# Comparison of Critical Requirements and Prospects for Stellarator and Tandem Mirror Fusion Power


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# CHAPTER 1

## Background for the Present Study

In a previous report[1], the "State of Tandem Mirror Physics" was examined up to the Fall of 1992. That report reviewed the US tandem mirror program up through its cancellation in 1986 as well as research efforts in Japan and the USSR. Only Japan and the Russian Federation (RF) currently maintain active programs. It was concluded that even though the tandem mirror has many technology advantages over the currently favored magnetic confinement approach, the tokamak, several critical plasma physics issues need to be solved and demonstrated experimentally before one could build a tandem mirror ignition experiment.

Another magnetic confinement approach, the steady state stellarator, has also competed with the tokamak for scarce financial resources on the physics level. It is also the first alternative to the tokamak in Europe. A logical question then arises as to the relative technological advantages presented by both approaches (tandem mirrors and stellarators). The purpose of this report is to examine the current state of technology required for both power reactor concepts and to quantify the level of physics extrapolation required for both approaches to get to a power reactor.

Another objective of this report is to make a more in-depth examination of the experimental facilities that will be needed for both concepts to overcome the current level of physics extrapolations necessary to operate a commercial power plant. The emphasis is on a development strategy that is on the one hand, aggressive, and on the other hand, realistic. No detailed cost requirements have been given for this development schedule, but since the tandem mirror can be built at smaller fusion power levels, it is possible that the R&D amount to get to a proof of principle would be less than that currently required by the world tokamak program. The international fusion community is now considering the construction of a 3000 MW experimental facility (ITER)[2]. As presently designed, the capital costs of ITER may approach 8 billion dollars[3]. Added to the large capital costs is a several billion dollar development program that would be needed in the US, EC, RF, and Japan before the fusion community is ready to build ITER. This brings the total investment in ITER to potentially 10 billion dollars or more.

Finally, an attempt is made to compare the technology requirements for both magnetic configurations. Such a comparison is difficult because not all technologies have an equal weight in determining the attractiveness of a concept. For example, it is possible that one plasma confinement approach could be much easier with respect to 9 out of 10 key technologies but the 10th requirement could nullify the whole feasibility of the approach if it is not solved. Therefore, caution must be taken not to compare just the number of problems to be solved, but the degree of difficulty needed to be overcome and the long range potential of a successful program along those lines.

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# CHAPTER 2

# Current Status of Relevant Technologies for Tandem Mirrors and Stellarators

## A. Magnets

Advances in superconductor development and magnet technology have continued without signs of saturation. This is based on the demand for large projects such as the accelerators SSC and LHC, high field laboratory magnets, the ITER R&D activities and the stellarators W 7-X and LHD [1]. The fastest growing field is fundamental material research on the new high temperature superconductors, even approaching basic magnet technology development at the present time.

### Low Field Superconductors

The only material in present use with a great deal of success is the NbTi alloy. Some improvements in recent years can be reported here. First of all for the basic multifilamentary wires there are improvements in the current density, and thin filament wires are now available for alternating current (AC) applications. Conductor tailoring for specific needs has been demonstrated in a broad variety for different projects. We refer specifically to fusion magnets where different cables in conduit conductors have been proposed. Multistage cables in a rectangular steel conduit have been fabricated and characterized for NET and APOLLO [2], as well as cables in an aluminum alloy conduit which is soft during forming and winding, but is strengthened during heating for epoxy curing, have been developed for W 7-X. NbTi conductors will continue to be the material of choice in near term applications for magnets up to ~9 T at 4.2 K and 10-11 T at 1.8 K. A scaled W 7-X conductor cooled to 1.8 K can be used for a Helias reactor at 10.7 T

### High Field Superconductors

While stellarator reactors can be designed with NbTi conductors, tokamaks and tandem mirrors require fields on the conductors of 13 T and much more. The most advanced high field superconducting material thus far is $Nb_3Sn$. Current densities in the past 20 years have been increased by almost a factor of 4, presently approaching $10^9$ $Am^{-2}$ at 14 T in optimized samples, and $6 \infty 10^8$ $Am^{-2}$ at 14 T in long industrial lengths of 1-3 km. There are several competing fabrication methods currently available in industry such as the bronze route which achieves small AC losses but not a maximal current density, the Jelly Roll process which gives high AC losses but good current density and is the cheapest, and finally the internal tin process which gives high current density, albeit in shorter lengths available at this time. In all cases the optimization in the 12-14 T range is different from that for 16-20 T range (1.8 K), where ternary alloying with Ta or Ti is required for good current density capability (Fig. 2.A.1).

After many years without a breakthrough, there are now available $Nb_3Al$ wires with current densities comparable to $Nb_3Sn$ in the 12-14 T range (see Fig. 2.A.1). Their major advantage is the superior mechanical properties with respect to tensile strength, but even more important, the much smaller strain related current density degradation of only 10-20%.

Considerable effort has been spent, successfully, in recent years to develop high current (50 kA) cable in conduit conductors using $Nb_3Sn$ multistage cables with reduced AC losses and high mechanical strength. Small AC losses can be achieved by coating the individual wires, while the strength is provided by making thick conduits from austenitic steels, or even Incoloy or Ti which are a better match with respect to thermal contraction and thus, reduce strain degradation. Long cable runs of 300-1000 m are

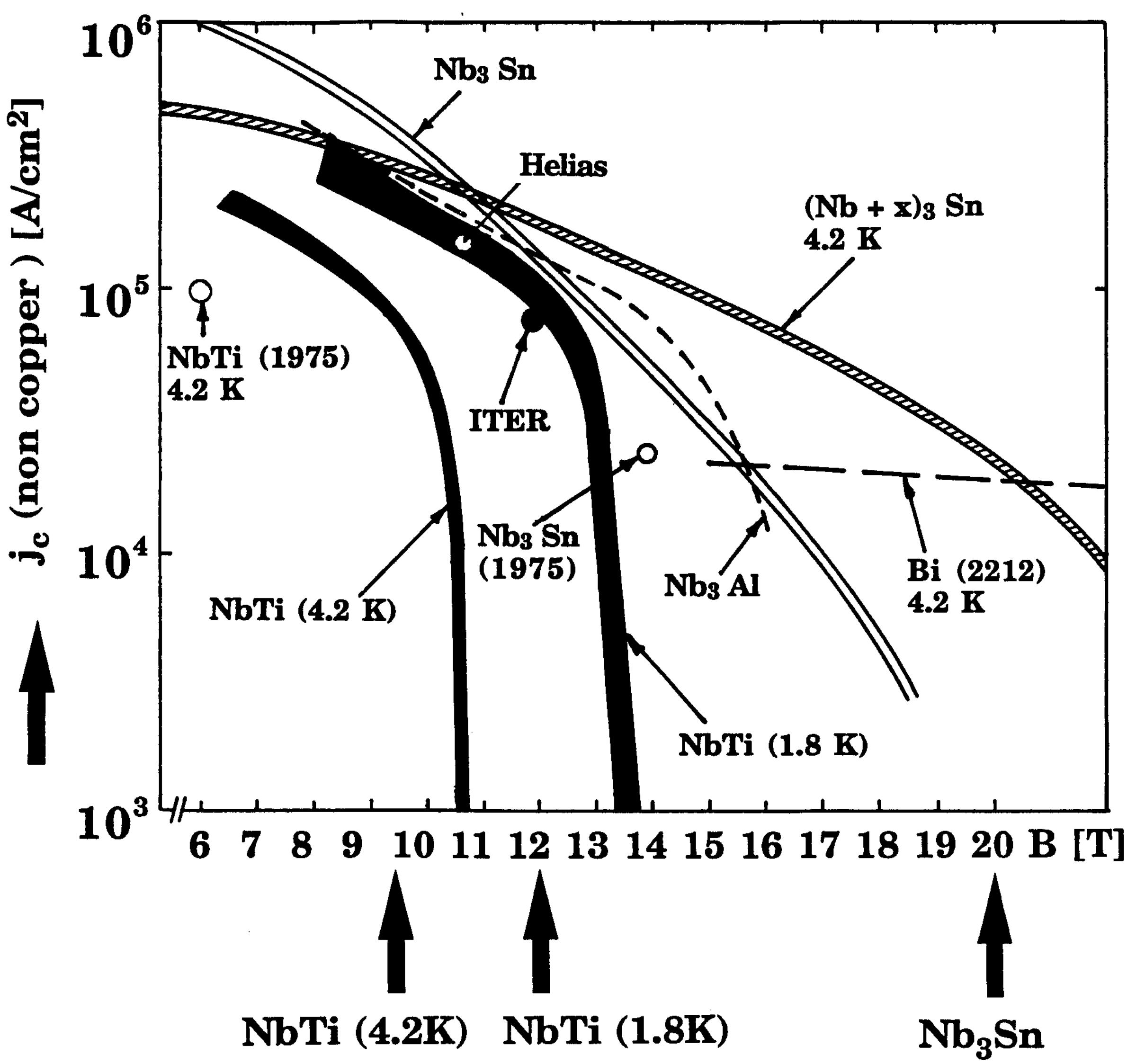


Fig. 2.A.1. Attainable current densities in actual conductors as of 1990.

presently fabricated by several companies worldwide within the ITER R&D program [3]. Fabrication of the ITER model coil will provide a strong impact on the industrial capacity of $Nb_3Sn$ production in the next several years, since the required quantity of ~25 tonnes of $Nb_3Sn$ strand material represents 4-5 times the present annual fabrication capacity worldwide.

## High Temperature Superconductors (HTSC)

Progress in basic research on HTSC since their discovery in 1986 has been dramatic, and fabrication techniques have been substantially improved [4]. Nevertheless, wire technology is still in a very early stage. Of the many HTSC materials, only $Bi_aSr_bCa_cCu_dO_8$, with a = b = c = 2 and d = 3, or a = b = d = 2 and c = 1 could be developed in thin tape-like wires. Their critical current density at 4.2 K and up to 20 K is excellent ($>10^9$ $Am^{-2}$), even at very high magnetic fields (~25 T); however, at 77 K it remains rather low, even at zero field. Present expectations are that Bi (2223) and in particular Bi (2212) can be made into usable wires for the innermost parts of high field magnets (>20 T) operated at 4.2 K, or for medium field magnets at about 20 K. The strong anisotropy in the critical current density, however, is critical.

The other attractive material thus far is $YBa_2Cu_3O_7$, which, however, could not be fabricated in wire or tape form with comparable current densities in spite of the very good results obtained in thin film preparations. These preparations are suited for electronic applications. By the use of sophisticated melt texturing processes, small bulk pellets and rods can be prepared which achieve attractive current densities at 77 K and low field. They have applications in current leads and magnetic bearings.

Thus far, there appear to be some fundamental problems which prevent the achievement of better properties at 77 K. For Bi (2223) this has to do with flux flow behavior starting at very low magnetic fields, while for the Y (123) it is the weak link behavior at the grain boundaries of the bulk polycristalline material.

In any case, because of their early stage of development, it is not expected that HTSC will play a role in magnets for fusion applications in the near term or medium term periods.

## Magnet Technology

Most of the tandem mirror magnets are based on the technology of coupled solenoids with sizes already demonstrated in MFTF-B and in huge particle detectors. The ultra-high field choke coil, however, will present a challenge. Ultra high field hybrid coils, using superconducting coils with normal coil inserts, have been developed and some have been constructed for use in the increasing demands within high field magnet laboratories. Superconducting solenoids with 20-21 T fields in a 5 cm bore, operated at 1.8 K are in operation today, while superconducting outer solenoids with 13-15 T in a bore of about 30 cm to be used for a 40 T hybrid magnet are in progress. Even more sophisticated magnets are currently under design.

Octupole and other end cell magnets in tandem mirrors are similar to the modular stellarator magnets in progress at the present time. Winding such coils with copper conductors has already been demonstrated in W 7-AS, and their superconducting versions will be demonstrated in W 7-X. All large scale winding technologies will benefit from the ITER R&D currently in progress which will ultimately produce large model coils.

**Table 2.B.1 Major Changes in Material's Data For Fusion Reactors, 1986 vs 1993**

| | *Irradiation Data Base* | | *Unirradiated Data Base* | |
|---|---|---|---|---|
| | **Major Positive Results** | **Major Negative Results** | **Major Positive Results** | **Major Negative Results** |
| **Structural Alloy** | | | | |
| Austenitic Steel Alloys | Low Act. High Mn Steels Show Same Swelling Resistance as Normal 316 SS | •Low Ductility After Irradiation 200-300 °C •Irrad. Induced Aqueous Stress Cracking >250°C | Mechanical Property Data On Low Activation Austenitics Shows Little Change | Stress Corrosion Cracking in High Temperature Water (> 200°C) |
| Ferritic Steel Alloys | Low Act HT-9 Shows Lower DBTT and Same Swelling as HT-9 | | Data on Low Act Ferritic Show Improved Properties | |
| V Alloys | V-5Cr-5Ti Showed Small DBTT Shift at Low He and H Content | Embrittlement at Low He Contents-V15Cr5Ti | Low DBTT in V-5Cr-5Ti Tested in Vacuum | Fabricated Cost Higher Than 400 $/kg |
| SiC | | Severe Embrittlement at Moderate T | Large Number of Non Fusion Applications | Fabricated Cost Higher Than 400 $/kg |
| Ti Alloys | | | Major Interest For TPX | |
| General | *There May Be No Basis For Low Level Waste , But General Public Acceptance Will Dictate Low As Possible Radioactivity For Both Short And Long Times* | | | |
| **Breeder** | Major In-Reactor Tests-Li2ZrO3 Low Swelling | LiO2 Shows High Swelling Rate | General Fabrication Techniques Developed for Ceramics | |
| **Neutron Multiplier** | | Swelling Due to High He Generation in Be | | Fabricated Be Costs Greater Than 600 $/kg |
| **Insulators** | | | Initial Tests of AlN, TiN on V | |
| (shaded) | Means That No Major Changes Have Occurred Over the Past 7 Years | | | |

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## B. Structural Materials

The major advances in structural materials since 1986 have occurred mainly in the area of increased irradiation data on low activation (LA) stainless steels, V alloys, and solid breeders. There has also been more interest in V, Ti alloys, and SiC because of concern for the level of long term radioactivity. The situation is briefly summarized in Table 2.B.1 and discussed below.

