730 (1990).