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CORIANDER

Comparison of Relevant Issues and Nuclear Development for Fusion Energy Research

F. Arendt, G. Böhme, L. A. El-Guebaly, G. A. Emmert, B. Haferkamp, W. Heinz, E. Hutter, R. Klingelhöfer, P. Komarek, W. Kramer, M. Küchle, G. L. Kulcinski, M. Kuntze, E. M. Larsen, G. Leppelmeier, S. Malang, W. Maurer, C. W. Maynard, R. A. Müller, D. Perinic, J. F. Santarius, M. E. Sawan, J. E. Scharer, I. N. Sviatoslavsky, W. F. Vogelsang, L. J. Wittenberg

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CORIANDER: COMPARISON OF RELEVANT ISSUES AND NUCLEAR DEVELOPMENT FOR FUSION ENERGY RESEARCH

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The authors of this report wish to dedicate it to the memory of Professor Dr. Werner Heinz. To a large degree, the idea for this study came from Prof. Heinz and it progressed rapidly under his leadership. After several years of design study of small, inexpensive tandem mirror test facilities, TASKA and TASKA-M, he felt it was important to address the larger picture of how such a device would facilitate the movement toward a fusion economy. The combined efforts of scientists from KfK, Fusion Power Associates and Lawrence Livermore National Laboratory have resulted in the present document, which, we hope, clearly spells out why Werner Heinz thought it was so important to investigate an alternate way to the ultimate goal of a safe, clean energy source. We, and all of his colleagues, will miss his enthusiasm for and dedication to scientific progress. Perhaps in the years to come, all will see the wisdom of his vision.

The authors

Abstract

This study compares two strategies to be followed on the way to a tokamak demonstration reactor (DEMO). The first is the present European conception of building an integrated physics and technology machine (NET) between the present large physics experiments (such as JET) and DEMO. In the Alternate Plan, NET would be replaced by a combination of an advanced physics tokamak and a mirror based dedicated fusion technology device ("TASKA-class"). It appears that the Alternate Plan could provide the required physics and most of the engineering data for building DEMO with less risk, in a shorter time, and perhaps less cost than the present route. While it is highly desirable to increase the neutron fluence for material studies in either strategy, the Alternate Plan may require extrapolations in the combination of blanket geometry effects and high neutron fluence.

CORIANDER: Vergleichsstrategien für die Fusionstechnologie auf dem Wege zum Tokamak-Demonstrationsreaktor

Zusammenfassung

Zwei Strategien auf dem Wege zu einem Tokamak-Demonstrationsreaktor (DEMO) wurden verglichen. Die erste ist der gegenwärtige europäische Plan, zwischen den laufenden großen Physikexperimenten wie JET und dem DEMO eine integrierte Tokamakanlage NET mit physikalischer und technologischer Aufgabenstellung zu bauen. In einem Alternativplan würde NET durch einen physikalisch ausgerichteten Tokamak mit gezündetem Plasma und eine spezielle Technologieanlage auf der Basis des Spiegelprinzips ("TASKA-Klasse") ersetzt. Es ist wahrscheinlich, daß der Alternativplan die erforderlichen physikalischen und fast alle technologischen Entwicklungen für DEMO mit geringerem Risiko, in kürzerer Zeit und eventuell mit geringeren Kosten bereitstellt. Während in beiden Strategien die Neutronenfluenz für Materialbestrahlungen erhöht werden müßte, erfordert der Alternativplan Extrapolationen für solche Blankettechnologien, bei denen Größe und hohe Neutronenfluenz zusammenwirken.

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1. INTRODUCTION

The recent progress in plasma physics has, for the first time, allowed devices with a burning DT-plasma and long burn times to be considered. As a result, worldwide studies are proceeding to define the next large step in magnetic confinement experiments for nuclear fusion. The INTOR design(1) is the best known example of a device of this class and is considered the major single experiment in the tokamak program between the current generation of large tokamaks (TFTR, JET, JT60, T15) and the demonstration reactors (DEMOs) expected after the turn of the century. Thus, the role of INTOR was defined by the physics and technology prerequisites for the design and construction of the DEMOs. A design effort with a similar mission has been carried out in USA for the Fusion Engineering Device (FED)(2) and has been started in Europe for the Next European Torus (NET).(3) The NET program, together with the general European fusion program, is expected to provide adequate information to design a tokamak DEMO.

The rather high costs and remaining uncertainties for a NET/INTOR device have initiated new discussions, worldwide, about the best strategy. An alternative to the NET/INTOR strategy could be to restrict the objectives for the next step tokamak to the physics of a burning plasma and the associated plasma engineering issues. An example of this approach is the Tokamak Fusion Core Experiment (TFCX)(4) which was proposed in the USA. The question then remains as to how to proceed with the technology developments needed for a DEMO, especially the nuclear technologies. Certainly for fundamental development, present experiments such as out-of-pile test stands, simulation loops, fission reactors, and plasma experiments are useful and necessary. But ultimately there remains the need for an integrated technology test facility with a thermonuclear neutron source which provides a sufficient test volume, neutron flux and fluence, and other relevant engineering data.

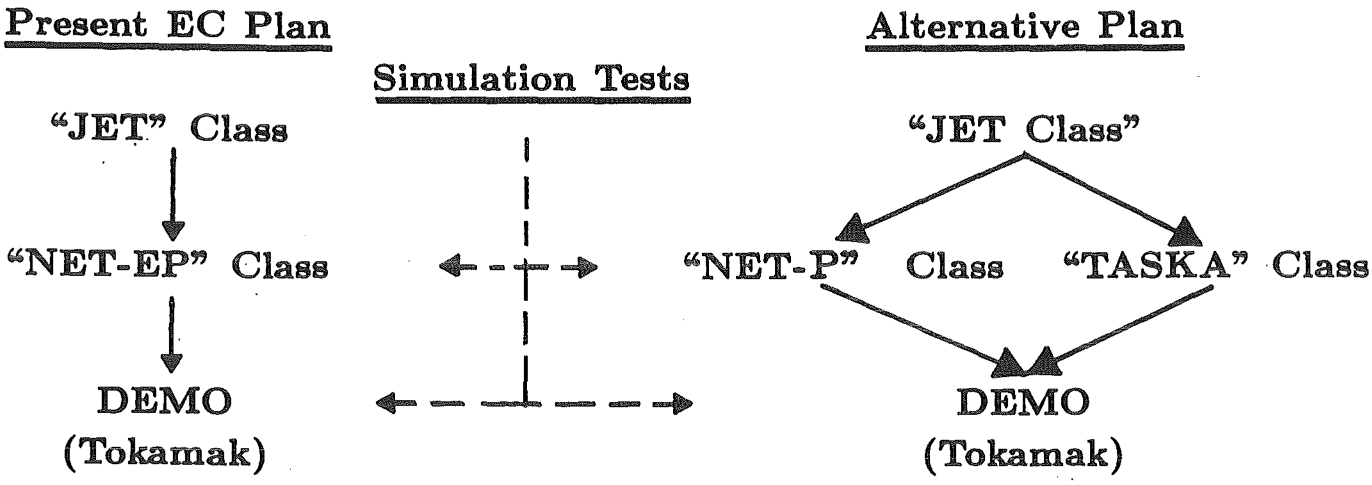
Recently, such fusion technology test facilities based on the mirror plasma confinement approach, the TASKA,(5) TASKA-M,(6) TDF,(7) and MFTF-B Upgrade(8) designs, have been studied. It was found that such mirror-based facilities can offer the most economical solution for integral technology test devices because they can be built in small units of less than 100 MW of thermonuclear power. Since the plasma physics principles need not be reactor relevant, the energy amplification of the thermonuclear plasma can be far below unity (highly driven machines) which allows the neutron flux, reactor costs and test objectives to be optimized.

In order to quantify the advantages or disadvantages of the various programmatic approaches, it is useful to compare two strategies for the development of a tokamak DEMO reactor, as shown in Fig. 1-1. In one strategy, the present large physics experiments (e.g., JET) are followed by just one tokamak device of the INTOR/NET class denoted "NET-EP." This device, together with accompanying simulation and test loops on the technology side and special smaller experiments on the physics side, would provide the necessary data for the design and construction of the tokamak DEMO. In the other strategy, the present large physics experiments are followed by a tokamak experiment called "NET-P" which mainly has physics and plasma engineering goals. In addition, a technology test facility of the "TASKA class," based on tandem mirror confinement, would focus on technology issues. For special questions, simulation experiments and smaller physics experiments would complement this strategy and complete the body of knowledge required for the construction of the tokamak DEMO.

A "TASKA class" technology test facility can be built using existing technologies with only moderate component extrapolations. Such a tandem mirror facility operates steady state while a "NET-P class" facility would be pulsed. The experience with both operational modes is valuable with respect to the DEMO because it cannot be determined today whether the DEMO is a steady state or pulsed reactor. A division into two facilities avoids the risk of the lower availability that would be associated with a "NET-EP" device.

It was the objective of this study, called CORIANDER (Comparison of Relevant Issues and Nuclear Development in Fusion Energy Research), to identify the advantages and disadvantages of the two approaches. The study was initiated in January 1984.