### Austenitic Steels

Because of the long lived radioactivity associated with Mo, Nb, and Ni, a major thrust has been to replace them with more benign alloying elements such as Mn and V or W. Table 2.B.2 summarizes the typical alloy substitutions which have been recently made in the high performance austenitic alloys PCA and HT-9, to reach a LA alloy status.

The resulting radioactivity in these two alloys is displayed in Fig. 2.B.1[1]. It is clear that substitutions such as those in Table 2.B.2 can reduce the long lived

**Table 2.B.2**
**Elemental Composition of Normal and Reduced Activation Steels**

| | Concentration in Wt. % | | | |
|---|---|---|---|---|
| **Element** | **PCA** | **Tenelon** | **HT-9** | **MHT-9** |
| **B** | 0.005 | 0.001 | 0.01 | 0.001 |
| **C** | 0.005 | 0.15 | 0.2 | 0.15 |
| **N** | 0.01 | 0.005 | 0.05 | 0.001 |
| **O** | | 0.007 | 0.01 | 0.007 |
| **Al** | 0.03 | 0.008 | 0.01 | 0.008 |
| **Si** | 0.5 | 0.2 | 0.35 | 0.2 |
| **P** | 0.01 | 0.13 | 0.02 | 0.013 |
| **S** | 0.005 | 0.004 | 0.02 | 0.004 |
| **Ti** | 0.3 | 0.003 | 0.09 | 0.1 |
| **V** | 0.1 | 0.002 | 0.3 | 0.3 |
| **Cr** | 14.0 | 15.0 | 12.0 | 11.0 |
| **Mn** | 2.0 | 15.0 | 0.55 | 0.53 |
| **Fe** | 64.88 | 69.4 | 85.0 | 85.2 |
| **Co** | 0.03 | 0.005 | 0.02 | 0.005 |
| **Ni** | 16.0 | 0.006 | 0.5 | 0.006 |
| **Cu** | 0.02 | 0.003 | 0.09 | 0.003 |
| **Zr** | 0.005 | 0.001 | 0.001 | 0.001 |
| **Nb** | 0.03 | 0.00011 | 0.0011 | 0.00011 |
| **Mo** | 2.0 | 0.00027 | 1.0 | 0.00027 |
| **Ag** | 0.0001 | 0.00009 | 0.0001 | 0.00009 |
| **Sn** | 0.005 | 0.003 | 0.003 | 0.003 |
| **Ta** | 0.01 | 0.0004 | 0.001 | 0.0004 |
| **W** | 0.05 | 0.01 | 0.5 | 2.50 |
| **Pb** | 0.001 | 0.0005 | 0.001 | 0.0005 |
| **Bi** | 0.001 | 0.0002 | 0.001 | 0.0002 |

activation by orders of magnitude (see Fig. 2.B.2). However, there is a penalty to be paid with these low activation alloys and that is the short term afterheat is aggravated. Figure 2.B.2 also shows that the short term afterheat is increased by a factor of 3 in Tenelon compared to PCA [2] because of the higher Mn content.

The development of LA austenitics has progressed to the point where there is considerable mechanical property data available from both unirradiated and irradiated alloys. The main conclusion from the work in these alloys is that they perform

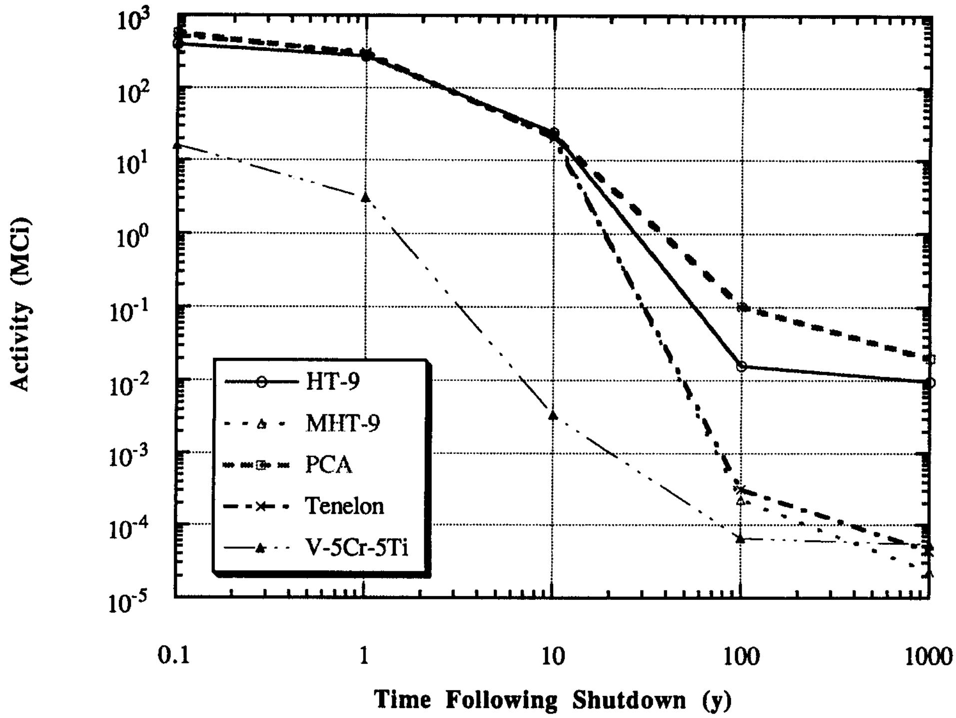


Fig. 2.B.1. The low activation austenitic and ferritic steels can compete with V alloys in the 200-300 year time frame [1].

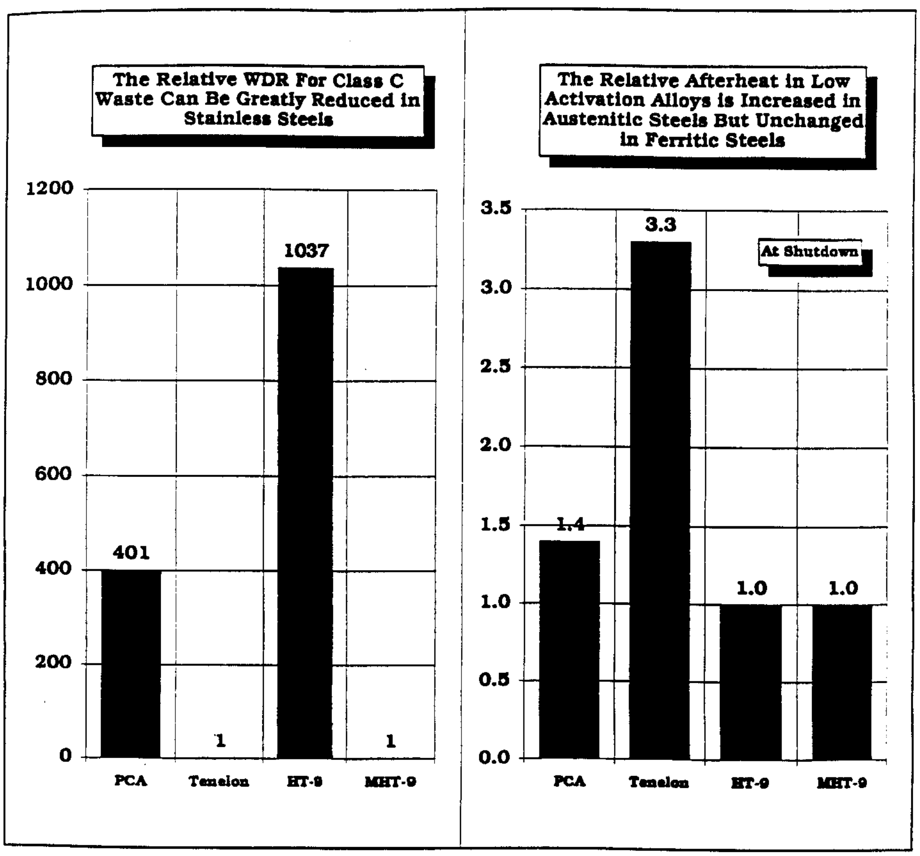


Fig. 2.B.2. The use of high Mn austenitic steels can greatly reduce the Waste Disposal Rating (WDR) but the afterheat is much larger [2].

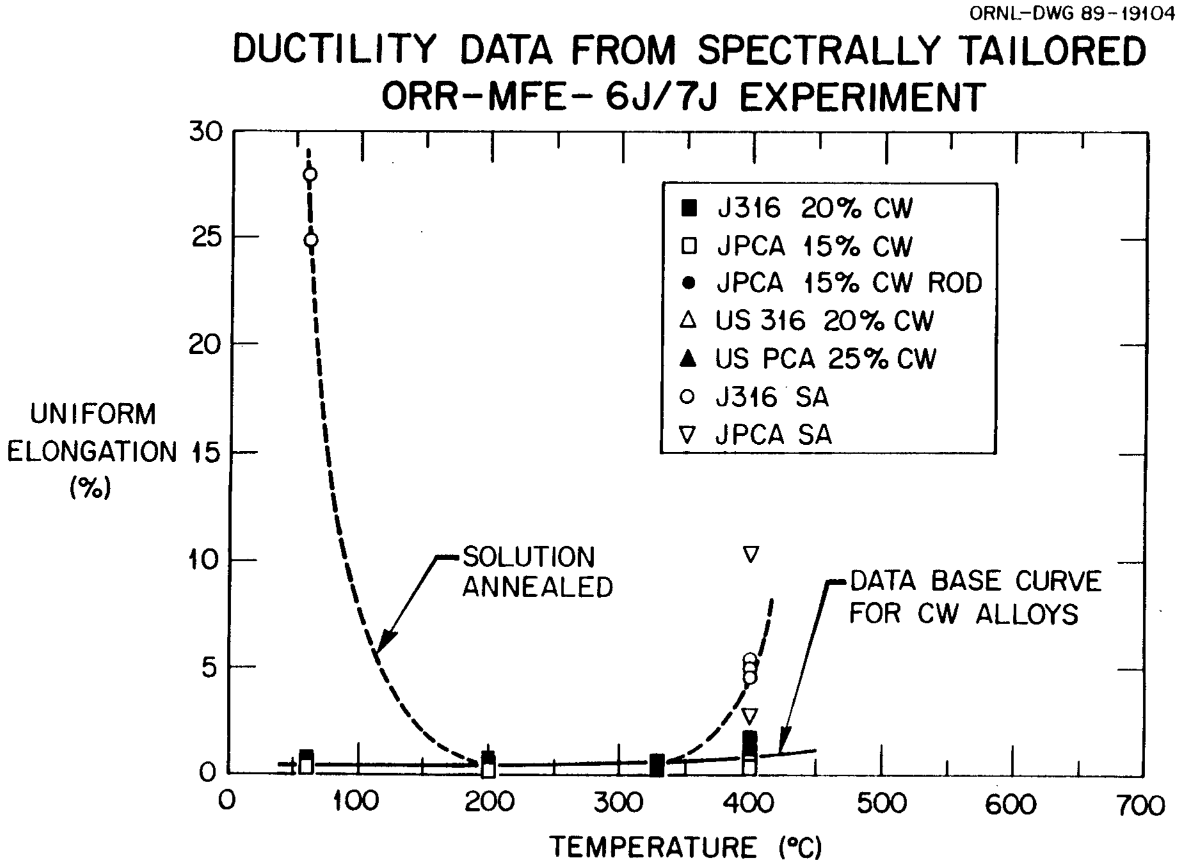


Figure 2.B.3

as well, or better, as the "normal off-the-shelf" austenitics with respect to swelling. These alloys also retained adequate ductility after high temperature irradiation to 50 dpa [3]. Unfortunately, the ductility after low temperature (200-300°C) irradiation is not as good (see Fig. 2B.3) [4].

The replacement of Mo, Ni, and Nb with Mn, W, V, and Ta has actually had a beneficial effect on the unirradiated yield strength and ductility of these alloys. Bloom et al., at ORNL [5], have shown that not only do the LA austenitic alloys compare favorably among themselves, but they are also superior to 316 SS as well.

