Could Advanced Fusion Fuels Be Used with Today's Technology?

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Abstract

Could today's technology suffice for engineering advanced-fuel, magnetic-fusion power plants, thus making fusion development primarily a physics problem? Such a path would almost certainly cost far less than the present D-T development program, which is driven by daunting engineering challenges as well as physics questions. Advanced fusion fuels, in contrast to D-T fuel, produce a smaller fraction of the fusion power as neutrons but have lower fusion reactivity, leading to a trade-off between engineering and physics. This paper examines the critical fusion engineering issues and related technologies with an eye to their application in tokamak and alternate-concept D-^{3}He power plants. These issues include plasma power balance, magnets, surface heat flux, input power, fuel source, radiation damage, radioactive waste disposal, and nuclear proliferation.

List of Figures

List of Tables

1 Introduction

The thesis of this paper is that the combination of high engineering power density and deuterium/helium-3 (D-^{3}He) fuel will lower the cost of fusion development because most of the required technology has already been demonstrated. Historically, physics figures of merit have driven fusion energy research and development [1, 2]. The two most important physics measures have been the plasma temperature, T, and the confinement parameter, $n\tau_E$, where n is the plasma density and τ_E is the energy confinement time. On the other hand, a few concepts pursued a path driven by high β (plasma pressure/magnetic field pressure). The fusion power density in the plasma scales as β^2 ($P_{fus}/V \propto \beta^2 B^4$, where P_{fus} is the fusion power, V is the plasma volume, and B is the magnetic field). Concepts with high β thus have high power density and tend to follow a path driven by engineering constraints. These paths are illustrated in Fig. 1.