The first task was to set "typical" parameters for the tokamak DEMO in order to know how well the NET-P and TASKA class facilities could meet the DEMO requirements. It must be emphasized that the projected tokamak DEMO values only represent the best guess of current reactor designers or stem from the STARFIRE-DEMO[9] design and are not meant to be precise numbers. With these values, the ability of the physics machine ("NET-P class") and the technology machine ("TASKA class") to meet those requirements was tabulated. This required a thorough examination of several "self-consistent" designs, not just



CLASS	PURPOSE	EXAMPLE
"JET"	Tokamak Breakeven Physics	JET, TFTR, JT-60, T-15
"NET-P"	Tokamak Ignition Physics	TFCX
"NET-EP"	Engineering and Physics in One Tokamak	INTOR, NET
"TASKA"	Mirror–Based Technology Test Facility	TASKA, TASKA-M, TDF
Simulation	Non–Fusion Tests	Fission Reactors, TSTA, LCT, FMIT

Fig. 1-1. Basic strategies of the CORIANDER study for fusion reactor development.

a wish list for each device. For the NET-P, the nominal superconducting TFCX design[4] was used. When specific parameters from this facility were not available, it was so noted.

The parameters for the "TASKA class" device were obtained from the previously mentioned studies.[5-7] None of these tandem mirror facilities was designed with the objective of supporting the tokamak DEMO technology needs.. It is likely that such a "TASKA class" device optimized for this objective might include features from all the current studies, and therefore the most appropriate numbers or range of numbers were used.

Once the DEMO requirements and the output from "NET-P" and "TASKA" level devices was obtained, it was necessary to consider the role of the non-fusion facilities that exist around the world. A great deal of restraint was required to keep from saying "device XYZ could be modified to provide the answers required." While that may in fact be true, it was felt that many uncertainties about facility lifetime, modification costs, and increased operating costs make such an extrapolation highly speculative. Therefore, we only based our conclusions on what the present, non-fusion facilities could provide. The major consideration in this area was determining the value of isolated tests. For example, if one could only provide corrosion data without magnetic fields or without the appropriate irradiation environment, then the facility was not considered to adequately provide the required information for a DEMO.

With the above information in hand, we proceeded to derive conclusions about the adequacy of the "Alternate Plan." Such conclusions take many factors into account other than just trying to provide the exact temperature, magnetic field, stress level, fluence, etc. The scaleability, the number of data points, the time in which those data points are accumulated, and many other features are important. It was concluded that the "Alternate Plan" is a viable strategy and may represent the route of lower risk and cost to a tokamak DEMO, provided simulation experiments for a few remaining issues are approved.

In addition, it is clear that a "TASKA class" technology test device would provide a rather complete engineering information base if the development of a tandem mirror DEMO were pursued in the future. Due to the simpler scaleability of tandem mirrors, such a test device together with physics and plasma engineering verification experiments could yield the information needed for a tandem mirror DEMO.

Chapter 3 gives an overview of the conceptual designs (TFCX, TASKA, TDF, TASKA-M) used as the basis for this study. The physics basis of the tandem mirror test devices is discussed more fully in Appendix 3A. Plasma engineering considerations, such as heating and impurity control, are discussed in Chapter 4. Testing of various blanket modules is considered in Chapter 5. Testing of materials using small samples in a neutron environment is discussed in Chapter 6. Chapter 7 covers the areas of tritium handling, separation, and the fuel cycle. The vacuum and exhaust systems are dealt with in Chapter 8. Magnets are discussed in Chapter 9. Instrumentation and control is treated in Chapter 10, and maintenance and operations in Chapter 11. Finally, safety issues are discussed in Chapter 12. The major conclusions of this study are given in Chapter 2 .

References for Chapter 1

1. International Tokamak Reactor, Phase Two A, (IAEA STI/PUB/638, Vienna, 1983).

2. C.A. Flanagan, D. Steiner and G.E. Smith, eds., "Fusion Engineering Device Description," ORNL Report ORNL/TM-7948 (1981).

3. Plasma Physics and Controlled Nuclear Fusion Research 1984, (IAEA, Vienna, 1985).

4. "TFCX Preconceptual Design Report," Princeton Plasma Physics Laboratory Report F-Axxx-8407-006 (1984).

5. B. Badger et al., "TASKA, A Tandem Mirror Fusion Engineering Facility," Kernforschungszentrum, Karlsruhe Report KfK-3311 (1982).

6. B. Badger et al., "TASKA-M, A Low Cost, Near Term Tandem Mirror Device for Fusion Technology Testing," Kernforschungszentrum Karlsruhe Report KfK-3680 (1983).

7. J.N. Doggett et al., "A Fusion Technology Demonstration Facility (TDF)," Lawrence Livermore National Laboratory Report UCRL-90824 (1984).

8. K.I. Thomassen et al., "An Upgrade of MFTF-B for Fusion Technology," Lawrence Livermore National Laboratory Report UCRL-90825 (1984).

9. M. Abdou et al., "A Demonstration Tokamak Power Plant Study (DEMO)," Argonne National Laboratory Report ANL/FPP/82-1 (1982).

2. CONCLUSIONS

2.1 Introduction

This study has considered an "alternate" approach to obtaining the data base required for building a tokamak demonstration reactor (DEMO). In this alternate approach the present generation of physics machines (JET, TFTR, T-15, JT-60) is followed by a larger tokamak physics machine (called a NET-P class device) which achieves ignition and perhaps long pulse operation with a deuterium-tritium plasma and a respectable neutron wall loading, but with a low duty factor and low neutron fluence. In parallel with this machine is a tandem mirror based technology test device (called TASKA class device), which would provide high neutron fluence operation with a much smaller plasma volume and fusion power level. This machine would provide extended neutron testing of blanket modules, material test samples, neutral beam and RF heating technology, magnets, tritium handling technology, and other components in an integrated facility. Furthermore, fission reactor facilities and simulation test stands would provide additional data.

Even though a study as this is not all-inclusive, some important conclusions can be drawn. Overall, it appears that the "Alternate Plan" could provide the required physics and most of the engineering data for building a DEMO with less risk, in a shorter time, and with perhaps less costs than the present approach of building a single large tokamak aimed at both physics and engineering testing. This conclusion is valid in an overall sense, but some drawbacks remain. In particular, neither the present approach nor the alternate plan will provide the full level of materials damage data ($\sim$150 dpa) during their envisioned lifetime. If a combination of field profile and size effects becomes essential (e.g. in liquid metal thermohydraulics), a large simulation test stand may be required in the alternate plan. The detailed conclusions with respect to the various physics and technology aspects are given in the following sections.

2.2 Plasma Engineering

The combination of NET-P and a TASKA-class machine can provide the heating data base required for the DEMO. The physics, electromagnetic transients, and surface load conditions are tested in NET-P. The TASKA-class machine provides data on neutron fluence effects for RF antennae, neutral beam sources, and other components under similar power and surface loading conditions, but at smaller dimensions for antennae.

Information concerning neutron fluence effects on high heat flux components is provided by the neutral beam and central cell inserts in a TASKA-class machine. NET-P would provide data on electromagnetic transients and surface conditions, but only at low neutron fluence. Not tested in this alternate plan are possible synergistic effects, such as the combination of neutron irradiation and electromagnetic effects.

2.3 Blanket Testing

Fusion facilities are mostly required for testing liquid metal blankets as compared to solid breeders. The main difficulty with non-nuclear test facilities is the lack of neutrons, which effects the rate of corrosion and the ability of metals to withstand stresses under thermal, magnetic, and static forces. Fission reactors can provide a reasonable nuclear environment, but not over a large enough volume or the correct geometry and cannot simulate the required magnetic fields for the required testing time (1,000 - 10,000 hours).

Since the magnetic field effects on solid breeders are not important, only the radiation damage, thermal profiles, and tritium production rates are critical parameters to test. This could be done in a fission reactor facility, but the geometry and volume are not well-suited to realistic solid breeder blankets. Nevertheless, more can be done with fission reactor testing of solid breeder blankets than with liquid metal blankets.

Many blanket and shield conditions of a tokamak DEMO can be properly simulated in a TASKA-class facility. Meaningful integrated blanket testing can only be performed in such facilities where the combined nuclear, thermal, chemical and magnetic field environment is duplicated. Because of the small size of the blanket in a TASKA-class facility proper scaling has to be used to interpret test results for a tokamak DEMO blanket. The issue of blanket response to plasma disruptions and size scaling can be tested in a tokamak physics test facility of the NET-P class.

2.4 Materials Testing

It is concluded that a great deal of non-fusion testing of metals can be done in auxiliary facilities. However, the restricted temperature range (e.g., tests can only be conducted above 375°C in fast reactors) and small individual test volumes, along with serious neutron energy spectral differences, make complete testing of materials in fission reactors impossible. Of particular importance is the drastic difference in gaseous production rates and the production of solid transmutants which might alter the chemistry of a metal or alloy.

Fission reactors can be helpful in a screening capacity but it is concluded that true fusion neutron facilities, such as a TASKA, are required to provide more realistic design data for the DEMO. However, neither the present approach nor the alternate plan will provide the full load of materials damage for DEMO ($\sim$ 150 dpa) during their envisioned operation time. Therefore, it is necessary to extrapolate, by a factor of 2 - 3, the damage data obtained from presently designed test facilities (e.g. TASKA) to desired DEMO operating conditions. Such an extrapolation is within the realm of acceptability at this stage of fusion development. A TASKA-class machine also provides the necessary experience required for the operation of superconducting magnets in a radiation environment.