**Ferritic/Martensitic Alloys**

Much the same conclusions have been obtained from the LA ferritic steel developments as was found in the austenitic alloy system. The removal of Mo, Ni, and Nb from HT-9 and the addition of Mn, W, or V have produced LA ferritics which reduce the long lived activation by a factor of 100 to 1,000 compared to normal HT-9 (see Figs. 2.B.1 and 2.B.2).

Irradiation of LA ferritic/martensitic alloys has shown that the induced DBTT shift for HT-9 is significantly reduced. This is illustrated in Table 2.B.3 which comes from recent work at ORNL [6]. It is clear that LA ferritics are much less sensitive to a DBTT shift than is normal HT-9.

**Vanadium Alloys**

There has been a great deal of interest in V alloys since 1986 stemming from the ARIES [7] studies and the ITER program.

### Ductility of bcc Fusion Reactor Alloys

| Alloy | Irradi. Conditions | Unirrad DBTT °C | Irrad. DBTT °C | ΔT °C |
|---|---|---|---|---|
| HT-9 | 35 dpa @ 420°C | -36 | 72 | 108 |
| 9Cr1MoVNb | 26 dpa ™ 390 °C | -25 | 27 | 52 |
| 9Cr2WVTa | 13 dpa @ 365 °C | -80 | -65 | 15 |
| V-10Cr-5Ti | 40 dpa @ 420 °C | -50 | 150 | 200 |
| V-5Cr-5Ti | 30-40 dpa @ 420 °C | -180 | -200 (degas) | -20 |

The driving force behind the interest is two-fold: to reduce long lived radioactivity and to withstand higher heat fluxes. We have seen from Fig. 2.B.1 that the long lived radioactivity (i.e., >10 years) can be a factor of 10 to 100 times less than LA ferritics. However, the penalty to pay for low long lived activity is higher short lived activity. This is particularly important when considering afterheat and the associated safety problems. Figures 2.B.4a-c illustrate that the magnitude of the afterheat in a V5Cr5Ti alloy is the same as for normal HT-9 during the first day after shutdown [8] and that there are times that the afterheat, or cumulative afterheat (energy) is even greater for V alloys than normal HT-9.

The effect of DT neutron irradiation on V alloys is essentially unknown because fission neutrons do a notoriously bad job of simulating high energy (n,p) or (n,α) reactions. Nevertheless, fission neutron irradiation does cause a major upward shift in the DBTT of the V-10Cr-5Ti system at 420□C (see Fig. 2.B.5) whereas irradiation of the V-5Cr-5Ti alloy (degassed of hydrogen after irradiation) shows little shift in the DBTT[9]. No data is available below 420□C and at high He contents where one might expect the DBTT shift to be even more pronounced.

### SiC Composites

Because of the low long lived radioactivity in SiC, it has been examined much more closely in the past 5 to 10 years. The fact that SiC can also withstand very high temperatures when cooled by He, also represents a major plus for this system. On the other hand, the irradiation behavior during and after 14 MeV neutron irradiation is completely unknown. It is known Si and C, both have very high (n,α) cross sections and these atoms are literally "burned up" at the rate of ≈1%/y [10]. The question of mechanical integrity is very critical to the viability of this material in a DT fusion environment.

Other issues, such as fabricated cost of very low impurity containing material with the ability to contain high pressure gas (currently > $10,000/kg) [11] and tritium inventory need to be investigated further .

### Solid Breeders

A major change from 1986 in this area is the loss of US neutron irradiation facilities and the cessation of internationally shared experiments on $Li_2O$ and other solid breeders [12]. However, recent experiments have shown that $Li_2O$ swells much more than other ceramics such as $Li_2ZrO_3$. It has also been shown that Li ceramics can be economically manufactured on a large scale.

Prior to the past few years, the Li ceramics were the prime candidates for static breeding material blankets. At the present time (i.e., for ITER), much more attention is being paid to liquid metals such as Li and PbLi.

### Neutron Multiplier

The main emphasis in this time period has been on the use of Be to multiply the neutrons. The problems of swelling (due to

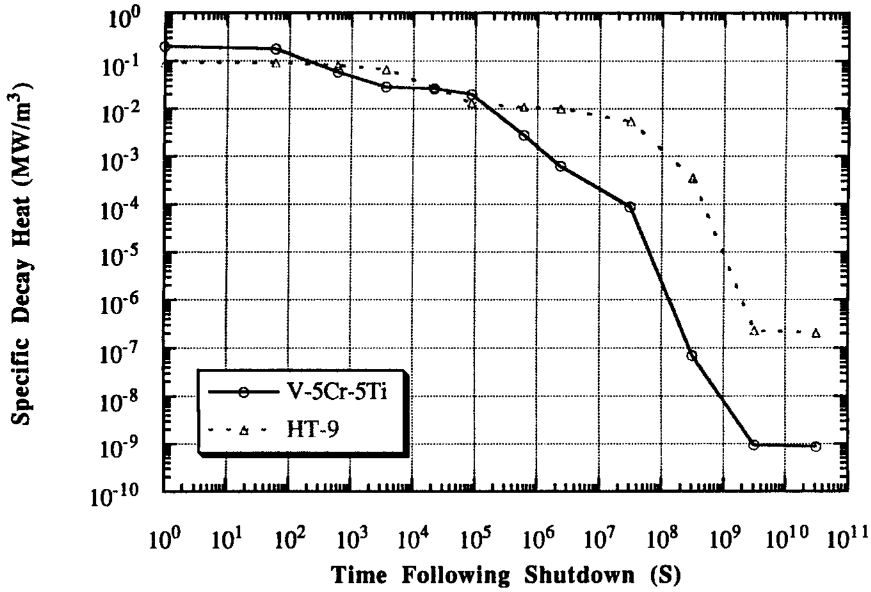


Fig. 2.B.4a. The afterheat density in V alloys is essentially the same as in normal HT-9 for the first few days after shutdown [8].

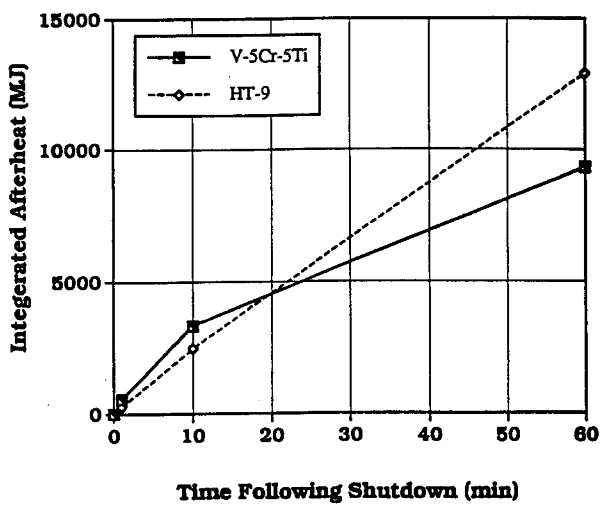


Fig. 2.B.4b. The afterheat generated for the first 20 minutes is higher than in normal HT-9 [8].

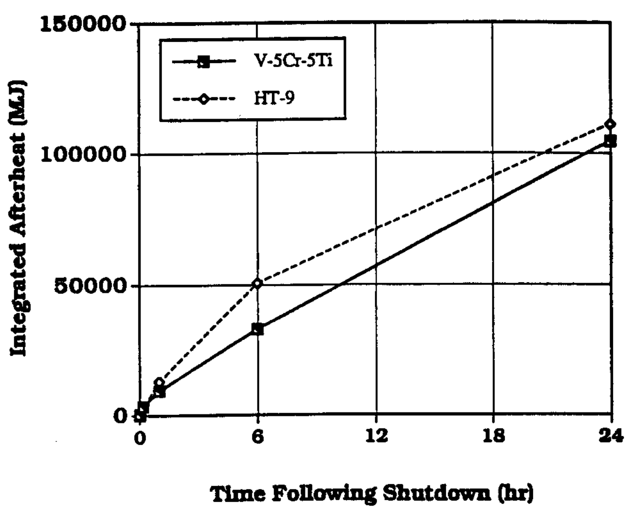


Fig. 2.B.4c. The total afterheat generated in the first day is essentially the same for V alloys and normal HT-9 [8].

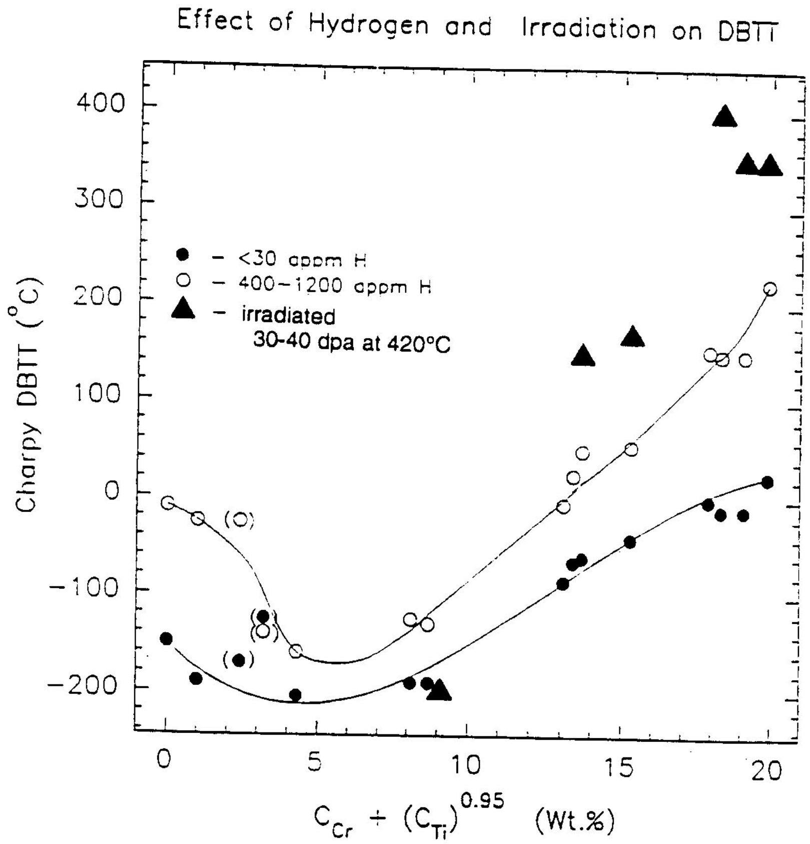


Fig. 2.B.5. Recent data indicate V-5Cr-5Ti alloy is highly resistant to radiation-induced embrittlement.

the accumulation of He gas) and the continued high cost of the Be (mainly because of its hazard potential) have not been solved.

**Electrical Insulators for Use in Liquid Metals**

The only progress of note here is the proposal to use AlN or TiN on V alloys to reduce both the MHD pumping losses and the amount of tritium absorbed by the V metal [13]. However, much more work needs to be done, especially in an irradiation field, before drawing any conclusion about the viability of this concept.

**Conclusions**

There has been enough progress since 1986 to be more confident that the ferritic/martensitic alloys should perform to the level required for commercial operation (i.e., have a useful lifetime of 1-2 FPY's). The possibility of using LA alloys also has been strengthened. However, the situation with respect to V alloys or SiC composites is not much clearer now than it was 7 years ago. Renewed interest in these systems will undoubtedly generate more experimental data in the next 5-10 years, after which one can make another assessment. It is expected that these conclusions will apply to structural materials in both tandem mirrors and stellarators.

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## C. Plasma Facing Components

Typically, plasma facing components (PFC's) are needed in three areas:

1. On the first wall (FW)
2. On divertor plates
3. In neutral-beam dumps.

Tokamaks need PFC's on the FW to protect against plasma disruptions and runaway electrons [1]. Mirror and stellarators do not need protection on the first wall.

Mirrors will need PFC's on beam dump surfaces and possibly on halo and direct conversion elements. Stellarators will need PFC's on divertor plates.

The primary elements of PFC's are:

1. A sacrificial surface
2. Structural material
3. A heat sink.

The desired features of the sacrificial material are:

- Low Z
- Low vapor pressure
- High thermal conductivity
- Low coefficient of expansion
- Low sputtering coefficient
- Low chemical reactivity
- Low activation
- Easy replacement.

Current candidates are: graphite, Be, liquid metals

- Graphite has fallen out of favor due to its high physical and chemical sputtering, the effect of irradiation on the thermal conductivity and the difficulty of replacement.

- Be is favored for its low Z, relatively low sputtering coefficient, high thermal conductivity, and the prospect of in-situ recoating. Disadvantages include the low melting temperature, chemical reactions, and the relatively high vapor pressure.

The desired features of the structural materials for PFC's are:

1. High thermal conductivity
2. Low coefficient of expansion
3. High strength
4. Low activation
5. Good radiation damage resistance.

The current candidates for PFC structural materials are: Cu-Al25, Cu-Be-Ni, Cu, and refractory metals. However, all of these generate some long lived radioactivity and there is little or no data on radiation damage on these materials at this time.

- The dispersion strengthened Cu such as Cu-Al25 and Cu-Be-Ni satisfy the first and third criteria but the maximum temperature is limited.

- Pure Cu is weak, but has the advantage of being able to be brazed to graphite directly.

- Refractory metals qualify to varying degrees of acceptance.