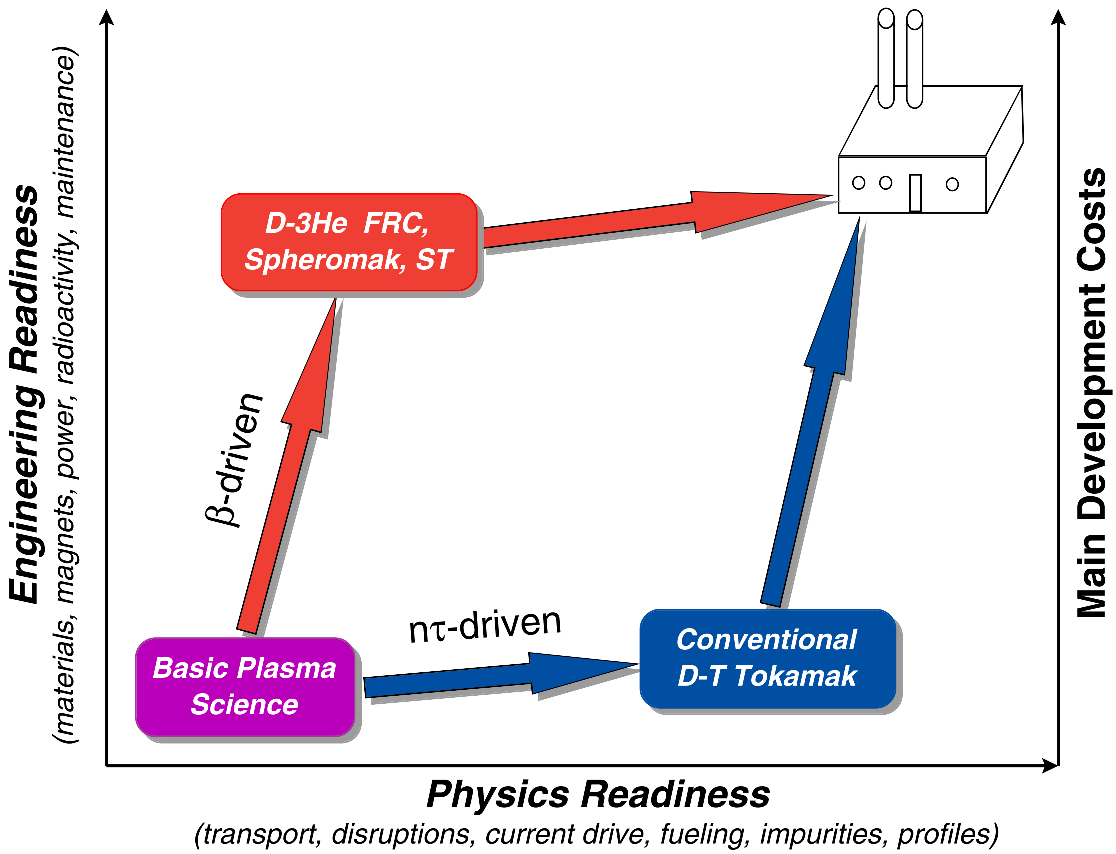

Figure 1: *Schematic view of potential fusion power development paths.*

The physics-driven path leads to concepts possessing good confinement but very low β ($\sim$0.05 for power plants), such as the tokamak. The engineering-driven path leads to concepts with high β (0.2–0.9) and presently uncertain confinement scaling, such as the field-reversed configuration (FRC), spheromak, and spherical torus (ST). Two canonical types of fusion power plants will serve in this paper to illustrate the differences in approach:

1. the "conventional" D-T tokamak, driven by its state of physics readiness and $n\tau_E$, and
2. the D-^{3}He compact toroids (FRC, spheromak, and, loosely, the spherical torus or ST), driven by their state of engineering readiness and β.

This paper examines the key issues for D-^{3}He fusion power plants. Parameters for D-^{3}He power plants will be drawn where possible from the small number of D-^{3}He conceptual design studies. The key issues are plasma power balance, magnets, surface heat fluxes, input power, fuel source, radiation damage, radioactive waste disposal, and nuclear proliferation.

2 Plasma Power Balance

A mixture of deuterium (D) and tritium (T) fuel ignites most easily and has high fusion power density in the plasma, while deuterium and helium-3 (^{3}He) ignite less readily. The relevant reactions are given in Table 1. At constant β and B, the fusion reaction rates averaged over a Maxwellian distribution result in fusion power densities in the plasma that are shown in Fig. 2. Historically, the relatively low fusion power density in the plasma for all fuels compared to D-T led to the dominance of D-T in fusion energy research. At present, when power-plant engineering, safety, and environmental issues are beginning to be faced, the equally important issues of damage and activation due to neutrons have become prominent. The D-T, D-D, and D-^{3}He neutron production is shown in Fig. 3 and clearly favors D-^{3}He fuel.

Table 1: *Main Fusion Reactions in D-T and D-^{3}He Power Plants.*

$$
\begin{array}{rcll}
\mathrm{D+T} & \longrightarrow & \mathrm{n}\ (14.07\ \mathrm{MeV}) + {}^4\mathrm{He}\ (3.52\ \mathrm{MeV}) & \\
\mathrm{D+{}^3He} & \longrightarrow & \mathrm{p}\ (14.68\ \mathrm{MeV}) + {}^4\mathrm{He}\ (3.67\ \mathrm{MeV}) & \\
\mathrm{D+D} & \longrightarrow & \mathrm{p}\ (\ 3.02\ \mathrm{MeV}) + \mathrm{T}\ (1.01\ \mathrm{MeV}) & (50\%) \\
\mathrm{D+D} & \longrightarrow & \mathrm{n}\ (\ 2.45\ \mathrm{MeV}) + {}^3\mathrm{He}\ (0.82\ \mathrm{MeV}) & (50\%)
\end{array}
$$

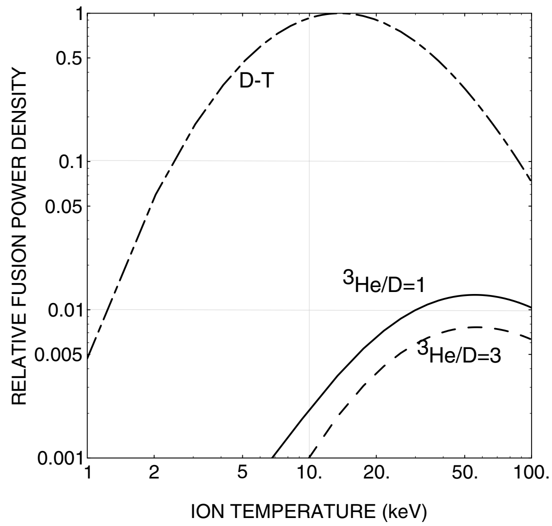

Figure 2: *Relative fusion power densities in the plasma for the main candidate fusion fuels.*