2.5 Tritium Cycle

All tritium subsytems and problems, except one, needed for the DEMO can be extrapolated with low risk from their performance in a TASKA-class machine. The one exception to this is that the TASKA type facilities do not simulate well the plasma-wall interaction in the DEMO.

Before a TASKA-class machine can be reliably designed, tritium breeder characterization studies must be initiated using fission reactors, and certain large equipment items, such as valves, need to be tested in non-nuclear, TSTA type facilities.

2.6 Vacuum and Exhaust

Because of present uncertainties in the physics and in the requirements imposed on a tokamak DEMO exhaust system, one of the main tasks of a NET-P device will be to provide non-fluence related experience in this area.

A TASKA-class device will provide sufficient fusion neutron fluence as well as sufficient integrated particle and heat flux on target plates. However, the energy spectrum of the charged and neutral particles will be considerably different from that in a tokamak.

The experience of operating NET-P and a TASKA-class machine will be a sound basis for designing the DEMO vacuum system.

2.7 Magnets

All non-nuclear technology requirements for the superconducting magnets of DEMO are developed in dedicated test stands such as LCT. Limited information can be obtained on magnets in fission reactors and irradiation test facilities. Full integrated magnet tests can be better performed in fusion devices such as a TASKA or NET-P class device. A combination of mechanical and magnetic performance information from a NET-P class device along with magnet material damage data from a TASKA-class device suffices to completely provide the required tokamak DEMO magnet data base.

2.8 Instrumentation and Control

There have been a few studies addressing the control question, mostly in a general way. The consensus is that the control task may be difficult, but still manageable. The specifics of control would be developed on NET-P and other tokamaks, while development and test of control tools could take place on a TASKA-class machine. One study addresses the problem of plasma diagnostics in a reactor environment. Simple estimates show that only the very simplest of sensors can be used in the first wall region, and then only with extensive effort, while with shielding and collimation, more sophisticated instruments can be used. The physical measurement and the operational performance of all proposed instruments can be tested on a TASKA-class machine. Lastly, a rough estimate of the data rate (in Mbytes/day) for a DC DEMO or TASKA-class machine is comparable to that planned for current large machines such as JET, TFTR, and MFTF-B, indicating that the size and power of current data systems are sufficient. DC operation , however, will require a very different architecture.

2.9 Maintenance and Operation

Both NET-P and a TASKA-class machine, with the help of full-scale realistic simulation tests, will satisfy the maintenance needs of a tokamak DEMO. The principal question remaining is whether the maintenance tasks can be performed on a time scale consistent with economic and availability requirements.

There is a presumption that there will be scheduled shutdown periods for first wall and other component replacement. However, in the studies to date there are no manning tables for operating personnel requirements, no procedures for startup, and no safety procedures. In brief, fusion reactor systems are not sufficiently well defined at this point to allow a significant statement regarding operations, but this task can be met by operating the NET-P and TASKA-class devices.

2.10 Safety

Much safety related information can only come from simulation experiments. This is especially true for those situations in which a direct test would put the plant at a significant risk of damage. The combination of the NET-P and TASKA-class devices provides design and operational experience plus confirmation of operation in a DEMO environment, i.e. in the presence of radiation fields, magnetic fields, and appropriate heat fluxes and temperatures.

3. DESIGN FEATURES OF NEAR TERM FUSION TEST FACILITIES

3.1 Introduction

In this chapter we review the basic design features and parameters of the TFCX design and the three tandem mirror test facilities, TDF, TASKA and TASKA-M. TFCX[1] is used as a model for the NET-P physics facility. The tandem mirror designs are used as possibilities for a tandem mirror test facility, dubbed a "TASKA" class facility, to provide the technological data, especially nuclear effects, necessary to commit to the construction of tokamak DEMO. These four preconceptual design studies are the basis for the "alternate scenario" presented in Chapter 2, and considered in more detail in the remaining parts of this report. We present an overview of the TFCX design in Section 3.2, and of the three tandem mirror designs in Section 3.3.

3.2 The TFCX Design

The basic mission of the TFCX facility is to achieve plasma ignition and self-sustaining equilibrium burn. It is to be a facility in which all the remaining physics questions associated with the tokamak confinement concept as a fusion device are addressed and resolved at one time. Thus, we consider TFCX to be a reasonable model to indicate the parameters of a "NET-P" class machine. The TFCX facility has a burning DT plasma of reactor-like parameters, but is deliberately a low neutron fluence facility to minimize costs and maximize flexibility in order to achieve the physics objectives. Consequently, it will not provide the extended operation under burning conditions necessary to obtain the relevant neutron fluence data for materials, blanket modules, first wall, RF antennae, impurity control components, and other elements of a fusion reactor.

Four different design options were considered for TFCX. These relate to the design of the toroidal magnet system and are: (1) a nominal performance superconducting TF system, (2) a high performance superconducting option, (3) a nominal performance normal-conducting (copper) option, and (4) a high performance copper option. The main difference between the nominal and high performance superconducting options is the peak nuclear heating in the coil. The nominal option uses a design value of 1 mW/cm^3, whereas the high performance option uses 50 mW/cm^3. This heat flux for the high performance option is tractable only for low duty factor ($\simeq$ 3%) operation, whereas the nominal performance option is typical of power reactor values. The high heat load for the high performance option allows the shielding between the plasma and the magnet to be reduced. This allows higher toroidal magnetic field on axis and reduces the overall size and capital cost of the machine. The copper options are also smaller compared with the superconducting options. They differ primarily by the mechanical design and materials used in the TF coils. In this study we have used the nominal performance superconducting option as the model for NET-P, since that is most prototypical of a fusion reactor. This is especially important in the "alternate scenario," where experience with superconducting TF magnet systems typical of a DEMO would have to be provided by the physics machine, NET-P, if the technology test machine were a tandem mirror. Unless otherwise specified, the remaining discussion of TFCX in this chapter refers to the nominal performance superconducting option.

3.2.1 Main Parameters of TFCX

The overall parameters of TFCX are given in Table 3.2-1 and the physics parameters are given in Table 3.2-2. Note that the fusion power (267 MW) is substantial. The wall loading (0.7 MW/m^2) is the average value at the plasma edge; the wall loading at the wall itself is somewhat less. The pulse length is in the range 250-600 s, of which the burn phase is 150-500 s. The length of the burn phase is determined by the ability of the impurity control system to remove helium "ash" and impurities and thereby determine the plasma reactivity. The design objective is to achieve a burn of sufficient duration such that the plasma density, temperature, and current density have evolved to their steady-state profiles. A schematic of the machine is shown in Fig. 3.2-1.

There is no provision for electrical power generation or tritium breeding in TFCX. The tritium consumed by the plasma or lost to the environment will have to be purchased. Because of the low duty factor, this cost is not large. The accumulated neutron fluence in the first wall is only 5 x 10^{-3} MW-yr/m^2 over the design life of the facility. The accumulated burn time (2 x 10^5 s) assumes an average burn time much less than the maximum burn time given in Table 3.2-1.

The plasma configuration is chosen to obtain an adequate value for beta in the first stability regime using a conventional poloidal magnet set. The ignition margin is obtained using Mirnov scaling for the energy confinement time; this scaling is more conservative than INTOR or neo-Alcator scaling in this region of parameter space.

Table 3.2-1. Main Parameters of TFCX

Fusion power	267 MW
Neutron wall load (plasma edge)	0.7 MW/m^2
Major radius	4.1 m
Minor radius	1.5 m
Aspect ratio	2.7
Magnetic field on axis	3.7 T
Plasma current	11 MA
Lower hybrid power	32 MW
ICRF power	31 MW
Pulse length	250-600 s
Maximum burn duration	150-500 s
Duty factor	3%
Accumulated burn time	2 x 10^5 s
Number of pulses - hydrogen	10^5
- DT	10^4

Table 3.2-2. Plasma Parameters of TFCX

Elongation	1.6
Triangularity	0.3
Beta	5.5%
MHD safety factor - edge	2.4
- center	1.0
Average ion temperature	10 keV
Average fuel ion density	6.3 x 10^{13} cm^{-3}
Ignition parameter (Mirnov)	1.5
Design effective charge (Z_{eff})	< 1.5
Helium concentration	5%