There are several acceptable heat sink materials:

- Hypervapotron; used on JET [2]
  Capability;
  - Steady state heat flux = 10 $MW/m^2$
  - Burnout heat flux = 18 $MW/m^2$

- MFTF-B beam dumps [3] ;
  Capability;
  - Steady state heat flux = 15 $MW/m^2$
  - Burnout heat flux = 30 $MW/m^2$

  Capability;
  - Steady state heat flux = 30 $MW/m^2$.
  - Uses subcooled flow water along with twisted tape, to give very high critical heat fluxes. These are especially used for highly peaked heat fluxes at the tube/water interface capable of 40-50 $MW/m^2$.

There is a limited number of places where plasma facing components are needed in tandem mirrors. Because there is no current, a failure of confinement will result in only the plasma thermal energy deposited on the central cell first wall, with enough attenuation in the scrape-off layer making time constants long. There is no need for protection. Similarly in the end cells, potentials will decay on a slow time scale, thus energy deposition is slow.

The only place where plasma facing components may be needed is in neutral beam dumps in the end cells. The steady state heat load is 10 $MW/m^2$ and is within the capability of a hypervapotron. However, if the beam is turned on with no plasma, the heat load rises to 34 $MW/m^2$. Instrumentation should prevent such an event; nevertheless, precautions must be taken. Even though such a heat load is too high for presently known heat dumps, it can be substantially reduced by increasing the incident area by moving it farther away.

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## D. Heating

Tandem mirror reactors will require input power to fulfill several functions:

1. Generating mirror-trapped, hot electrons and ions,
2. Pumping the thermal barrier, and
3. Providing MHD stability.

This power will be needed in the form of electron cyclotron range of frequencies heating (ECRF), ion cyclotron range of frequencies heating (ICRF), and neutral beams. The various tandem-mirror regions where this power will be required, the present state-of-the-art, and the projected reactor needs are shown in Table 2.D.1. The tandem-mirror reactor, in the axisymmetric embodiment envisioned for the present work, requires ICRF power in the central cell for MHD stabilization. The other powers are all injected into the end cell, at the midplane (barrier ECRF and barrier-pumping ICRF) and at the electrostatic potential peak (plug ECRF and plug neutral beam). In Table 2.D.1, the present state-of-the-art is given for systems presently operating in fusion experiments of any configuration but not for test-stand systems. The listed reactor needs are based primarily upon MINIMARS [1], modified for axisymmetric operation. Stellarator reactors will require the injection of some heating power into the core plasma during a short startup phase, but particle distribution requirements will not be important, and any convenient power technology could probably be used.

### ECRF

ECRF power is used in tandem mirrors to create a "hot" electron population at the midplane of the thermal barrier and to heat the "warm" electrons in the plug region on the magnetic-field slope. The first application will be at the second harmonic of the electron cyclotron frequency, while the second application is typically at the fundamental frequency. Two main technologies exist for ECRF power: gyrotrons and free-electron masers (FEM's).

ECRF power has become a well-established heating method for both tokamaks [2] and stellarators [3]. The confining magnetic fields in present day devices are in the range of $B_0$ = 1-3.5 T. Long pulse and continuous-wave (CW) gyrotron oscillators delivering output powers of 100-400 kW at frequencies of 28-48 GHz have been used very successfully for plasma ionization and start-up, electron cyclotron resonance heating (ECRH), and local current density profile control by noninductive electron cyclotron current drive (ECCD) at power levels up to 4 MW. As fusion machines become larger and operate at higher magnetic fields and higher plasma densities in steady-state, it will be necessary to develop CW gyrotrons that operate at both higher frequencies and higher mm-wave output powers. Single-mode 110-140 GHz gyromonotrons capable of high average power (0.5-1 MW/tube, CW) are currently under development. There has been continuous progress towards higher frequency and power, but the main issues are still the long pulse or CW operation and the appropriate mm-wave vacuum window. Gyrotrons at 140 GHz and 0.58 MW output power in the Gaussian free space $TEM_{00}$ mode with pulse length up to $\tau$=2.0 s and efficiency $\eta$=34% are commercially available in Russia [4]. High order rotating TE-modes (e.g. $TE_{22,6}$ at 140 GHz) are used as working modes in the cavities of the tubes [5]. The ITER conceptual design activity (CDA) reference case proposed a 20 MW, 120 GHz system consisting of 1 MW, continuously operating gyrotrons. In the case of gyrotron oscillators only slow frequency step tuning by variation of the magnetic field (change of operating cavity mode) is possible.

## Table 2.D.1

## Heading Technology Status in Present Fusion Experiments and Requirements for a Tandem Mirror Reactor

| REGION | PARAMETER | | Unit | Experiment Best to Date | Experimental Device | Current View D-T Reactor |
|---|---|---|---|---|---|---|
| Central Cell ICRF (MHD stability) | Frequency | | MHz | 43 | JET | 23 |
| | Injected Power | | MW | 22 | JET | 25 |
| Plug ECRF (warm electrons) | Frequency | | GHz | 60 | DIII-D | 70 |
| | Injected Power | | MW | 1.6 | DIII-D | 3 |
| Barrier ECRF (hot electrons) | Frequency | | GHz | 140 | W 7-AS | 110 |
| | Injected Power | | MW | 0.42 | W 7-AS | 15 |
| Plug neutral beams (ions) | Energy | | keV | 95 | JT-60U | 412 |
| | Injected Power | | MW | 40 | JT-60U | 7 |
| Barrier-pumping (ions) | Mechanism | | | Neutral beams | TMX-U | Ponderomotive drift pumping |
| | RF system | Frequency | MHz | 43 | JET | 46 |
| | | Injected Power | MW | 14 | JET | 30 |
| | Neutral beam | Energy | keV | 15 | TMX-U | Not used |
| | | Injected Power | MW | 18 | TMX-U | Not used |

Free-electron masers potentially have the capability of high power and good tunability [6], but key questions remain regarding FEM efficiency and cost. Fast and continuous frequency tuning by variation of the beam acceleration voltage is feasible for free electron masers. The most impressive output parameters are $P_{out}$ =2 GW, $\tau$ = 20 ns, and $\eta$ =13% at 140 GHz (LLNL) [7] and $P_{out}$ =15 kW, $\tau$ =20 µs, and $\eta$ = 5% in the range from 120-900 GHz (UCSB). Up to now, however, the cylindrical cavity gyrotron is the only mm-wave source which has gained an extensive data base in ECRH experiments over a wide range of frequencies and power levels.

### ICRF

Presently, two applications are envisioned for ICRF power in tandem mirrors: pumping the thermal barrier and providing RF-stabilization of MHD modes. This requires frequencies of 20-50 MHz and, in the MINIMARS design, a total absorbed power of ~55 MW [1]. The JET tokamak experiment has injected up to 22 MW of ICRF power at 43 MHz [8]. The present state-of-the-art on the test stand at these frequencies is ~4 MW per antenna for either a folded waveguide, with a power density of ~40 MW/m$^2$, or a strap antenna with a power density of ~16 MW/m$^2$ [9]. Thus, the ICRF requirements for a tandem mirror could be met with very little extrapolation beyond existing technology.

### Neutral Beams

Neutral beams are used in tandem mirrors to create a population of mirror-trapped, hot ions. In a D-T tandem reactor, the beam energy must be ~400 keV at a power of $\lesssim$ 10 MW. In TMX-U, 18 MW of 15 keV neutral beams were injected into the end cells for pumping the thermal barrier and maintaining the plug-ion density [10]. The JT-60U experiment has injected 40 MW of 95 keV neutral-beam power [11], and TFTR has injected 33 MW of 80 keV neutral-beam power [12]. The neutralization efficiency for positive-ion sources, such as used in JT-60U, is low at 400 keV, and negative-ion source technology must be used. The main difficulty for these sources is getting a sufficiently high yield, but good progress has been made during the past decade. An experimental source has provided 145 mA of $D^-$ [13]. At the Japan Atomic Energy Research Institute (JAERI), a 10-A negative ion source has been tested which produces a 50 keV beam with a current density of 370 $A/m^2$ and a pulse width of 0.1 s [13]. A 500 keV, 10 MW, 10 s neutral beam injection system has been proposed for the JT-60 experiment.

**Summary**

The heating technologies required for tandem mirror and stellarator development have made good progress in the past decade, and it can be expected that systems with the necessary performance will be available on the time scale of fusion power development. The key outstanding questions are the efficiencies of the ECRF and neutral-beam systems, which are primarily economic, rather than technical issues. Heating is much less of an issue for stellarators, with only startup power probably required, and several technology options are likely to be possible.

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## E. Impurity Control

Tandem mirror experiments generally operate with much lower impurity levels than do tokamaks and stellarators. This is theoretically expected, because the cylindrical halo (scrape-off layer) plasma that surrounds the core plasma is an extremely efficient vacuum pump [1]. Thus, plasma that transports radially across the magnetic flux tubes swiftly moves axially and builds up a pressure against the end walls. This gas is then pumped efficiently by standard vacuum pumps at the ends of the vacuum chamber. In addition, the core plasma operates at a positive electrostatic potential relative to the chamber walls. This tends to expel the impurities from the plasma core. The experimental value of $Z_{eff} + \Re n_i Z_i^2 / n_e$ is usually very close to 1 in tandem mirrors, while it is generally 2-3 in tokamaks. Recent work also indicates that fusion products may help purify the core plasma by colliding with impurities and sometimes transferring sufficient energy to push them over the confining electrostatic potential [2-3].

Nevertheless, there remain some impurity control issues to be addressed and solutions to be demonstrated. The key issue is that, should even a small amount of impurities get into the core plasma, they will quickly scatter into the thermal barrier due to their high Z value. This is potentially a problem, because such impurities reduce the thermal barrier depth proportionally to Z. Therefore, the thermal-barrier pumping system must be designed to pump impurities ( not only the fuel and fusion-ash ions. Although the same ponderomotive drift pumping techniques anticipated for use on the D-T fuel ions should work, this method remains to be demonstrated [4].

In summary, both the experimental data base and theoretical considerations indicate that impurity control should not be a major issue for tandem mirror reactors. Techniques that must be developed to solve the fundamental problem of thermal barrier pumping should be able to handle, without excessive power or complexity, the small number of impurity ions expected to accumulate in the thermal barriers.

Stellarator confinement systems possess a natural helical diverter which occurs as a consequence of the existence of the magnetic separatrix bounding the region of closed nested flux surfaces. Inside the separatrix, the enclosed magnetic flux links all of the magnet coils; however, on the outside of the separatrix, the flux links some, but not all the coils. For this region some of the flux must emerge from spaces between the coils, but to conserve flux, it must reenter the device at some other location. This phenomenon has been observed on small and large stellarator experiments as evidenced by well defined burn marks on the vacuum chamber walls [5,6,7,8]. Particle diverter plates or collection areas can be located inside the vacuum chamber before the flux lines emerge between the coils, or they can be located outside the coils [9]. For power reactors it may be more practical to locate them inside the vacuum chamber.

In modular stellarator designs, the same type of diverter action occurs. In the HELIAS concept, optimization can provide sharp edges formed in the outermost magnetic surface which are helix-like, and lead, on the outboard side of each field period, from the lower to the upper ends of indented cross sections one period apart [10]. The field lines in the region beyond, but close to, the last closed flux surface are displaced radially when they cross these edges. Trough-like collector surfaces are arranged to follow these edges at some distance from the plasma, extending somewhat longer than a field period. Assuming an anomalous diffusion coefficient at the plasma boundary on the order of 1 $m^2/s$, the diverter plates can be made sufficiently large as to produce a smooth particle load distribution leading to a power density of several $MW/m^2$ which is considered tolerable [10]. Such a concept also allows sweeping by very moderate AC magnets to reduce the peak power density on the collector plates.

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## F. Maintenance

Maintenance of tandem mirror reactors primarily entails blanket module changeout due to radiation damage. The linear geometry is a big help, allowing lateral movements of the central cell components. However, because the blanket modules are inside the central cell (cc) coils, the most expedient method of removal is to displace whole cc modules. Careful planning is needed to insure that coolant and power connections are made in such a way as to avoid interference with maintenance operation. Because the blanket module are concentric with the cc coils, their removal and replacement once the cc module is removed from the reactor is relatively simple.

The normal coil inserts of the choke coils have a typical life of ~3.5 years, and have to be replaced. Here again, lateral displacement of the choke coil module will be needed to perform this operation. These coils weigh on the order of 200 tonnes.

The end cell coils will also be moved as a single unit. However, the only maintenance required here is if a coil fails and must be replaced. These coils are normally lifetime components and cumulatively weight ~470 tonnes.

Frequent replacement of the neutral beam dumps may be needed. These can be designed as bayonets and are easily accessible and replaceable. A typical weight of a neutral beam dump bayonet is 0.5 tonne.

The stellarator has a toroidal geometry and thus resembles the tokamak rather than a tandem mirror in that respect. Much work on maintenance that has been done for tokamaks would be applicable to stellarators, particularly in the area of diverter plate maintenance, and dust collection [1]. Maintenance of diverter plates is by means of articulated booms inserted between coils [2].