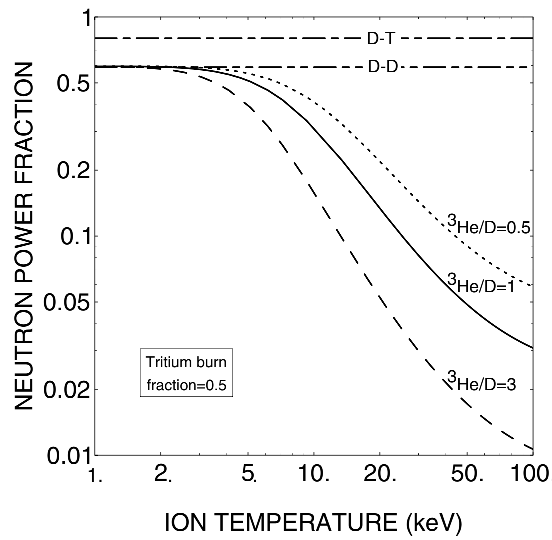

Figure 3: *Neutron production rates for the main candidate fusion fuels.*

Neither of these figures in itself, however, adequately describes their net effect on the engineering power density (kWe/tonne), sometimes called the mass power density, of a power plant. Three factors increase the relative fusion power density for a D-^{3}He power plant, and their cumulative effect is shown for a typical case in Fig. 4:

- **Increased magnetic field:** The lower neutron wall load allows increasing the B field or expanding the plasma into a region of increased B field to increase the fusion power density ($\propto B^4$).

- **Reduced shield thickness:** Because D-^{3}He fuel requires a shield for the magnets of only about one-half that of D-T fuel, the shield mass is significantly reduced.

- **Direct energy conversion:** The large charged-particle fusion power fraction in a D-^{3}He fusion core can be directly converted to electricity at high efficiency in some configurations.

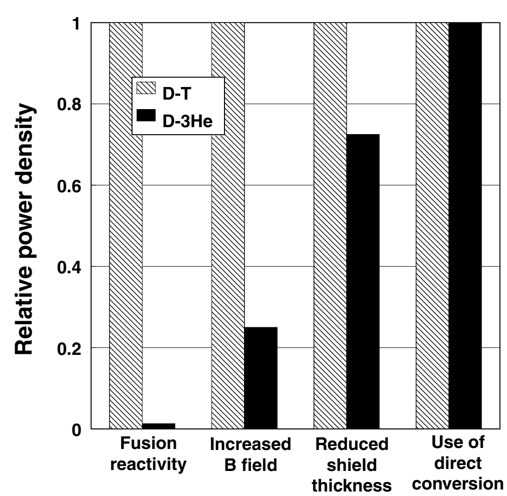

Figure 4: *Relative power density for power plants using D-T or D-^{3}He fuel when direct conversion to electricity and the optimized magnetic field and shield thickness are included (cumulatively) in the analysis.*

3 Magnets

Typical magnetic fields at the coils in conceptual D-^{3}He power plant designs are ≤ 10 T for advanced concepts [3, 4] and ≤ 16 T for tokamaks [6, 8, 9]; some examples are given in Table 2. Various superconducting magnets have been constructed and have demonstrated the required performance [10, 11, 12, 13]; some are listed in Table 3. The required magnetic fields at the coils in D-T power plants are generally comparable. The main point demonstrated by Table 3 is that the D-^{3}He power plants, unless based upon the tokamak configuraton, require only demonstrated magnet technology. In general, the high β of alternate concepts leads to magnetic field requirements of $B \leq 8$ T and dimensions of at most a few meters, which clearly falls within the capabilities of the existing magnets of Table 3. In particular, even for complicated geometries such as the MFTF-B yin-yang and LHD coils, fields of 6–7 T at dimensions of $\lesssim 4$ m have been demonstrated. Tokamaks burning D-^{3}He fuel, like their D-T counterparts, are predicted to require 14–20 T at the necessary scale, which has not yet been demonstrated.