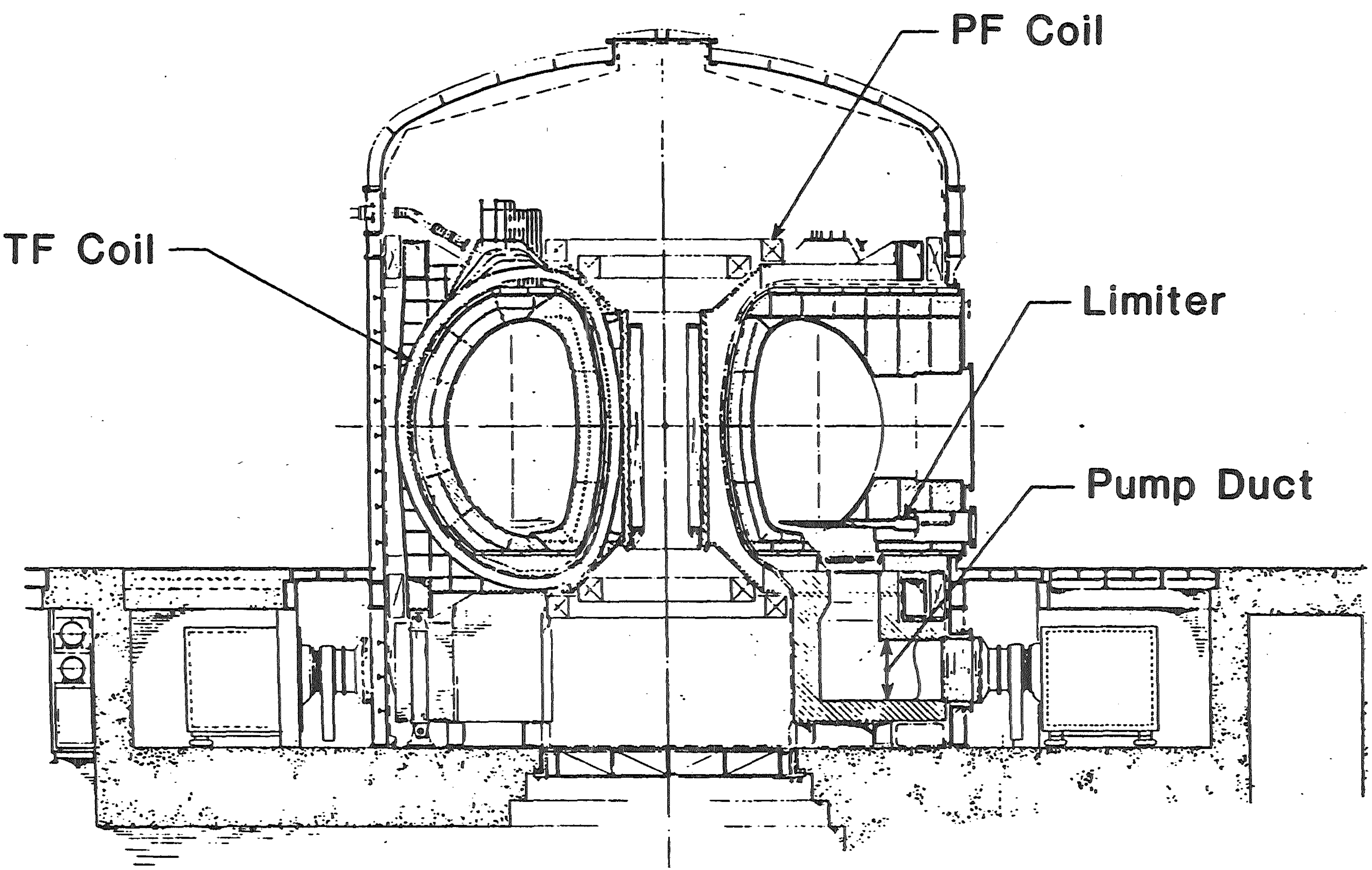


Fig. 3.2-1 TFCX nominal superconducting option--elevation view.

3.2.2 Discharge Cycle

A single plasma discharge is composed of four phases. During the first phase (≈ 50 s) the plasma is formed and the plasma current is brought to its full value of 11 MA using lower hybrid current drive to ramp up the current. This is done at low density (< 10^{13} cm^{-3}) and temperature to maximize the current drive efficiency and enhance the rate of evolution of the current density towards its steady-state profile. This also saves volt-seconds from the poloidal field set for the burn phase. In the second phase (≈ 10 s) the density is increased to its final value and ICRF heating is applied to augment the lower hybrid system in heating the plasma to ignition. During the third phase (≈ 150-500 s) the burn is maintained while the profiles evolve towards steady-state. The final phase (≈ 50 s) is shutdown; the ignition is terminated and the plasma current is ramped down with the aid of the lower hybrid system. The auxiliary systems (e.g., cooling) have been sized to repeat this cycle with a duty factor of 3%.

3.2.3 RF Power Systems

The lower hybrid coupler system is a phased waveguide-array operating at a frequency of about 2.7 GHz. There are 32 waveguides arranged in four groups of eight. The waveguides are 1.6 cm by 10 cm, and are located on the outboard, horizontal midplane of the torus; the power density is about 5 kW/cm^2. The waveguides are driven by klystrons; the total power transmitted by a single coupler system is about 4 MW.

The ICRF heating system operates at about 60 MHz. Several types of launchers have been considered. The best choice appears to be a ridge-loaded waveguide. They are also located on the outboard, horizontal midplane of the torus, where access is good. The power density is about 1 kW/cm^2. Six such waveguides, each with a power capability of 6 MW, could provide the required ICRF power. The associated power generation requirement is within the capabilities of present commercially available tetrode tubes.

3.2.4 Impurity Control

The primary choice for impurity control is the pumped limiter; a poloidal divertor is the backup option. The pumped limiter is the blade type and is located at the bottom of the discharge chamber. It is capable of being extracted between the TF magnets for maintenance. The total heat load to the limiter is 34 MW with a total ion current of 2.5 x 10^{23} s^{-1}. The charge exchange neutral current to the limiter is 1.9 x 10^{23} s^{-1} with an average particle energy of 280 eV. The peak heat flux has been estimated to be 260 W/cm^2. Both graphite and beryllium tiles on a water-cooled copper substrate have been considered for the design of the limiter blade. The estimated erosion lifetimes for graphite are 1 to 3 years depending on the temperature limit of the graphite, and 0.1 to 0.5 years for beryllium depending on whether the melt layer stays intact during a disruption.

The pumping for the pumped limiter is provided by 8 cryosorption pumps located below the torus.

3.2.5 Magnet Systems

The nominal performance superconducting TF magnet option of TFCX utilizes a forced-flow cooled Nb_3Sn conductor based on the Westinghouse magnet for the Large Coil Task experiment. This conductor choice is based on the higher current density allowed with Nb_3Sn and its greater tolerance for nuclear heating. These factors allow the radial build of the coil to be reduced, thereby resulting in a smaller major radius of the machine. Some of the TF magnet parameters are shown in Table 3.2-3. The magnets are shielded on the inboard side by 63 cm of stainless steel cooled by water; the volume fraction of the water cooling passages is 20%. The peak nuclear heating in the TF conductor is 1.3 mW/cm^3. The end-of-life dose to the electrical insulator is 2 x 10^7 rads.

Table 3.2-3. TF Magnet Parameters for the Nominal Performance Superconducting Option

Magnetic field at the TF coil	10 T
Number of TF coils	16
Winding bore size	5.2 m x 7.2 m
Conductor	Nb_3Sn
Stabilizer	copper
Total ampere-turns	76 MAT
Conductor current	20 kA
Winding current density	3.3 kA/cm^2
Overall current density	2 kA/cm^2
Stored magnetic energy per coil	440 MJ
Peak nuclear heating in the conductor	1.3 mW/cm^3
End-of-life dose to insulator	2 x 10^7 rads

The poloidal field (PF) system consists of the ohmic heating (OH) solenoid and the 6 equilibrium field (EF) coils. They are also superconducting using basically the same Nb_3Sn conductor as the TF system. The peak fields are limited to 8 T in the PF system. The OH solenoid provides the volt-s needed to sustain the plasma current during the burn. The EF coils provide the necessary vertical field for radial equilibrium.

3.2.6 First Wall/Shield

The first wall consists of an inboard and a outboard region. The inboard region consists of graphite tiles mechanically attached to the inboard shield surface. They are cooled by radiation to the water cooled shield and to the actively cooled outboard first wall. The design surface heat load to the inboard region of the first wall is 76 W/cm^2; the peak equilibrium temperature

Table 3.2-4. Estimated Direct Costs for TFCX Nominal Performance Superconducting Option

Category		M$ (1984)
Energy and particle removal		48
First wall assembly		37
Vacuum vessel assembly		167
Shielding system		60
TF magnet system		189
PF magnet system		139
Tokamak structure		12
Remote maintenance		22
Diagnostics		41
Lower hybrid RF system		128
ICRF system		98
Electrical power system		81
Instrumentation and control		56
Water cooling system		10
Cryogenic systems		19
Fueling systems		15
Vacuum pumping systems		22
Buildings and facilities		208
Miscellaneous		17
	Total	1369

of the graphite tiles is 1200°C. The outboard region consists of water cooled stainless steel panels. The panels have a design surface heat flux capability of 19 W/cm^2, and a maximum temperature of 220°C. A schematic of the vacuum vessel/shield configuration is shown in Fig. 3.2-2.

3.2.7 Costs

The estimated direct costs for the nominal performance superconducting option are given in Table 3.2-4. Any credits for existing facilities at possible sites have not been included in this table. The total direct cost is $1369 M. The costs are based on an 8 year design and construction schedule. The annual operating cost during the ignited phase of operation is limited to about $150 M for experimental operations, personnel, and electrical power.

3.3 "TASKA Class" Tandem Mirror Test Facilities

3.3.1 Introduction

In this section we discuss the three tandem mirror test facilities used as the basis of this study and point out similarities and differences in these designs. They are TASKA,(2) TDF,(3) and TASKA-M.(4) The question of how well these designs satisfy physics criteria is treated in Appendix 3.A.