Blanket changeout in stellarators differs considerably from tokamaks. This is particularly true of modular stellarators such as the Helias class. The maintenance concept for these reactors is based on the modularity of the coil system. The absence of poloidal field coils and other interlocking coil systems in the Helias reactor allows it to be modularized according to periods. Six coils making up a period can be displaced horizontally such that the first wall and blanket are accessible from the open ends. To perform this, the vacuum vessel has to be cut on either side of the coil period, and for this, techniques developed for NET can be used [3]. This procedure, however, is complicated by the fact that the blanket cross section is not uniform over a field period. A maintenance scheme for a Helias type reactor, ASRA6C [4] was developed in which the exchange of all the blanket units in all the field periods was performed during a single shutdown. One field period after another, each weighing 2200 tonnes, was to be removed sequentially from the main ring by special transport vehicles. No estimate of the time needed to perform the blanket replacement was made. In another scenario, groups of six or four coils are taken out radially. Because of flip-over symmetry, the group can be separated into two subunits of either three or two coils. Two of these subunits can then be stored for exchange, constituting only 10% of all the subunits in the reactor. More studies on the maintenance of Helias type stellarators are needed.

Other remote maintenance functions such as cutting/welding coolant lines, disconnecting NB lines, servicing RF antennae where applicable, locating and fixing leaks in vacuum vessels and servicing cryogenic systems are all common to tandem mirrors, stellarators and tokamaks. In this regard, the systems being developed for NET/ITER would all be applicable here.

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# CHAPTER 3

# Physics Extrapolations Required for Commercial Tandem Mirror Reactors

## A. Introduction

The state of tandem mirror research has not advanced as far as that of the tokamak, because the tandem mirror is a more recent invention and there have been only a few tandem mirror experiments compared with the more numerous and much larger tokamaks. Nevertheless there have been many successes in tandem mirror physics. In this chapter we present a plan which builds on these successes to bring the tandem mirror from its present state to the parameters needed for a commercial reactor. In the first report, "Status of Tandem Mirror Research-1992" [1], we discussed critical issues which need to be addressed if the tandem mirror is to be considered for a reactor. We also presented a suggestion for some important "next steps" in the tandem mirror program. In this section, we elaborate on those "next steps", i.e. the next experimental facilities needed to advance the tandem mirror program.

A few selected parameters for the present experimental database and the anticipated requirements for a commercial power reactor are shown in Table 3.A.1. These data are taken from [1]. It should be noted that the experimental values were not achieved simultaneously, but represent the best values obtained in several machines; the references for this data are given in [1]. The gap between the present database and the expected reactor parameters is wide and needs to be filled in with some intermediate points in order to establish the suitability of the tandem mirror reactor concept for commercial energy production. We propose that at least two facilities are needed between the present situation and a reactor. The parameters for these two facilities are also given in Table 3.A.1. The first, called "Proof-of-Principle Experiment", is a device about the same size as the current GAMMA-10 machine, but modified for purely axisymmetric operation by replacing the quadrupole magnets with RF stabilization and axisymmetric end cells. The second facility, called "Prototype Reactor", is an upgrade of the MFTF-B facility to axisymmetric operation using RF stabilization and somewhat more advanced plasma parameters.

## B. Proof of Principle Experiment

The goals of the Proof-of-Principle Experiment are to demonstrate the feasibility of purely axisymmetric operation and to demonstrate thermal barrier operation at higher central cell density and thereby achieve substantial electron temperature and parallel ion energy in the central cell. The Proof-of-Principle Experiment is essentially a modification of GAMMA-10 for axisymmetric operation using RF stabilization and with improvements in the thermal barrier to produce better axial energy confinement and generate higher electron temperatures (2 keV) in the central cell. If successful, this machine will clearly demonstrate significant parallel energy confinement using electrostatic potentials and resolve many of the critical physics issues concerning tandem mirror operation. Critical issues to be addressed by this machine include loss of end-plugging and improved thermal barrier pumping, axisymmetric operation with RF stabilization, microstability, radial transport losses, and impurity control. The Proof-of-Principle Experiment would not use tritium fuel since the temperature in the central cell is too low for meaningful fusion energy production.

## C. Prototype Reactor

The Prototype Reactor has the objective of scaling up the parameters to values closer to what is needed for a reactor. This facility represents a modest scale-up of the now-canceled MFTF-B, but with RF-stabilization and axisymmetric end cells instead of the Yin-Yang magnets constructed for MFTF-B.

**Table 3.A.1**

**Summary of Possible Experimental Steps to a Tandem Mirror Fusion Power Plant**

| | Unit | Experimental Best-to-Date | Proof-of-Principle Experiment | Prototype Reactor | Current View D-T Reactor |
|---|---|---|---|---|---|
| | | | | | |
| **Configuration** | | Yin-Yang | Axisymmetric | Axisymmetric | Axisymmetric |
| **Central Cell** | | | | | |
| Fuel | | D-D | D-D | D-T | D-T |
| Fusion power | MW | – | – | 50 | 1600 |
| Ave. beta | % | 13 | 25 | 50 | 60 |
| Electron temperature | keV | 0.28 | 2 | 10 | 20 |
| Parallel ion temperature | keV | 0.4 | 5 | 15 | 25 |
| Perpendicular ion temperature | keV | 5.6 | 5 | 15 | 25 |
| Ave. ion density | $m^{-3}$ | 1E+19 | 1.3E+19 | 1.7E+20 | 2.7E+20 |
| n-tau parallel | $m^{-3}$ s | 1E+19 | 2E+19 | 1E+20 | 1E+21 |
| Length | m | 10 | 10 | 15 | 90 |
| Plasma radius | m | 0.25 | 0.25 | 0.45 | 0.45 |
| Vacuum magnetic field | T | 0.4 | 0.4 | 2 | 3 |
| Fueling | | Gas or Beams | Pellets | Pellets | Pellets |
| **Barrier** | | | | | |
| Hot electron energy | keV | 100 | 100 | 200 | 300 |
| Barrier electrostatic potential | keV | 1.1 | 5 | 30 | 125 |
| Maximum choke-coil field | T | 3 | 6 | 15 | 24 |
| **Plug** | | | | | |
| Hot ion energy | keV | 10 | 50 | 200 | 500 |
| Warm electron temperature | keV | – | 10 | 80 | 275 |
| CC ion-confining potential | keV | 2 | 20 | 70 | 150 |
| Plug peak magnetic field | T | 3 | 4 | 6 | 10 |

The parameters of the Prototype Reactor call for a substantial ion temperature in the central cell (15 keV); operation with tritium would provide useful data important to the various physics questions associated with burning plasmas. In addition, the Prototype Reactor would address MHD stability, microstability, thermal barrier physics, and impurity control issues at larger plasma size and higher density and temperatures than achieved in the Proof-of-Principle Experiment. The step from the Proof-of-Principle Experiment to the Prototype Experiment is roughly a factor of 3 in central cell density, 5 in central cell electron temperature, and 5 in $n\tau_{||}$. Achieving these parameters requires an increase in the thermal barrier hot electron energy by a factor of 2, an increase of the plug hot ion energy by a factor of 4, and an increase of the plug warm electron temperature by a factor of 8. The step from the Prototype Experiment to reactor parameters is a factor of 6 in central cell density, 2 in electron temperature, and 10 in $n\tau_{||}$. Shown in Fig. 3.C.1 is an $n\tau$ versus $T_i$ plot with the Proof-of-Principle and Prototype Reactor parameters shown. Fig. 3.C.2 shows the progression of some relevant physics parameters from where we are now to the Proof-of-Principle Experiment, Prototype Reactor, and finally the commercial power reactor.

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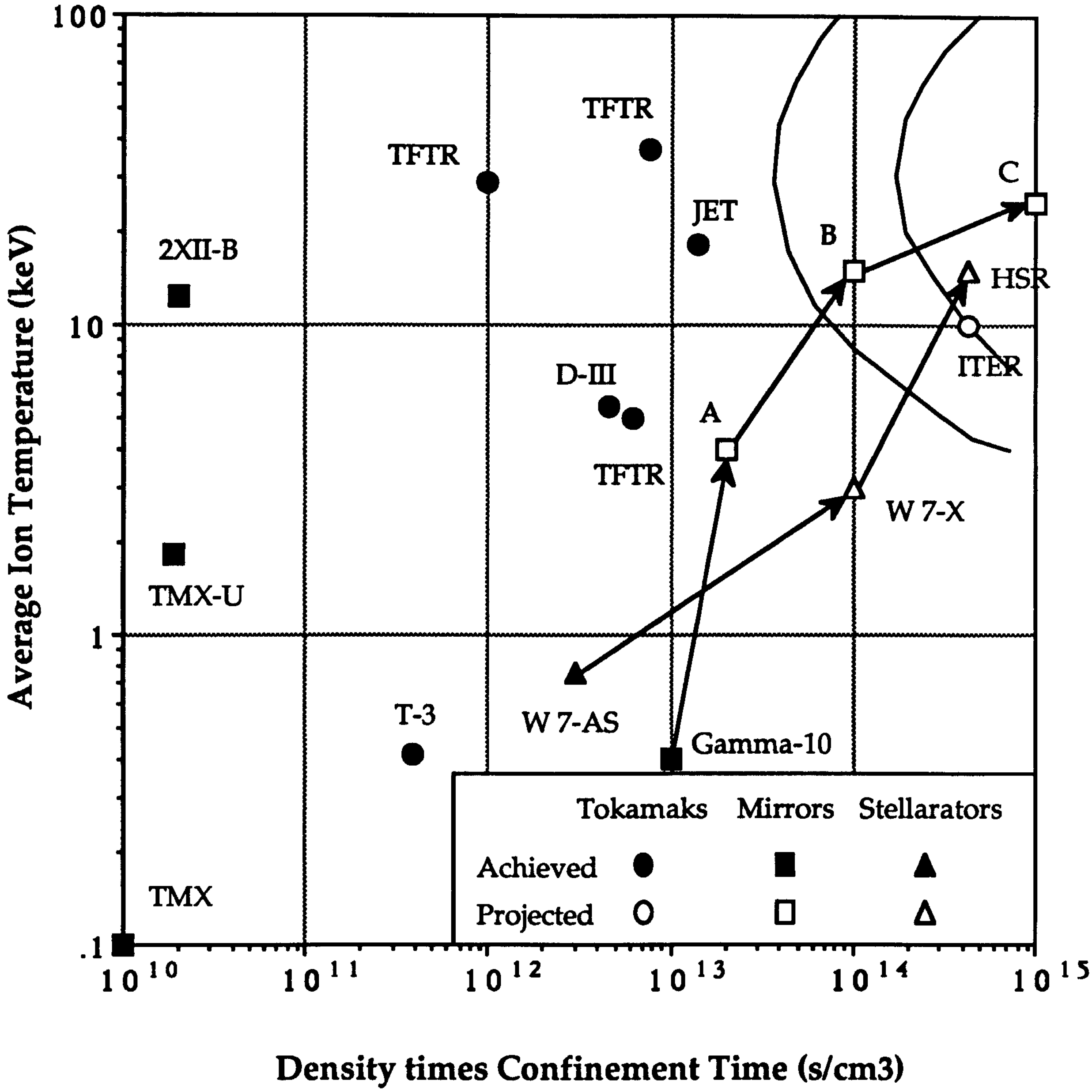


Fig. 3.C.1. Lawson diagram showing tokamaks, mirrors, and stellarators. The development path of tandem mirrors is Gamma-10, Proof-of-Principle (A), Prototype Reactor (B), and the Commercial Reactor (C). The development path for stellarators is W 7-AS, W 7-X, and HSR; this is discussed in Chapter 4.