Table 2: *Magnetic Fields at the Coils in Conceptual D-^{3}He Power Plant Designs.*

Name	Geometry	B Field at Superconductor (T)	Coil Radius (m)
Artemis [3]	FRC	6.7	3.5
Ra [4]	Tandem mirror		
Central cell		6.5	1.25
Choke coil		16 (+8 T from Cu solenoid insert)	0.25
Apollo [6]	Tokamak	20	3.4×6.9
ARIES-III [8, 9]	Tokamak	14	4×7.2

Table 3: *Some High-Field Magnets That Have Been Operated or Proposed.*

Device	Geometry	B Field at Superconductor (T)	Largest Inner Dimension (m)	Conductor
MFTF-B [10] end plug	yin-yang	5.8	2.5	NbTi
LCT [11]	D-shape	8.1	2.5×3.5	NbTi or Nb_3Sn
LHD [12]	continuous-coil stellarator	6.9	3.9	NbTi
ITER [13] solenoid	cylinder	13	1.6	Nb_3Sn

4 Surface Heat Fluxes

At constant plasma volume and fusion power density, a D-^{3}He plasma produces nearly five times the charged-particle power of a D-T plasma. In practice, the plasma volume and fusion power density will vary for fusion chambers burning the two fuels. Assuming rough similarity, however, has caused some researchers to argue that the higher D-^{3}He charged-particle power will lead to five times higher surface heat fluxes than D-T fusion burn chambers will experience, and that these heat fluxes will be unmanageable [14]. With proper engineering, however, the heat fluxes in D-^{3}He power plants would *not* be a significant problem for several reasons:

- Engineering design constraints on the first-wall surface heat flux relax considerably when the additional constraint added by the complexity of a tritium-breeding blanket is removed, so that much higher heat fluxes can be handled.
- Allowing for reasonable progress in toroidal plasma energy confinement, part of the fusion energy could be transported by waveguides as synchrotron radiation out of the fusion burn chamber.
- The charged-particle power in configurations with linear topology for the external magnetic field will be transported out the ends of the fusion burn chamber, so that only bremsstrahlung and synchrotron radiation will contribute to surface heat fluxes.

Conceptual design heat-flux parameters for several D-T and D-^{3}He fusion power plant designs are shown in Fig. 5. The heat flux values are separated into three qualitatively different classes: (1) D-^{3}He alternate concepts, (2) D-T and D-^{3}He tokamak divertors, and (3) D-T and D-^{3}He tokamak first walls. The designs used for this comparison are the Artemis D-^{3}He field-reversed configuration [3], the SAFFIRE field-reversed mirror [16], the General Atomics spherical torus (GA ST) [15], the Ra thermal barrier tandem mirror [4], the ITER tokamak engineering test reactor [17], the ARIES D-T tokamak series [18, 19, 20, 21], the ARIES-III D-^{3}He tokamak [8, 9], the Apollo D-^{3}He tokamak [5, 6, 7], and the TITAN D-T reversed-field pinch [22, 23]. Except for the divertor of the General Atomics spherical torus design (GA ST) [15], the D-^{3}He alternate-concept designs have similar surface heat fluxes to D-T and D-^{3}He tokamak first walls and much lower heat fluxes than tokamak divertors.

The surface heat flux in a fusion chamber could be reduced from the values in Fig. 5 by using waveguides to carry synchrotron radiation to a separate chamber. This would be accomplished by increasing the $n\tau_E$ value, so that charged particle losses would be reduced, and the 'excess' fusion power could be produced as synchrotron radiation by increasing the magnetic field or the plasma temperature. This would require progress in plasma confinement to continue at approximately the pace it has over the past two decades.