3.3.2 Physics, Heating, and Fueling Features

The essential physics feature of all three tandem mirror designs under discussion is that the fusion power produced is less than the total input power required to generate the plasma. This driven operating mode allows the efficient creation of a strong neutron source in a moderate volume. Even though each design relies on this key element, the designs differ considerably in approach and details. A review of their physics basis is given in Appendix 3A.

The earliest design, TASKA,(2) began with the tandem mirror reactor configuration as it was then envisioned, scaling the size to the point where the neutral beams (NB) pumping the thermal barriers totally fueled and partially heated the central cell. The central cell plasma was close to equilibrium and was confined by electrostatic potentials generated in the end cells, in contrast to the other two designs. The magnetic field and electrostatic potential axial profiles are shown in Fig. 3.3-1a. Other important features were that ion cyclotron range of frequency (ICRF) heating was used to minimize total central cell heating power, negative-ion source neutral beams were needed for the end plugs, and the transition region between central cell and

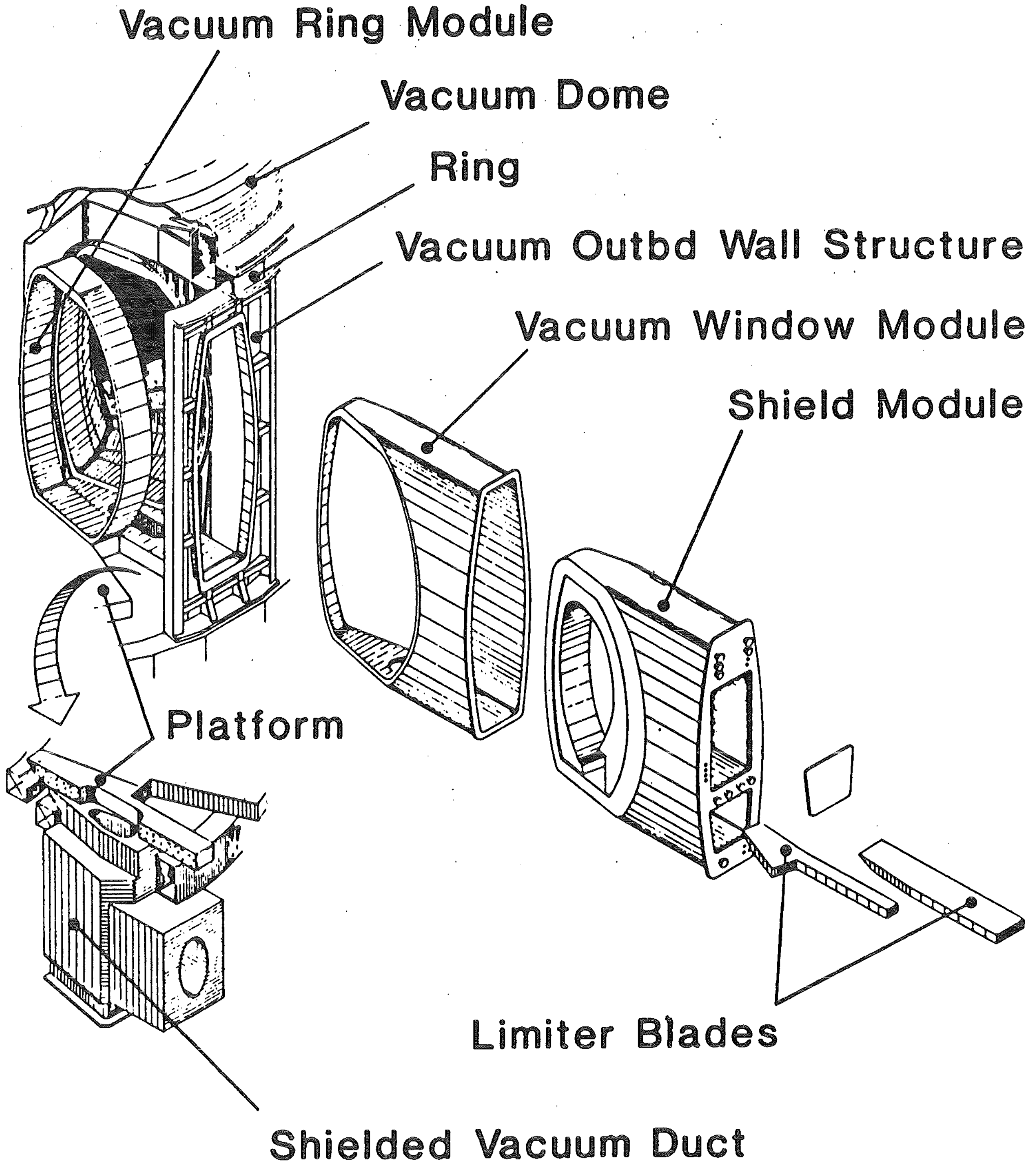


Fig. 3.2-2 TFCX vacuum vessel/shield configuration.

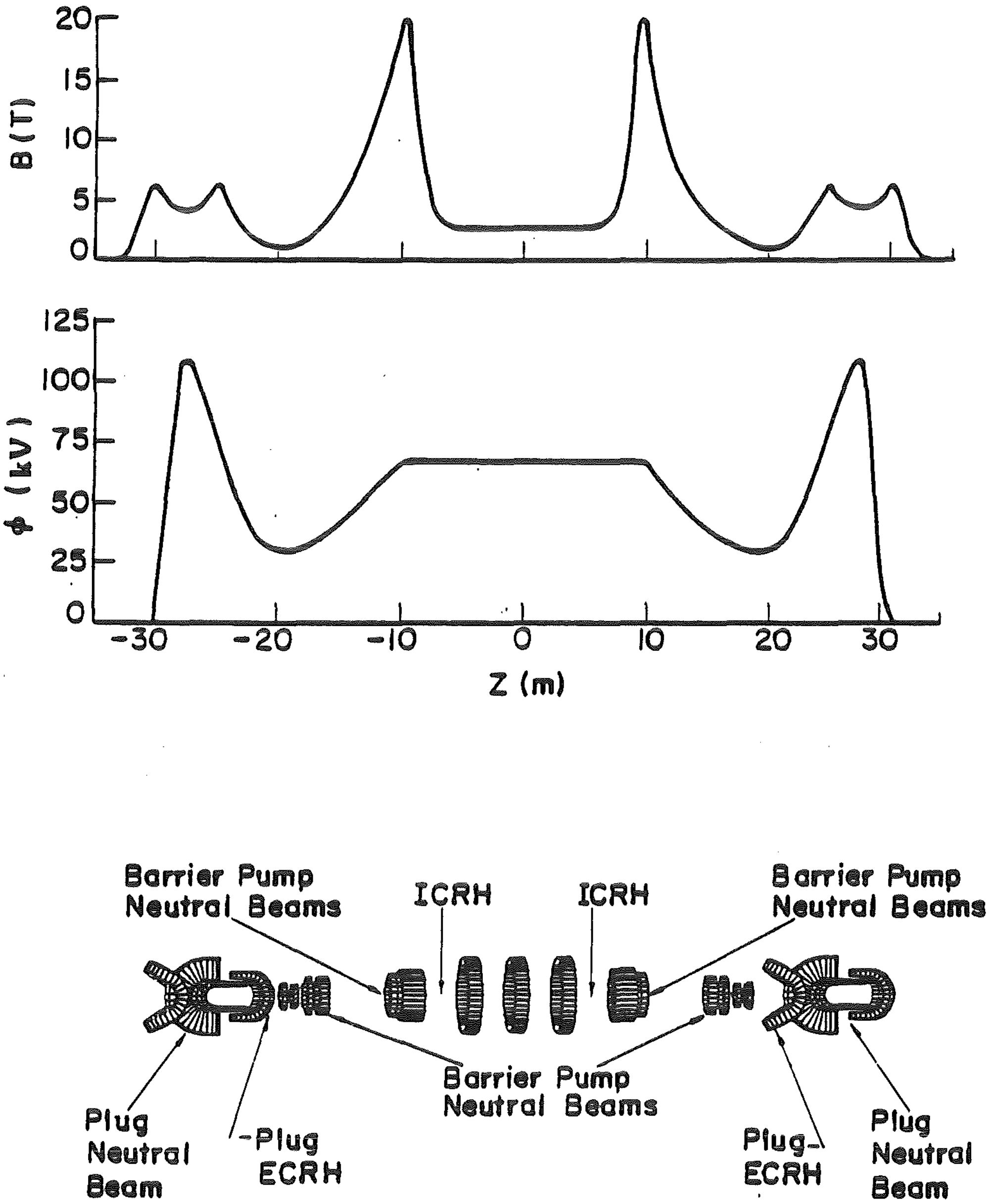


Fig. 3.3-1a. Magnet and electrostatic configuration of TASKA.

end plug was very long to allow a small angle for the main thermal barrier pumping beam.

The TDF[3] design also used the reactor end cell conception of its time. Most of the central cell plasma, however, was mirror-confined and generated by neutral beams. The end cell electrostatic potential was used only to confine a low temperature "stream" plasma needed to stabilize the main central cell plasma. The neutral beams fueled the main plasma, while pellets were used to fuel the stream plasma. The magnetic field and electrostatic potential axial profiles are shown in Fig. 3.3-1b. An alternative scenario, with performance enhanced as presented in Ref. 3, may be available if the stream plasma is not needed for stability.