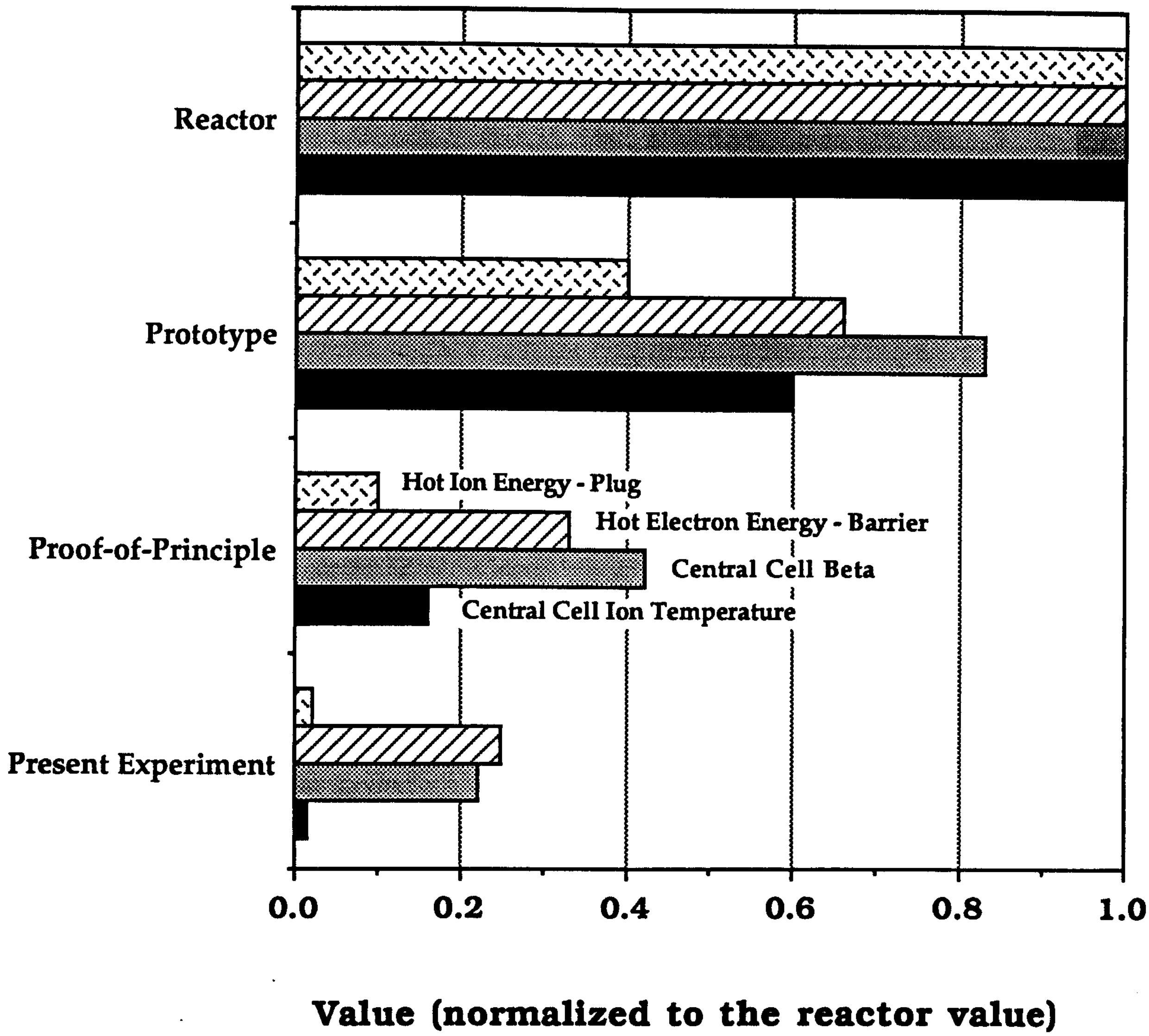


Fig. 3.C.2. Progression of physics parameters from present experiments to the reactor.

# CHAPTER 4

## Physics Extrapolations Required for Commercial Stellarator Reactors

### A. Introduction

While the stellarator is one of the earliest magnetic confinement concepts, its development has lagged behind that of tokamaks. The recent development of modular coils replacing the helical windings, and optimization of the magnetic field topology made possible with the introduction of the helical magnetic axis have generated renewed interest in the stellarator as a reactor concept. The stellarator has many features in common with the tokamak since both are based on nested toroidal flux surfaces for plasma confinement, but it also avoids some of the disadvantages of the tokamak concept:

- Because there is no net toroidal current in the plasma, plasma disruptions cannot occur and there is less free energy available for plasma instabilities.

- The stellarator has local diamagnetic currents driven only by the plasma pressure and thus does not need a current drive system. Hence, true steady-state operation depends only on being able to refuel the plasma and to remove the fusion produced ash.

A leading stellarator experiment is the Wendelstein 7-AS device in Garching; a more advanced stellarator, Wendelstein 7-X, utilizes the helical magnetic axis concept and is in the planning stage. Reactor system studies based on modular stellarators [1] and the helical magnetic axis [2] and continuous coil [3] concepts have been carried out. The discussion here focuses only on the helical magnetic axis concept. Table 4.A.1 shows a few plasma parameters achieved in Wendelstein 7-AS [4] and the corresponding parameters for Wendelstein 7-X [2] and for the reactor, HSR [2].

### B. Wendelstein 7-X

The task of the Wendelstein 7-X experiment is to provide an integrated concept test needed for establishing the properties of ignited plasmas in Advanced Stellarators. The size is chosen to meet the minimum conditions needed for optimization of the stellarator configuration with reasonable space for successful divertor performance. W 7-X will provide needed data on the extrapolation of present confinement scaling relationships towards the reactor regime and reduce the uncertainty in the predicted performance of the reactor. It will also be capable of achieving the beta values needed for the reactor which is presently a serious concern. The value of the confinement parameter, $n\tau T$, predicted for W 7-X depends on the scaling law used. Lackner-Gottardi scaling has the proper density and iota dependencies and gives the larger value while LHD scaling gives the smaller value. An important role for W 7-X will be to provide confinement data for larger, denser, and hotter plasmas and reduce the uncertainty in the confinement to be achieved in the reactor.

W 7-X will not use D-T fuel so it will not be able to study burning plasma physics. One can argue that, because of the similarity with tokamaks, the burning plasma physics results obtained from tokamaks can be applied to Advanced Stellarators.

### C. The Stellarator Reactor, HSR

The HSR stellarator reactor is about a factor of three larger than W 7-X in its linear dimensions and has twice the magnetic field strength, but with the same magnetic configuration. The value of $n\tau T$ in Table 4.A.1 for the reactor is determined by power balance considerations and not confinement scaling relationships. Lackner scaling provides the required confinement in HSR without any improvement in the coefficients in the scaling expression, while LHD scaling requires improvement by about a factor of

**Table 4.A.1**

**Plasma Parameters for the Wendelstein Series of Stellarators**

| Parameter | Unit | Experimental Best to date (W 7-AS)* | Next Step Machine (W 7-X) | Current View DT-Reactor (HSR) |
|---|---|---|---|---|
| Major radius | m | 2.0 | 5.5 | 19.5 |
| Minor radius | m | 0.17 | 0.5 | 1.6 |
| Magnetic Field | T | 2.5 | 3.0 | 5.0 |
| Plasma Density | $10^{20}$ $m^{-3}$ | 2.5 | 2.5 | 2.8 |
| Ion Temperature | keV | 0.75 | 3.0 | 15 |
| Beta | % | 0.6 | 4.3 | 4.6 |
| Fusion Power | GW | -- | -- | 2.9 |
| $n\tau T$ | $10^{20}$ $m^{-3}$ s keV | 0.025 | 0.5 - 3.0 | 65 |

* parameters not achieved simultaneously.

two. The extrapolation in the value of the confinement parameter, $n\tau T$, from W 7-X to HSR is about a factor of 20 for Lackner scaling and 100 for LHD scaling. Fig. 3.C.1 shows the values of $n\tau$ and T expected for W 7-X and HSR.

### References for Chapter 4

[1] G. Boehme et al., "Studies of a Modular Advanced Stellarator Reactor ASRA6C", KfK Report KfK 4268, June 1987.

[2] G. Grieger et al., "Modular Stellarator Reactors and Plans for Wendelstein 7-X", *Fusion Technology* **21**, 1767 (1992).

[3] J.F. Lyon, B.A. Carreras, V.E. Lynch, J.S. Tolliver, I.N. Sviatoslavsky, "Compact Torsatron Reactors", *Fusion Technology* **15**, 1401 (1989).

[4] C.D. Beidler et al., Proc. of the 14th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research, Würzburg, Germany, 1992, IAEA-CN56/G-I-2, Vienna (to be published).

# CHAPTER 5

## Comparison of Physics and Technology Requirements for Both Concepts

### A. Physics

Figure 3.C.1 illustrates in a simple schematic fashion the state of affairs for tandem mirror reactors and for stellarator reactors. Both concepts lag behind tokamaks in the parameters achieved so far. The extrapolation in parameters needed to achieve the reactor is large for both concepts, but has a different character for the tandem mirror than for the stellarator. For the stellarator, the reactor represents a large increase in plasma size, magnetic field strength, ion temperature, and energy confinement time from present day experiments. Present scaling expressions provide the needed increase in confinement time (at least within a factor of two) because of the increase in size, field strength, plasma density and temperature.

The tandem mirror approach, on the other hand, represents less of an extrapolation in some parameters, but more in other parameters to achieve the reactor parameters. For example, the required extrapolation in central cell beta and hot electron energy is less than a factor of 10 and the magnets for the axisymmetric tandem mirror are simple coils. More extrapolation is required in the central cell density and electron temperature. The tandem mirror depends on more complicated and subtle physics to achieve MHD and microstability (e.g. hot electrons, warm electrons, passing and trapped particles, etc.). These concepts have been demonstrated in isolation, but a complete and simultaneous demonstration of all the required concepts has yet to be done.

### B. Magnets

A comparison of the typical magnet systems required for stellarator and tandem mirror power reactors is given in Table 5.B.1. It can be seen that the required maximum magnetic fields for both tandem mirrors and stellarators are well within the range of available NbTi and $Nb_3Sn$ superconductors. Thus, the major issues have to do with optimizing and tailoring the conductors for the specific application, and improving coil winding technologies.

For the tandem mirror central cell coils, it can be stated unequivocally, that such solenoid technology at the required field and current density utilizing the well tested NbTi superconductors, even for full scale power reactors, is currently available in industry. NbTi conductors, subcooled to 1.8 K, will also suffice for stellarators. It should also be said that if prototype conductors as presently developed for W 7-X and the coil technology in progress for it turns out to be successful, then it should be possible to extrapolate them directly to the reactor size.

The choke coils for tandem mirrors are seen to be in the direct path of the development of high field solenoid technology which is advancing very rapidly. Coils of 20 T in 15 cm bore, utilizing cooling at 1.8 K are envisaged within the next several years (e.g. HOMER-II at KFK). These parameters are already close to the choke coil specifications, and so their realization is not far off.

High field octupole and mirror coils for the end cells of tandem mirrors remain somewhat of a challenge. Whereas octupole coil technology can be seen as similar to that of modular stellarator coils, their high magnetic field will not allow the use of the same conductor and winding technique. A possible choice might be a cable in conduit $Nb_3Sn$ conductor encased in a steel or Incoloy conduit. Such a conductor will require complicated winding machines, and heating the whole coil to 700°C in order to react the superconductor (the latter technology will be available from the ITER experience).

**Table 5.B.1**
**Comparison of the Magnet System Parameters Required for Stellarator and Tandem Mirror Power Reactors**

| | | Stellarator | Tandem Mirror | | |
|---|---|---|---|---|---|
| **Issue** | **Unit** | | **Central Cell** | **Choke Coil** | **End Cell** |
| Peak Magnetic Field | T | 10-11 | 4.7 | 16 S/C<br>8 Normal | Octupole 15<br>Mirror 10 |
| Current Density in Winding | kA/cm 2 | 2.5-3.0 | 3.9 | 1.9-2.9 S/C<br>0.17-2 Normal | Octupole 7.05<br>Mirror 6.0 |
| Stored Energy | GJ | 70-90 | 3.17 | 4.4 | NA |
| Inside Diameter of SC Magnet Coils | m | 10-12 | 3.46 | 0.17 | Not App |
| n & $\gamma$ radiation load | mW/cc | 0.1-0.5 (local) | 0.1 | 0.61-4.8 | 1.0 |
| Cold Mass | ktonne | 12-15 | 0.938 | 0.034 S/C<br>4.02 Normal | 0.93 |
| Supporting Structure | ktonne | 8-10 | 0.064 | NA | 0.15 |
| Cyrogenic Cooling Power at 4.2 K | kW | 50-100 | 21 | 2.9<br>(0.11@ 1.8°K) | 13.5 |
| Mechanical Sresses<br>• in-plane<br>• out-of-plane<br>• shear | MPa | 120-150<br><br>30-50 | 210<br><br>60 | Cond. Struct.<br>228 483<br><br>65 150 | 200<br><br>55 |
| Possible Additional Coil Systems Req'd | | Divertor Sweeping Coils | | | |

**Table 5.C.1.**
**Critical Parameters for Structural Materials in DT Fusion Reactors**

| Parameter | Tandem Mirror MINIMARS [1] | Stellarator ASRA-6C [2] |
|---|---|---|
| Average First Wall Neutron Load $MW/m^2$ | 3.3 | 1.4 |
| Average First Wall Heat Load $MW/m^2$ | 0.36 | 0.24 |
| Coolant | PbLi | PbLi |
| Breeder | PbLi | PbLi |
| Dynamic Behavior | Steady State | Steady State |
| Structural Material | HT-9 | HT-9 |

The cryogenic cooling load requirements will be somewhat higher for stellarators due to the larger coil volume and the need for 1.8 K cooling in all the coils. However, this is not a critical issue in either case.

In conclusion, due to the well advanced state of the art in fusion magnet technology, the coil system for both confinement schemes are feasible up to power reactor size. While central cell coil technology for tandem mirrors is fully available today, modular stellarator coil technology will be demonstrated in this decade. Some development effort which is not presently being undertaken will be needed for the tandem mirror end cells.

## C. Structural Materials

Unlike the physics and plasma heating technologies, the structural materials requirements for tandem mirrors and stellarators are quite similar. The critical parameters that can influence the structural material choice are:

- Neutron flux, fluence, and spectra
- Heat flux on the wall facing the plasma
- Coolant/breeder choice
- Dynamic stress effects.

The conditions that reactor designers have thus far chosen for these parameters are listed in Table 5.C.1.

It is obvious from a quick perusal of Table 5.C.1 that there is very little difference between the two confinement concepts when it comes to the choice of a structural material. Both the neutron fluxes and heat fluxes are modest and the steady state operation of both types of concepts makes liquid metal cooled steel combinations a reasonably attractive choice.