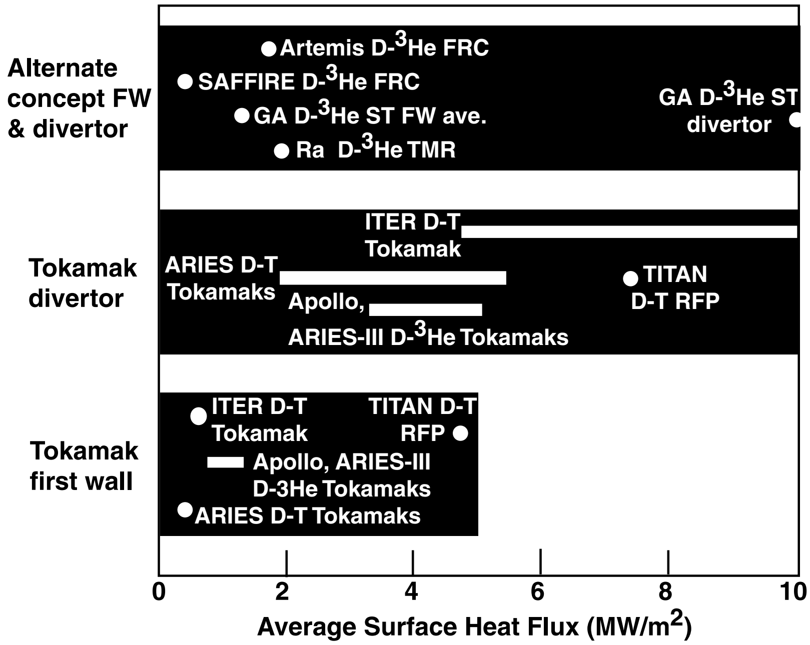

Figure 5: *Average surface heat fluxes for some fusion power plant conceptual designs.*

In linear configurations, the axial plasma flow mitigates problems with the surface heat flux to the first wall, because almost all of the charged-particle power would leave the fusion chamber along magnetic flux tubes of linear topology. Expanding the magnetic flux tube in a separate, non-fusion chamber at the ends of the device would then allow the surface heat flux to be reduced to an even lower value, if desired. Directly converting part of the charged-particle energy to electricity would further reduce the surface heat flux, as the converted fraction of the energy would go directly to the electric circuit and not appear as heat.

5 Input Power

In general, input power technologies are well in hand and modular. The requirements for presently operating experiments typically are met by sets of radio-frequency (RF) or neutral-beam (NB) injectors where each unit has a power level of approximately 1–10 MW [24, 25, 26, 27]. The input-power systems for the four most important magnetic fusion options (neutral beams, ion-cyclotron range-of-frequencies, lower-hybrid, and electron-cyclotron range-of-frequencies) are shown in Table 4. Three categories are shown: demonstrated systems, fusion test reactors, and conceptual power plants. Due to the modularity of the input-power units, reaching the required power levels of $P_{in} \lesssim 100$ MW becomes a question of economics and available space, rather than a technology issue as such.

Table 4: *Some Input-Power Sources for Operating or Future Devices.*

Neutral Beams	**Status**	**Power**	**Energy**
JT-60U tokamak [24]	operated	10 MW	500 keV
TFTR tokamak [25]	operated	40 MW	115 keV
ITER tokamak [17]	proposed	50 MW	1 MeV
ARIES-III tokamak power plant [9]	conceptual	170 MW	3–6 MeV
Artemis FRC power plant [3]	conceptual		
Startup		100 MW	1.4 MeV
Steady-state		5 MW	1 MeV
ICRF	**Status**	**Power**	**Frequency**
JET tokamak [26]	operated	17 MW	55 MHz
ITER tokamak [17]	proposed	50 MW	90 MHz
Apollo tokamak power plant [6]	conceptual	57 MW	110 MHz
LH	**Status**	**Power**	**Frequency**
JET tokamak [26]	operated	7 MW	3.7 GHz
ITER tokamak [17]	proposed	50 MW	5 GHz
ECRF	**Status**	**Power**	**Frequency**
Heliotron-E stellarator [27]	operated	0.4 MW	106 GHz
ITER tokamak [17]	proposed	6 MW	170 GHz
Ra tandem mirror power plant [4]	conceptual		
Plug region		7 MW	195 GHz
Barrier region		17 MW	14 GHz

The $\sim$90 MHz ICRF required for ITER and Artemis and the 110 MHz required for Apollo have essentially been demonstrated, because the frequency range is just below that of lower-hybrid RF technology. The 1-MeV neutral beams needed for ITER and Artemis necessitate longer acceleration stages but otherwise are straightforward extrapolations of the negative-ion LHD technology. The key input-power technology development necessary for complete demonstration of the capabilities required for D-^{3}He fusion power plants is the extension of present pulse times to steady-state operation. This is an active area of investigation and ITER, for example, requires 1000 s pulses, which is steady state for practical purposes.