TASKA-M[4] achieved small size by utilizing neutral beams in the central cell to create a density and electrostatic potential profile peaked off-midplane. This allowed the efficient placement of test modules over the density peaks, and also gave microstability due to warm plasma trapped in the potential dip. No thermal barrier was required, and the end cells were needed only for MHD stability. The magnetic field and electrostatic potential axial profiles are shown in Fig. 3.3-1c. ICRF was used for electron heating, while separate neutral beams fueled both the main plasma and the small warm component.

Parameters for the three designs are given in Tables 3.3-1, 3.3-2, and 3.3-3. Common to the designs is reliance on high magnetic field "choke" coils which create the mirror ratios needed to confine the plasma or to make an effective thermal barrier. The neutron wall loadings are similar, ranging from 1.3 to 1.5 MW/m^2. TDF and TASKA-M require only positive-ion source neutral beams, while TASKA depends on the development of high power, negative-ion source neutral beams. ICRF is used in TASKA to control the central cell ion temperature and in TASKA-M to heat electrons. Electron cyclotron range of frequencies (ECRF) heating is used in TASKA and TDF to do bulk electron heating and also to create a hot, mirror-trapped electron population in the thermal barriers of TDF.

3.3.3 The Magnet Systems

Each of the tandem mirror test devices reviewed here contains the same types of magnets, namely

- large solenoids with moderate magnetic field levels referred to hereafter as central cell coils,

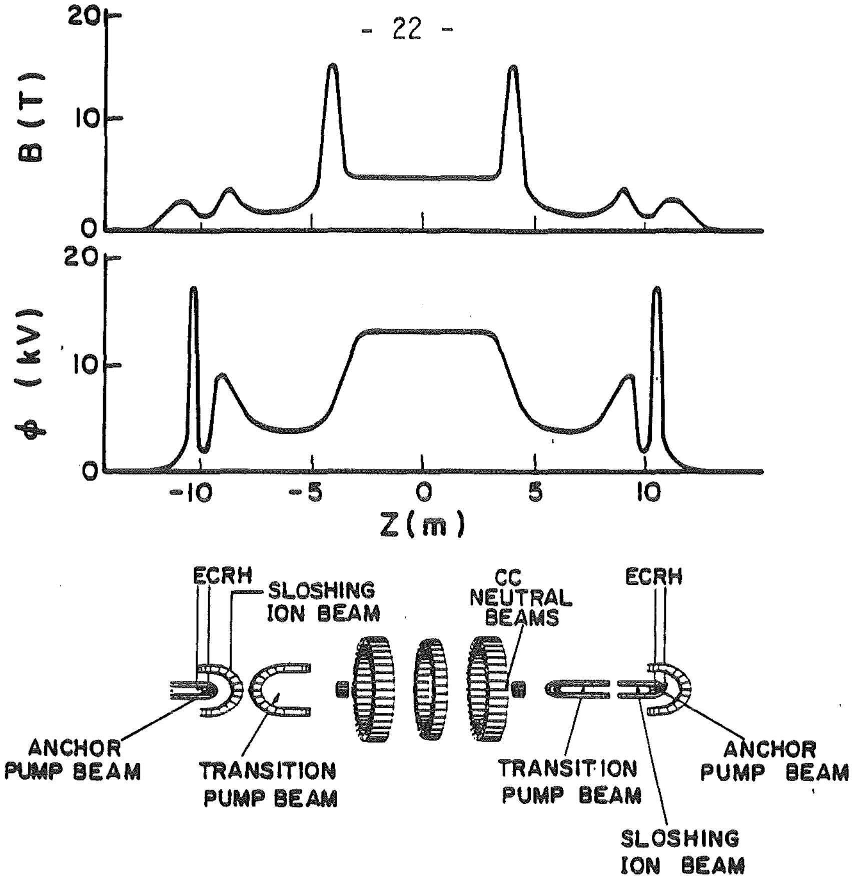


Fig. 3.3-1b. Magnet and electrostatic configuration of TDF.

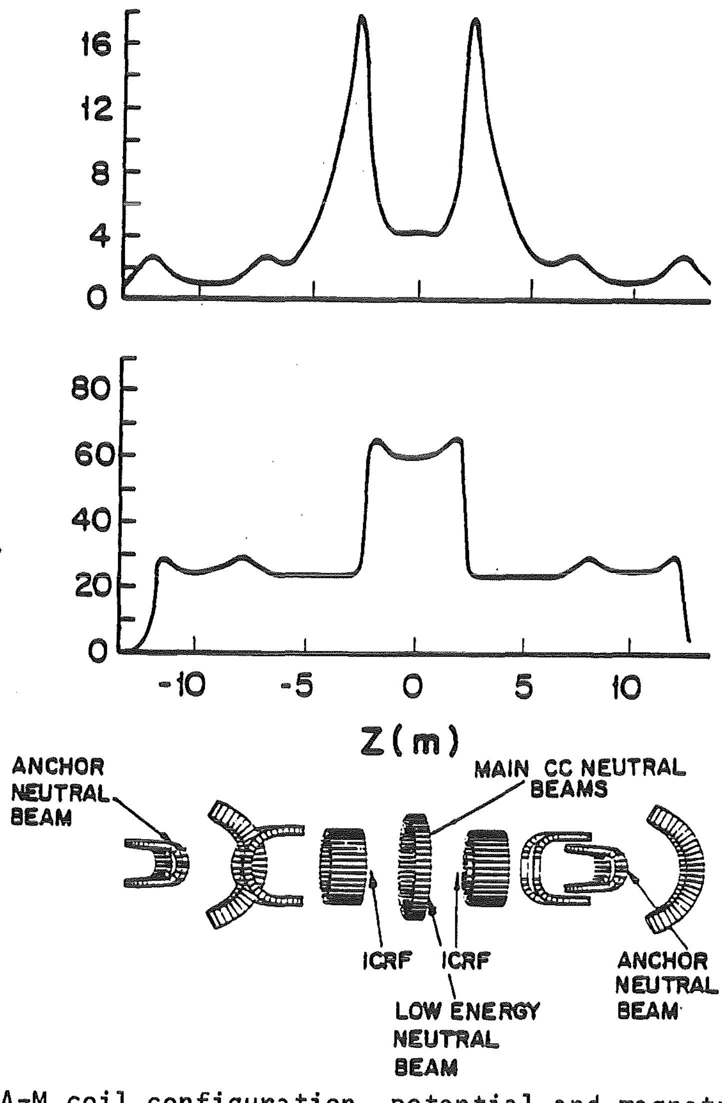


Fig. 3.3-1c. TASKA-M coil configuration, potential and magnetic field.

Table 3.3-1. Central Cell Plasma Parameters

PARAMETER	UNIT	TASKA	TDF*	TASKA-M
DT Fusion Power	MW	86.0	20.0	6.8
Q		0.74	0.39	0.17
Peak Neutron Wall Loading	MW/m^2	1.5	1.4	1.3
Plasma Radius	m	0.32	0.10	0.12
Plasma Length	m	19	8	5
Average Ion Energy	keV	45	37	84
Electron Temp.	keV	12.0	2.1	14.0
Average Ion Density (midplane)	m^{-3}	1.9×10^{20}	3.6×10^{20}	2.0×10^{20}
Energy Confinement Time	s	0.280	0.012	0.030
Midplane, On-Axis Magnetic Field	T	2.7	4.5	4.2
Mirror Ratio		7.4	3.3	4.2
Volume-Averaged Beta	%	50	24	30
Neutral Beam Power	MW	---	51	21/0.6
Neutral Beam Voltage	kV	---	80	90/12
ICRF Power	MW	40	---	12
Fueling Method		NB	NB/Pellet	NB

* Reference, stream-stabilized case.

Table 3.3-2. Thermal Barrier Plasma Parameters

PARAMETER	UNIT	TASKA	TDF	TASKA-M
Minimum Density	m^{-3}	6.8×10^{18}	3.0×10^{18}	(No Thermal Barrier)
Plasma Length	m	9.0	4.8	
Minimum B Field	T	0.8	1.0	
Hot Electron Energy	keV	---	400	
Barrier Potential	kV	38	12	
Total NB Power	MW	6.6/50/0.2	3.6/0.2	
NB Voltage	kV	76/50/5	80/80	
Total ECRF Power	MW	---	0.5	

Table 3.3-3. Plug Plasma Parameters

PARAMETER	UNIT	TASKA	TDF	TASKA-M
Midplane Density	m^{-3}	6.3×10^{19}	5.0×10^{18}	2.6×10^{19}
Plasma Length	m	5.0	3.3	4.3
Average Ion Energy	keV	388.0	*	60.0
Electron Temperature	keV	59.0	6.5	14.0
Ion Plugging Potential	kV	43	4	---
Maximum B Field	T	6.3	3.0	2.7
Minimum B Field	T	4.0	1.0	1.0
Volume-Averaged Beta	%	64	30	30
Total NB Power	MW	5.4	0.25	7.0
NB Voltage	kV	250	80	73
Total ECRF Power	MW	15.0	0.7	---

* Not available

- C-shaped coils, also with moderate magnetic field levels, referred to as transition coils or yin-yang magnets,
- ultrahigh field solenoids with less than 60 cm bore referred to as choke coils.

The different design parameters for these coils stem from the constraints imposed by the plasma physics models, as shown in Fig. 3.3-1.