One should not interpret the information in Table 5.C.1 too literally because there would be many possible choices of structural materials for both tandem mirrors and stellarators. It is mainly a coincidence that exactly the same structure/coolant/breeder materials were chosen. In summary, it is safe to say that one could not decide on the viability of either concept on the basis of structural material requirements.

### References for Section 5.C


[1] J.D. Lee, Technical Editor, "MINI-MARS Conceptual Design: Final Report", Lawrence Livermore National Laboratory Report, UCID-20773, Vol. I & II, Sept. 1986.

[2] G. Böhme et al., "Studies of a Modular Advanced Stellarator Reactor ASRA-6C," Max-Planck-Institut für Plasmaphysik, Report IPP 2/285, May 1987.


## D. Plasma Facing Components

A comparison of the requirements for plasma facing components in stellarator and tandem mirror power reactors is given in Table 5.D.1. Plasma facing components for tandem mirrors and stellarators are not nearly as demanding as those for tokamak

**Table 5.D.1.**

**Comparison of Requirements for Plasma Facing Components in Stellarator and Tandem Mirror Power Reactors**

| | | Stellarator | Tandem Mirror | | |
|---|---|---|---|---|---|
| **Issue** | **Unit** | | **Central Cell** | **Choke Coil** | **End Cell** |
| Type/Position of PFC | | "island or ergodic divertor" | First Wall | First Wall | Neutral Beam |
| PFC Surface Material. | | C,W,Mo,Be,Li | None | None | Cu-1Cr |
| Peak Heat Load | $MW/m^2$ | ≈5 | 0.36 | 0.30 | 34 |
| Average Heat Load | $MW/m^2$ | 1-3 (local) | 0.36 | 0.30 | 10.1 |
| Peak Surface Temperature | °C | 560 | 524 | <500 | NA |
| Maximum Neutron Wall Loading | $MW/m^2$ | <5 | 3.3 | 2.8 | <0.5 |
| Neutron Exposure Before Changeout | $MWy/m^2$ | 20 | 20 | 20 | NA |
| Peak Displacement Rate | dpa/s (steel) | $3 \bullet 10^7$ | $1.1 \bullet 110^6$ | $8.3 \bullet 10^7$ | |
| Maximum Displacements per Life | dpa | 200 | 200 | 200 | 200 |
| Peak Particle Fluxes | $\#/cm^2$-s | | | NA | NA |
| Ave. Energy of Particles | keV | | 25 | NA | 420 |
| Number of Pulses-Lifetime | | <100 | <100 | | NA |
| Pulse Length | s | $110^6$ to $10^7$ | $10^6$ to $10^7$ | $10^6$ to $10^7$ | $10^6$ to $10^7$ |

NA = Not Applicable

reactors. Neither one requires protection on the first wall. Stellarators will have particle collection plates distributed over a large area, reducing the average heat load. For example in a Helias reactor, the total diverter area is ~100 $m^2$ and the heat load is on the order of several $MW/m^2$. If necessary the peak loads can be reduced by using moderate field sweeping coils. Such loads are very low compared to tokamaks.

The steady state heat load on a beam dump in the end cell of a tandem mirror is on the order of 10 $MW/m^2$ and is within the capability of a hypervapotron. However, if the beam is turned on when there is no plasma in the chamber, the heat load rises to 34 $MW/m^2$ and is too high. It can be substantially reduced by moving the beam dump further out and making it within the range of an MFTF-B type of beam dump. We can therefore say that the state of plasma facing components for both system is within the capability of present day technology.

## E. Heating

For both tandem mirrors and stellarators, the necessary heating technologies are reasonable extrapolations beyond present systems. The type of heating and the functions of this power, however, differ greatly between the configurations. The input power in a tandem mirror will be steady-state and mainly localized in the relatively small volume of the end cells, whereas the input power in a stellarator will be only for startup, but it must be injected into the fusion core plasma. Table 5.E.1 compares heating parameters for tandem mirror and stellarator reactors.

Tandem mirrors require power for pumping ions, including fusion ash and impurities, out of the thermal barrier potential well. As presently envisioned, the barrier-pumping power would be ion cyclotron range of frequencies power (ICRF). Axisymmetric tandem mirror geometry, which we have used for defining typical reactor parameters in this report, may require some ICRF in the central cell to provide MHD stability if wall stabilization does not suffice. The intrinsic geometry of the stellarator configuration gives it its MHD stability, albeit at much lower $\beta$ values, and no input power is needed for this purpose. The tandem mirror requires low power (≳ 10 MW) neutral beams in the end cells, but these must be high energy (≲ 400$ keV) and will require negative-ion source technology. The tandem mirror will utilize electron cyclotron range of frequencies power (ECRF) to create a population of magnetic mirror-trapped hot electrons in the thermal barrier and to heat the electrons trapped in the plug potential to a temperature of 3-5 times the central cell electron temperature. The required frequencies will be ~28-140 GHz and the total power will be ~50 MW. The steady-state power in a tandem mirror would likely be sufficient for startup, while a stellarator will need an auxiliary input power system to operate for a short startup period. Potentially, this auxiliary power may be in the form of ECRF ICRF.

## F. Impurity Control

Both experimental and theoretical considerations indicate that impurity control for tandem mirrors should not be a major issue. This is due to the fact that the cylindrical halo plasma that surrounds the core plasma is an extremely efficient pump. Also the core plasma operates at a positive potential relative to the chamber walls, expelling the impurities from its core. However, the thermal barrier pumping system must be designed to pump impurities as well as fuel and fusion-ash ions. Yet to be demonstrated, ponderomotive drift pumping techniques is considered to be capable of accomplishing this task.

**Table 5.E.1**

**Comparison of Heating Requirements for Stellarator and Tandem Mirror Power Reactors**

| | | Stellarator | Tandem Mirror | |
|---|---|---|---|---|
| **Issue** | **Unit** | | **Central Cell** | **End Cell** |
| ECRF Power/Frequency | MW/GHz | 40-50/140 (Startup Only) | | 18/110 |
| Plug ECRF (Warm Electrons) Power/Frequency | MW/GHz | | | 3/70 |
| ICRF Power/Frequency | MW/MHz | | 25/23 | 30/46 |
| Plug Neutral beams (Ions) Power/Energy | MW/keV | | | 7/412 |

**Table 5.G.1**
**Power Flow Parameters in Tandem Mirror and Stellarator Reactors [1]**

| Parameter | Symbol | Stellarator | Tandem Mirror |
|---|---|---|---|
| Fusion power | $P_f$ | 3000 MW | 1290 MW |
| Injected power | $P_{inj}$ | -- | 80 MW |
| Injection efficiency | $h_{inj}$ | - | 0.7 |
| Alpha-heating power | $P_\alpha$ | 600 MW | 258 MW |
| Neutron power | $P_n$ | 2400 MW | 1032 MW |
| Blanket energy multiplication | M | 1.36 | 1.36 |
| Power to thermal converter | $P_{th}$ | 3666 MW | 1611 MW |
| Thermal efficiency | $h_{th}$ | 0.4 | 0.4 |
| Power to direct converter | $P_{dc}$ | -- | 131 MW |
| Direct converter efficiency | $h_{dc}$ | -- | 0.63 |
| Total recirculating power | $P_c$ | 80 MWe | 127 MWe |
| Recirculating power fraction | | 0.055 | 0.175 |
| Gross electric power output | $P_e$ | 1466 MWe | 727 MWe |
| Net electric power output | $P_{net}$ | 1386 MWe | 600 MWe |
| Net plant efficiency | $h_{net}$ | 0.38 | 0.34 |

The stellarator possesses a natural helical divertor which occurs as a consequence of the existence of a magnetic separatrix bounding the region of closed nested flux lines. The area available for divertor plates placed in the vicinity of the natural divertor is quite large, reducing the particle and heat flux on it. This prolongs the lifetime of the divertor requiring less frequent replacement. The heat loads are on the order of several MW/m$^2$, well within current technology being developed for tokamaks. Similarly the same technology of particle pumping, such as pumped limiters, being developed for tokamaks can be applied here.

## G. Power Flow

Power flow diagrams for tandem mirrors and stellarators are shown in Figs. 5.G.1 and 5.G.2. Typical reactor power parameters are shown in Table 5.G.1. Both configurations absorb the neutron power in a blanket and shield, with the resulting thermal energy converted via one of several thermal power cycle options. In both, the fusion products slow down from their birth energy by colliding with the background plasma so that most of the plasma losses are as thermal ion and electron transport power or as radiation. In the stellarator, the divertor absorbs most of the plasma losses, and a standard thermal cycle will then be used.

In a tandem mirror reactor, the open field line geometry allows the option of directly converting the plasma losses to electricity via electrostatic direct converters. Because only about 20% of the fusion power is in charged particles in a D-T plasma, there is not always sufficient leverage to warrant the additional complexity of a direct-conversion system. In a D-$^3$He plasma, where >95% of the energy is in charged particles, there is a large incentive to use the demonstrated technology of direct electrostatic conversion, and it would almost certainly be used.

The input powers for the two types of devices differs considerably. The tandem mirror requires several varieties of power for sustaining the end cell configuration and possibly providing MHD stabilization. The stellarator requires startup heating and

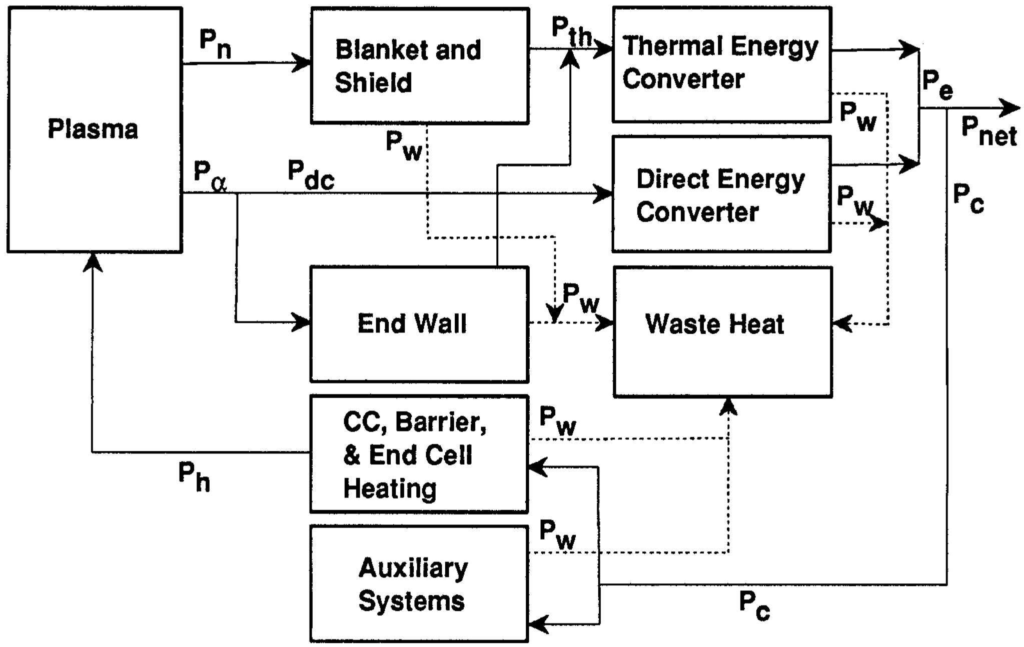


Fig. 5.G.1. Steady state tandem mirror power flow.

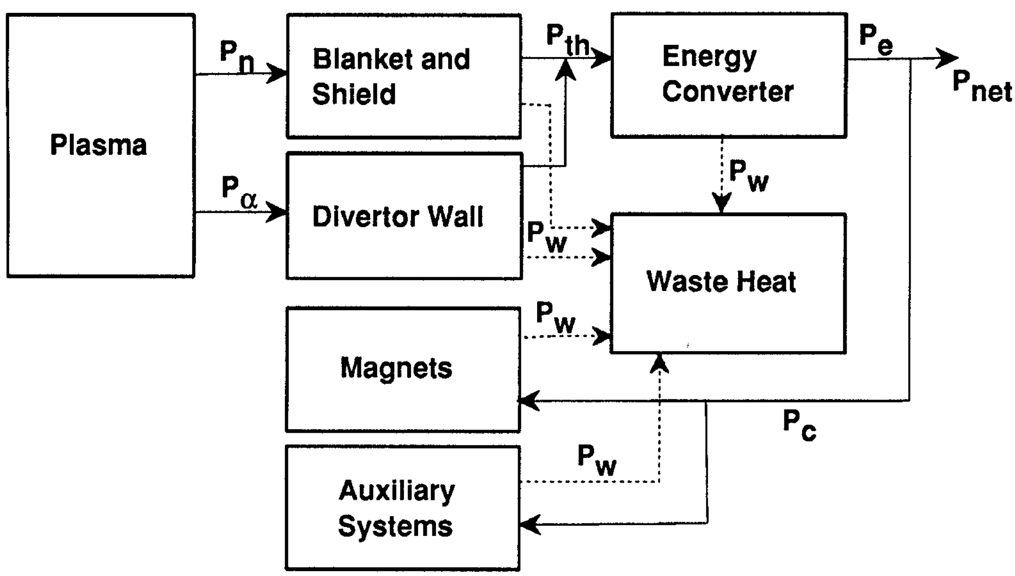


Fig. 5.G.2. Steady state stellarator power flow (not shown is the power for startup).

possibly impurity control. This power will be recirculated into the plasma and subsequently lost as transport power or radiation. Both configurations will require some auxiliary power. It has been assumed in these diagrams that some of the power will appear as waste heat and cannot be converted to a useful form.