6 Fuel Source

The most frequently asked questions related to burning D-^{3}He fuel in fusion power plants are whether sufficient ^{3}He resources exist and whether they could be obtained economically. In principle, the first of these problems has been solved by the recognition that the Moon's surface contains a large source of ^{3}He [28, 29]. The economic projections are intrinsically uncertain for a large-scale project of the sort required for lunar ^{3}He mining, but arguments have been put forth that the cost of lunar ^{3}He should be in the range of $400–$1,000/g [30]. This would add ~10 mill/kWh to the cost of electricity, which should be more than offset by the lack of a tritium-breeding blanket and the reduced replacement cost for the first wall, blanket, and part of the shield.

The requirements for lunar ^{3}He mining are generally straightforward extrapolations of terrestrial mining technologies: bucket wheel excavators, conveyor belts, and process (solar) heat. The key difficulty will likely be maintaining the integrity of seals against the finely pulverized lunar regolith. A conceptual ^{3}He lunar miner design appears in Fig. 6 [29].

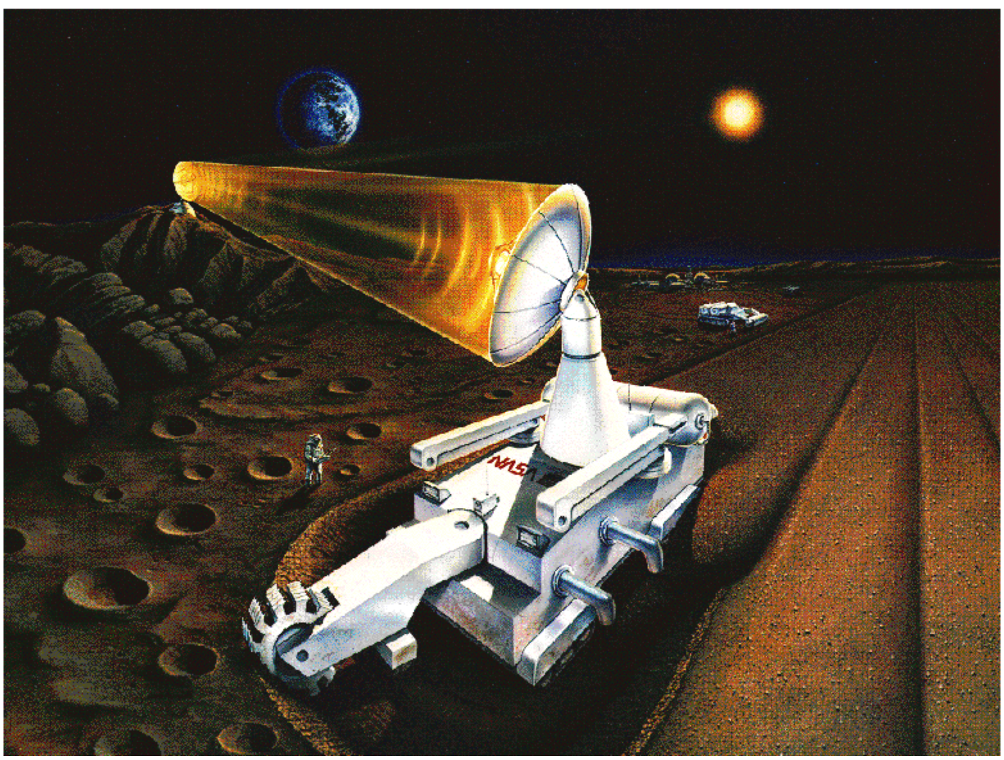

Figure 6: *Conceptual lunar ^{3}He miner design.*

The technology with the largest leverage on the cost of lunar ^{3}He will be terrestrial launch vehicles. The Apollo program demonstrated many of the required technologies for reaching the Moon, and the development needed is to improve the Saturn V rocket technology of the Apollo missions. A potential trade-off with reduction of Earth-launch cost is the use of in-situ materials to construct lunar ^{3}He miners, because present lunar ^{3}He cost estimates assume that all of the lunar miners and related systems will be built on Earth and then shipped to the Moon. This trade-off has not yet been examined in depth.