3.3.3.1 The C-Coils

Table 3.3-4 summarizes some major design values for the three devices. For the purpose of comparison, a column for the MFTF yin-yang coils is also added, which represents a successfully fabricated type of such C-coils.(5) Some general conclusions can be drawn from Table 3.3-4:

- Based on the field level at the magnets of below 8 T, standard NbTi conductors for d.c. operation mode can be used in all cases.
- The magnet size and associated forces allow cryogenic stabilization with LHe I cooling to be used even though this results in limited current density.
- Sufficient neutron shielding can be included to ensure end of life radiation effects in the magnet materials which do not exceed conservative criteria for Cu, NbTi and organic insulators. This shielding will not dominate the ultimate dimensions of the coils.

Table 3.3-4. Summary on the Design Data for the C-Coils in Mirror Fusion Test Devices

C-Coils	UNIT	TASKA	TASKA-M	TDF	For Comparison MFTF (Existing)
Maximum Magnetic Field	T	7.0/7.9	5/6	4.3/4.7	7.7
r_{min}	m	1.6/0.9	1.4/0.6	0.35	0.56
r_{max}	m	1.4/1.4	0.8/2.1	1.25	2.1
Current Density	MA/m^2	19/16.3	25	30/34.3	25.3
Stored Energy	MJ	487/411	33/96	13.7/12.9	192
End of Life Radiation Effects					
Electrical Insulator	rad	1.1×10^9	1.3×10^9	5×10^9	N. Approp.
Fast Fluence (S/C)	n/m^2	N. Avail.	6.1×10^{21}	N. Avail.	N. Approp.
Cu Stabilizer	dpa	4.5×10^{-4}	5×10^{-4}	$1.1 \cdot 10^{-4}$	N. Approp.

- Even for TASKA, the largest device proposed, the C-magnets are only slightly larger than the tested MFTF magnets.

Thus, one can conclude that the C-coils for tandem mirror test devices can be built with confidence, based on present-day technology.

3.3.3.2 The Central Cell Coils

Table 3.3-5 summarizes the major design values for the central cell solenoidal magnets. If one compares these data with those of the MFTF-B solenoids already built (but not tested), it can be seen that they are representative in diameter and current density, but not in field strength, volume and stored energy. On the other hand, as seen from the table, the test facility design approaches are rather conservative, using:

- standard NbTi conductors for d.c. operation mode,
- cryogenic stabilization with LHe I cooling,
- sufficient shielding to guarantee well acceptable radiation damage levels.

Thus, based on the experience gained with large NbTi magnets in general, e.g. the much more complicated LCT D-shaped torus coils,[6] one can conclude that these magnets can also be built with confidence, based on present-day technology.

3.3.3.3 The High Field Choke Coils

In spite of their simple solenoidal geometry, the choke coils require advanced technology because of the very high magnetic fields generated and the reduced space for radiation shielding.

If one takes into account the limits on present-day superconductors, it is found that all designs must use normal conducting inserts to enhance the field above 15 T (see Table 3.3-6). The TDF design even includes a complete normal conducting coil with rather high ohmic losses, in spite of limiting the field to 15 T. In the other designs the normal conducting insert also serves as a radiation shield, but additional shielding will still be needed between the insert and the innermost superconducting turns to protect them from nuclear heating and damage. Unfortunately, this enlarges the superconducting coil.

A further step toward reducing the need for ohmic power would be the development of advanced superconductors which will reach at least 20 T. There is a possibility that this goal can be met within the next decade.

In summary, the choke coils proposed for the devices discussed represent credible designs, but optimal solutions remain an item for development.

Table 3.3-5. Summary on the Design Data for the Central Cell Solenoids in Mirror Fusion Test Devices

CC-Coils	UNIT	TASKA	TASKA-M	TDF
B - on axis	T	2.7	4.2	4.5
B - maximum at S/C	T	5.5	7.4	7.6
Inner Radius	m	2.8	1.8	2.0
Outer Radius	m	3.4	2.4	3.0
Axial Length	m	1.2	0.6	0.86
Stored Energy	MJ	368	232	700
Current Density	MA/m^2	13.5	22	10.7-21.9
End of Life Radiation Effects				
Electrical Insulator	rad	1.5×10^8	1.8×10^8	5×10^9
Fast Fluence (S/C)	n/m^2	N. Avail.	5.2×10^{21}	4×10^{22}
Neutron Damage (Al Stabilizer)	dpa	8.5×10^{-5}	6.3×10^{-4}	3.7×10^{-4}

Table 3.3-6. Summary on the Design Data for the Choke Coils in Mirror Fusion Test Devices

Choke Coils	UNIT	TASKA	TASKA-M	TDF
B - on axis	T	20.0	17.5	15.0
B - maximum at S/C	T	15.0	12.0	---
Inner Radius - S/C	m	1.3	1.15	---
Stored Energy - S/C	MJ	2600	375	---
$P_{loss_{NC}}$	MW	11.6	18.0	26.0
Current Density	MA/m^2	16-24	23.8-27.5	---
End of Life Radiation Effects (S/C)				
Electrical Insulator	rad	5.6×10^7	1.5×10^8	N. Approp.
Fast Fluence (S/C)	n/m^2	N. Avail.	6.3×10^{21}	N. Approp.
Al Stabilizer	dpa	3×10^{-4}	7.6×10^{-4}	N. Approp.

3.3.4 Device Layout

Figures 3.3-2a to 3.3-2c show the overall layout for the three devices considered here. The overall machine lengths are 100 m for TASKA, 59 m for TDF and 50 m for TASKA-M. If we define the central cell as that part of the machine which has solenoidal coils, then the central cell lengths are 23.4 m, 6.4 m and 8.7 m, respectively. The lengths are dominated by the end plugs, accounting for 76% of the length of TASKA, 89% for TDF and 83% for TASKA-M.

With the exception of TASKA, the vacuum chambers surround the machines on all sides. Thus all the coils and their shields are inside the vacuum chamber and the intercoil structures are integrated into the shield. TASKA has the vacuum chamber on the inside of the central cell coils, but on the outside of the end plug coils. For this reason, the central cell intercoil support structure is prominent in TASKA (Fig. 3.3-2a).

The diameter of the vacuum chambers in the end regions is 15 m in TDF. In TASKA and TASKA-M, the end region vacuum chambers are oval shaped with dimensions of 5 x 15 m and 8 x 4 m, respectively. All three machines have cryopumps or cryopanels integrated into the end regions and all incorporate a direct convertor test module. Further, they all have beam dump modules that are easily accessible for changeout and maintenance.

3.3.5 Blanket and Materials Test Possibilities

3.3.5.1 General Considerations

One of the major tasks of the facilities addressed here is to provide a test bed for fusion blankets and materials. Synergistic effects of stress, corrosion, neutron-induced mechanical property changes, magnetic fields, neutronic and thermal hydraulic effects cannot normally be tested without such nuclear facilities. Thus, it was a major goal in all designs to provide sufficient space and a relevant nuclear environment for testing of blanket modules. Table 3.3-7 gives a survey on the general data for the installation of such test modules into the test facilities considered. The table shows that in all machines, at least 1-2 blanket modules with volumes > 1.5 m^3 and surface areas > 1 m^2 at the first wall can be tested for sufficiently long time (0.7-7.8 FPY) to gain relevant information. Detailed studies are now underway to define test needs and requirements, as e.g. the FINESSE study carried out in USA.[7]