**References for Section 5.G**


[1] H. Wobig, private communication, 1993.


## H. Maintenance

A comparison of the maintenance requirements for stellarator and tandem mirror power reactors is given in Table 5.H.1. Maintenance of blanket components in tandem mirrors and stellarators have a common element in that both require coils to be moved in order to get at the blanket segments. Here the commonality ends. The mirror central cell is linear and the blanket/shield is exactly concentric with the central cell coils. Once a central cell coil module is moved out of the assembly, the blanket segment can be easily slipped out and a new one inserted. Stellarators are closed toroidal devices where the plasma chamber cross section changes within a field period. Early versions of Helias reactor had coils sufficiently interlocked and could only be taken apart as whole periods consisting of six coils. Present versions have cases where four or six coils can be moved out radially. This unit can then be separated into two subunits of two or three coils. Blanket replacement thus entails removing blanket units by pieces rather than as a whole blanket unit.

Tandem mirrors also must have normal inserts in choke coils replaced periodically. These coils weigh ~200 tonnes and like the cc modules, must be moved out of the reactor assembly. Other end cell coils are lifetime components, but also have to be maintained in the event of a failure. They too can be moved out similarly to the choke coils.

The remaining maintenance functions such as dust cleaning, cutting and welding vacuum chamber components and coolant lines, maintaining NB and the beam dumps, and maintaining cryogenic components all have commonality between mirrors, stellarators and tokamaks. Many of these functions have been worked on in NET/ITER publications. The experience gained on ITER R&D will be applicable in these areas. Stellarator details would require an actual reactor study.

**Table 5.H.1**
**Comparison of the Maintenance Requirements for Stellarator and Tandem Mirror Power Reactors**

| | Stellarator | Tandem Mirror | |
|---|---|---|---|
| **Issue** | | **Central Cell** | **(Choke Coil)/End Cell** |
| **Scheduled Tasks and Operations Carried Out Annually by the In-Vessel Teleoperation System** | | | |
| Replace RF Launchers | N/A | N/A | N/A |
| Replace Divertor Modules | Annually? | N/A | As Needed |
| Inspect FW & Vacuum Vessel Components | Annually | Annually | Annually |
| Remove Sputtered Material | As Needed | As Needed | As Needed |
| **Unscheduled Tasks and Operations Carried Out by the In-Vessel Teleoperation System** | | | |
| Replacement of SC magnets | As Needed | As Needed | As Needed |
| Detect Vacuum Leaks | As Needed | As Needed | As Needed |
| Repair Vacuum Leaks | As Needed | As Needed | As Needed |
| Recover Debris | As Needed | As Needed | As Needed |
| **Tasks and Operations Carried Out Annually by Out-of-Vessel Systems** | | | |
| Blanket Handling | 1260 t every 8 y | 110 t every 8 y | As Needed |
| Reactor Module Handling | 2180 t | 140 t | (200/3.5 y )/470 t |
| Transport to Hot Cells | 2180 t | 140 t | (200)/470t |
| Welding, Cutting, Machining | As Needed | As Needed | As Needed |
| Positioning | As Needed | As Needed | As Needed |
| Dismounting,Assembling | As Needed | As Needed | As Needed |

a= mass of component, time required to replace, frequency of replacement
N/A = Not Applicable
t= tonnes

# CHAPTER 6

## Summary and Conclusions

After reviewing the physics and technology requirement to take today's tandem mirrors and stellarators to the state of commercial readiness, one would like a clear statement as to which of the two approaches represents the least risk. Unfortunately, this question does not have a simple answer. One is tempted to say that the much more modest extrapolation in technology required by tandem mirrors could override the uncertainty in the physics extrapolation. On the other hand, the physics extrapolations required of the stellarator are less than those for the tandem mirror but the technology requirements are nearly as challenging as for the tokamak.

The level of financial support for stellarators in Germany, Japan, and the US is much larger than the relatively small effort for tandem mirrors in Japan and Russia. In some European and Japanese circles, the stellarator is viewed as a legitimate backup for the tokamak. There is no tandem mirror constituency in Europe, and for this reason it will be difficult for the tandem mirror to receive an adequate review. The relative support for the stellarator (vs. the tokamak) in the US is smaller and less visible and only a small tandem mirror constituency remains. The demise of the mirror program in the US was dominated by the budgetary problems of the US fusion program, although some critics point to the failure of TMX-U to meet its design goals.

There is little doubt that, if the plasma physics issues for the tandem mirror were solved, the linear geometry and relatively simple magnetic coil sets should translate into a more reliable, and probably more economical electrical power producing unit. The lack of pulsed conditions (as is also the case in stellarators) and the modest neutron and particle fluxes to the first wall also contribute to the attractiveness of the tandem mirror. So the question boils down to the issue of physics extrapolation.

The level of physics extrapolation required to move from today's tandem mirror experiments to a prototype device is felt to be challenging, but feasible. The additional extrapolation to a power reactor is not as great as the step from current devices to a prototype device. Successful operation of thermal barriers at high central cell density and electron temperature needs demonstration before the US fusion program will invest in the tandem mirror approach.

Stellarator experiments have achieved very good energy confinement and plasma parameters. The key physics issues for stellarator development are the demonstration of reactor-relevant beta values and the control of impurity and fusion-ash levels. In addition, present scaling laws, which predict adequate confinement for reactor plasmas, need to be confirmed at densities, temperatures, and plasma dimensions expected in stellarator reactors.

Overall, the international stellarator program seems adequately funded (if W 7-X is built) to determine if it will be a viable competitor to the tokamak. The tandem mirror represents a high risk approach with a higher potential than the stellarator and therefore deserves a much greater level of effort around the world.

## Acknowledgment

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# APPENDIX

# Comparison of Tandem Mirrors and Stellarators

| | | | | | | |
|---|---|---|---|---|---|---|
| | ***Comparison of Tandem Mirror and Stellarator*** | | | | | |
| | *Tandem Mirror* | | | *Stellarator* | | |
| | | **Experiment** | **Current View** | | **Experiment** | **Current View** |
| **Parameter** | **Units** | **Best To Date** | **DT Reactor(a)** | | **Best To Date(c)** | **DT Reactor(b)** |
| | *Overall Reactor/Device* | | | | | |
| Fusion Power | MW(th) | None | 1290 | Fusion Power | None | 2900 |
| Net Electric Power | MW(e) | None | 600 | Net Elec. Power | None | |
| Plasma Power Gain | | None | 34 | Plasma Power Gain | | |
| Recirculating Power | MW(e) | None | 114 | Recirculating Power | | |
| Reactor Length(CC+EP) | m | | 98 | Reactor Circumference | | |
| | *Central Cell* | | | *Plasma Chamber* | | |
| CC Length | m | | 91 | Major Radius | 5.5 | 19.5 |
| Plasma Chamber Radius | m | | 0.54 | Plasma Chamber Radius | | |
| Plasma Radius | m | | 0.37 | <Plasma Radius> | 0.265 | 0.8 |
| Max. CC Mag. Field on Axis | T | | 3.08 | Chamb. Mag Field on Axis | 3 | 5 |
| <Ion density> | 10^14/cm3 | 0.1 | 3.96 | <Ion density> | | 2.8 |
| <Ion Temp> | keV | 5.6 | 25 | <Ion Temp> | | 7.5 |
| <Electron Temperature> | keV | 0.28 | 20 | <Electron Temperature> | | |
| Energy Confinement Time | s | | 2.5 | Energy Confinement Time | | |
| n wall loading | MW/m2 | None | 3.3 | n wall loading | None | |
| <beta> | % | 13 | 90 | <beta> | | 5.1 |
| Electron Confining Potential | keV | | 154 | Rotational Transform (0) | 0.84 | 0.84 |
| ntauT | 10^20 s-keV/m3 | | 250 | ntauT | | 52 |
| Fueling | | | Compact Toroid | Fueling | | |
| | *Edge Plasma in Scrape-Off Region* | | | *Divertor* | | |
| Ion Density | 10^18/m3 | | 1 | | | |
| Ion Temperature | <keV> | | 1 | | | |
| Ion Temp. at First Wall | <keV> | | 0.01 | Edge Ion Temp.- eV | | |
| Edge Plasma Thickness | <m> | | 0.17 | | | |
| Edge Plasma Volume | m^3 | | 40 | | | |
| Ave. FW Heat Flux | <MW/m^2> | | 0.028 | Heat Flux | | |
| Max. Halo Scraper Heat Flux | <MW/m^2> | | 1.1 | | | |
| | *Magnets* | | | | | |
| Type-CC | | S/C | S/C | Type | Normal | S/C |
| Conductor-CC | | | NbTi | Conductor | Cu | NbTi (1.8K) |
| Geometry-CC | | Solenoid | Solenoid | Geometry | Twisted | Twisted |
| Number of Coils | | | 24 | Number of Coils | | |
| Coil Radius-Id | m | | 1.73 | Coil Radius-Id | 2.28 | 7.8 |
| Max Field-CC | T | | 4.7 | Max Field | 6.1 | 10.7 |
| Stored Energy in CC | GJ | | 3.17 | Stored Energy | 0.6 | 74 |
| Type-Choke Coil | | S/C | S/C | | | |
| Conductor-Choke Coil | | | (Nb3Sn)Ti | | | |
| Max Field-Choke Coil | T | | 24(S/C+ Norm) | | | |
| Type-End Cell | | | Octopole | | | |
| Conductor-End Cell | | | Nb3Sn | | | |
| Max Field-End Cell | T | | 11 | | | |
| Mag. | GJ | | | Total Stored Energy | 0.6 | 74 |
| | *Central Cell Materials* | | | | | |
| Structural | | | HT-9 | Structural | | |
| Max Temp. of Structure | °C | | 525 | Max Temp. of Structure | | |
| neutron damage | dpa/FPY | None | 46 | neutron damage | None | |
| Helium Production | appm/FPY | None | 380 | Helium Production | None | |
| Neutron Multiplier | | None | None(Pb) | Neutron Multiplier | None | |
| n Energy Multiplication | | None | 1.36 | n Energy Multiplication | None | |

| | | | | | | |
|---|---|---|---|---|---|---|
| | *Coolant* | | | | | |
| Type | | | Li17Pb83 | Type | | |
| T(in)/T(out) | °C | | 350/500 | T(in)/T(out) | | |
| | *Breeder* | | | | | |
| Tritium Burn Rate | g/d | None | 221 | Tritium Burn Rate | None | |
| Type | | None | Li17Pb83 | Type | None | |
| Tritium Breeding Ratio | | None | 1.07 | Tritium Breeding Ratio | None | |
| Blanket T2 Inventory | g | None | 6 | Blanket T2 Inventory | None | |
| | *External Power-Steady State* | | | *External Power- Start Up* | | |
| Hot Ion NBI Beams | MWe/keV | | 12/412 | NBI | | |
| RF Stabilizing RF | MWe/MHz | | 38/23 | ICRF | | |
| Warm-elec. plug ECRF | MWe/GHz | | 4/70 | ECRF | | |
| Hot-elec. barrier ECRF | MWe/GHz | | 23/110 | | | |
| | *Direct Convertor* | | | | | |
| Type | | | Gridless | | | |
| Power From CC | MWth | | 131 | | | |
| Heat Flux | MW/m2 | | 3.7 | | | |
| Voltage-Direct Convertor | kV | | 75/150 | | | |
| Convertor | % | | 63 | | | |
| Direct Converter | MW(e) | | 82 | | | |
| Coolant | | | H2O | | | |
| | *Plug Region* | | | | | |
| Type | | | Axisymmetric | | | |
| Magnetic Field at Plug Peak | T | | 9.7 | | | |
| Beta at Potential Peak | % | | 10 | | | |
| Plug | <keV> | | 277 | | | |
| Sloshing Ion Density | | | 2 | | | |
| Mirror ratio | | | 1.65 | | | |
| Potential | <keV> | | 148 | | | |
| | *Barrier Pumping* | | | | | |
| Mechanism | | | Pondero-motive drift pumping | | | |
| Ion Traping Current | A | | 447 | | | |
| Hot Elect. Temp. | keV | | 297 | | | |
| Total Ion Density (midplain) | 10^18/m^3 | | 18 | | | |
| Max Choke Field | T | | 24 | | | |
| Midplain Magnetic Field | T | | 2 | | | |
| Barrier Electrostatic Potential | keV | | 126 | | | |
| Neutral Beam Injector (2) | kWe/keV | - | 412/3.5 | | | |
| RF system | MWe/MHz | | 50/46 | | | |
| | | | | | | |
| (a)= MINIMARS | | | | | | |
| (b)= HSR (Boston) | | | | | | |
| (c)= W 7-X | | | | | | |