7 Radiation Damage

The relatively low radiation damage rates in D-^{3}He fusion chambers would allow permanent first walls to be designed with unmodified austenitic or ferritic steels [5]. In contrast, a D-T power plant would require the first wall, blanket, and part of the radiation shielding to be changed out on time scales of every few years. This stems from both the intrinsically softer neutron energy spectrum and lower strength of the D-^{3}He neutron source. The D-^{3}He and D-T operating regimes and the end-of-life damage for several conceptual design studies are illustrated in Fig. 7. Note that all of the D-^{3}He designs lie in the area where material data is available, and they can utilize permanent first-wall materials which have already been tested to end-of-life exposures. The D-T designs require 10–30 replacements of the first wall, blanket, and part of the shield over a reactor lifetime.

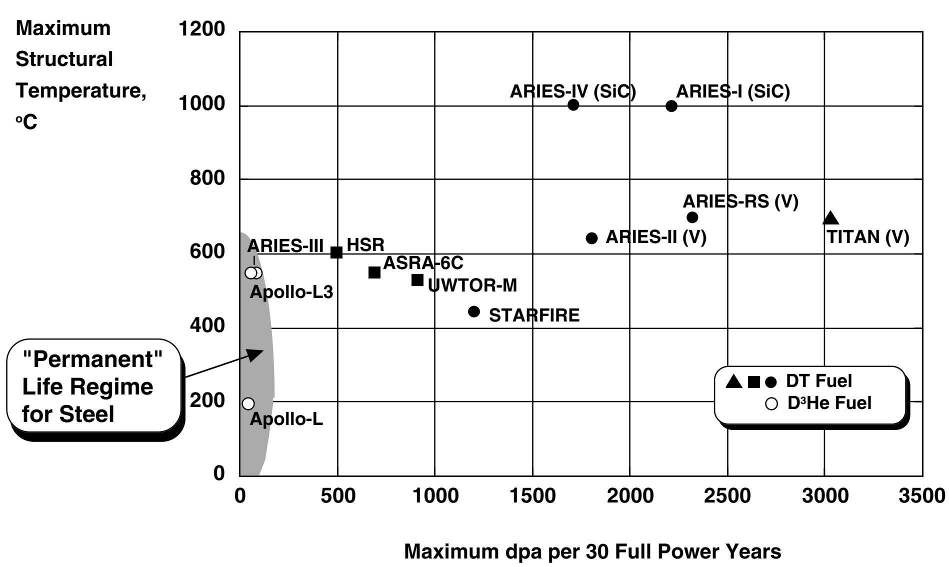

Figure 7: *Radiation damage levels and operating temperature regimes for conceptual D-^{3}He and D-T power plants.*

8 Radioactive Waste Disposal

Power plants burning D-^{3}He fuel produce relatively few neutrons, as was shown in Fig. 3. The resulting radioactive waste could be hospital-level (Class A) using low-activation steels, such as tenelon [31], even allowing the material to remain the power plant lifetime of 30 full-power years. The best that a D-T power plant could do with tenelon appears to be Class C, assuming that each module would be changed out every 5 full-power years. With HT-9 steel modified for low activation, which would require very little development, D-^{3}He waste would still be Class C, while D-T would require deep geologic burial. These disposal regimes are illustrated in Fig. 8. Some tritium would be generated in the plasma due to D-D reactions, and part of it would diffuse into the plasma exhaust. This tritium would require disposal by re-injection into the plasma, storage until it decays into ^{3}He, or sale to D-T power plants.

The differences between Class A and Class C apply only to radioactive waste in the U.S., while European guidelines dictate that *all* radioactive waste be treated as high level. The distinction, nevertheless, illustrates the qualitative advantage of D-^{3}He fuel over D-T fuel in an arena where the target will remain a moving one for many years. Class A waste requires minimal encasement and burial site monitoring for 100 years. Class C waste requires concrete encasement, burial at least 5 m underground, and site monitoring for 500 years.

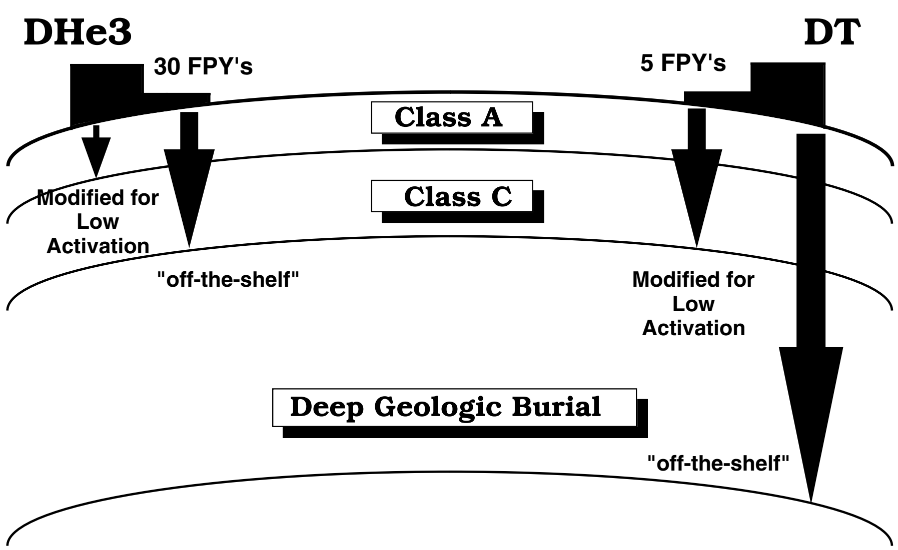

Figure 8: *Radiation waste levels and disposal requirements for conceptual D-^{3}He and D-T power plants.*

9 Nuclear Proliferation

An important distinction between D-^{3}He and D-T power plants is that D-^{3}He power plants could not produce fissile fuel and contribute to nuclear proliferation. This statement is based on the fact that the thicknesses of radiation shields for the superconducting magnets in D-^{3}He fusion cores are typically a factor of two less than required for D-T fuel. This difference stems from the 10–30 times lower neutron particle production of D-^{3}He fuel compared to D-T fuel. In such reactors, the use of D-T fuel instead of D-^{3}He fuel would both overheat and damage the superconducting magnets to unacceptable levels. Inserting a fissile-fuel breeding blanket with increased shielding into a D-^{3}He fusion chamber would be very time consuming at best, because the fusion chamber would not be designed for routine changeout and would require significant, possibly infeasible, plasma and heating parameter alteration. As discussed in the previous section, some tritium would be produced, but it is not considered a serious concern for nuclear proliferation in comparison to fissile materials. Although the contention that a D-^{3}He power plant would avoid nuclear proliferation hazards remains to be quantified, the neutron production difference from D-T is sufficiently large that the statement can be made with considerable confidence.

10 Conclusions

Fusion power plants burning D-^{3}He fuel and based on advanced physics concepts appear possible to build with essentially today's technology. The key candidate configurations for D-^{3}He fuel (field-reversed configurations, spheromaks, and spherical tori) share the attribute of high β. For the FRC and spheromak, linear geometry for the external magnetic field also facilitates direct conversion of charged-particle energy to electricity. Conventional tokamak power plants burning D-^{3}He fuel would be more difficult to build, and they face obstacles similar to those of D-T power plant designs: e.g., disruption handling, steady-state divertor heat flux, and current-drive power.

The engineering readiness of D-^{3}He in comparison to D-T fusion technology derives mainly from (1) the reduced neutron production, which greatly eases materials constraints; (2) operating at high beta, because surface heat fluxes remain manageable and magnetic fields fall or stay constant; (3) requiring standard input-power technologies, which are essentially developed; (4) eliminating the potential for nuclear proliferation. The key remaining engineering challenges lie in demonstrating the economic acquisition of lunar ^{3}He, steady-state input power, and modest modifications to structural material alloys.

Acknowledgments

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