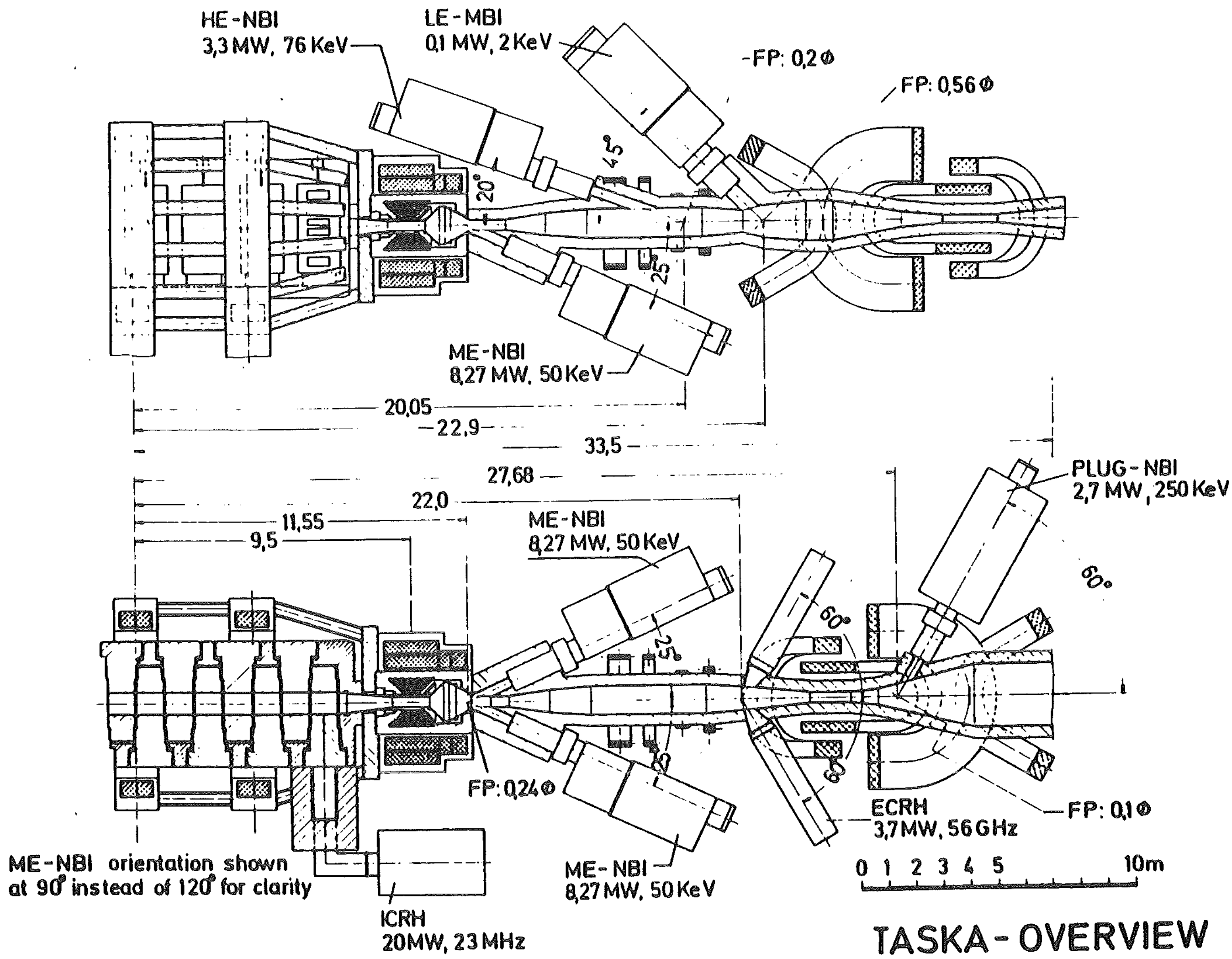


Fig. 3.3-2a. Overview of TASKA.

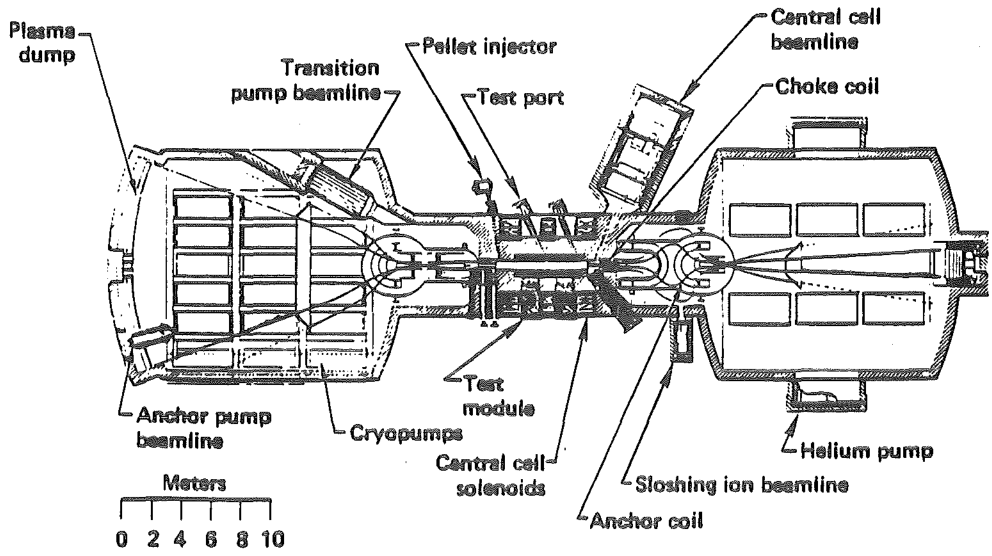


Fig. 3.3-2b. Overview of TDF.

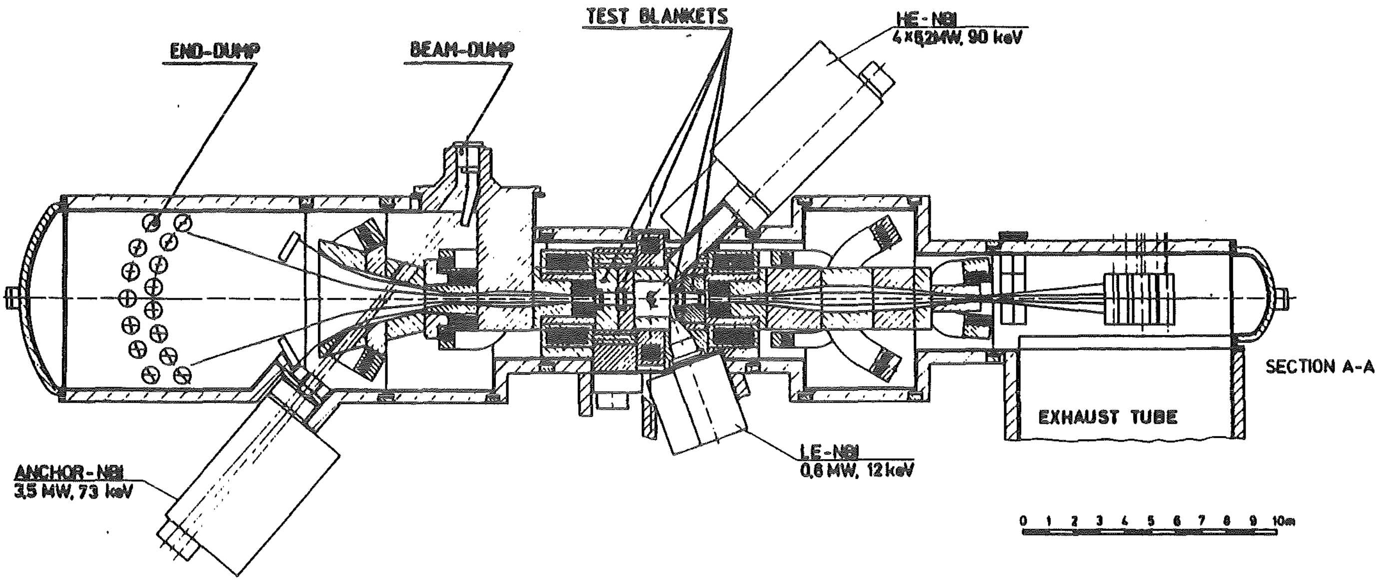


Fig. 3.3-2c. Schematic of TASKA-M technology test facility.

Table 3.3-7. Summary of Relevant Data for the Capability of Test Modules in Selected Mirror Fusion Test Devices

	TASKA	TASKA-M	TDF	MFTF-α+T
Number of possible test modules	2	4	2	1
First wall radius (cm)	46	22/22/25*	25	25
Axial length of one module (cm)				
Blankets	101	83/83/43	100	100
Materials	70	108	100	---
Useful radial thickness (cm)				
Blankets	96	45/64/15	50	---
Materials	20	20	---	---
Possible sector angle (degree)	360	360	3 x 120	360
Area of one module at first wall (m^2)	2.9	1.15/1.15/0.7	1.6	1.6
Maximum active module volume (m^3)				
Blankets	2.5	1.0/1.8/0.13	1.6	---
Materials (capsules)	0.14	0.15	---	---
Peak neutron wall loading (MW/m^2)	1.5	1.2/1.2/0.8	1.4	2.0
Surface heat flux density (kW/m^2)	---	100/100/36	100	---
Maximum test time (FPY)	5.3	7.8	3.6	0.7

* For LiPb/Li/Li_2O Blanket Modules

3.3.5.2 Blanket Test Modules

The TASKA and TASKA-M studies have shown that test blankets can be designed to demonstrate the suitability of these facilities for large module blanket testing. Table 3.3-8 shows some major data supporting this idea. The values on power density, tritium production, coolant temperature, and energy multiplication support the opinion that these blankets can serve as appropriate test modules of relevant size in accordance with the overall dimensions from Table 3.3-7.

3.3.5.3 Materials Test Modules

The TASKA[2] and TASKA-M[4] reports included detailed analyses of the material test modules. A summary of the designs is given in Table 3.3-9 and

Table 3.3-8. Major Data for Test Blanket Examples in TASKA and TASKA-M

	TASKA-M			TASKA
Breeder	Li	$Li_{17}Pb_{83}$	Li_2O	Li
Enrichment of ^{6}Li (%)	7.5	90	30	---
Coolant	Breeder	Breeder	H_2O	Breeder
Structural material	HT-9	HT-9	316 SS	1.4970 steel
Coolant temperature (°C)	300/450	300/450	150/200	300/450
Local TBR	1.19	1.15	0.71	1.32
Energy multiplication M	1.37	1.34	1.31	0.9
Module power (MW)	1.0	1.0	0.38	3.5
Power density (MW/m^3)	3.5-0.4	6-0.55	7-2.5	2.5-0.1
T-production rate (gT/d)	0.22	0.21	0.08	0.82
Blanket thickness (cm)	64	45	15	96
Reflector thickness (cm)	26	45	110	28

Table 3.3-9. Summary of Design Data for Two Tandem Mirror Materials Test Modules

		TASKA	TASKA-M
Peak Neutron Wall Loading	MW/m^2	1.5	1.34
Potential Number of Test Capsules		351	456
Peak dpa Rate	dpa/FPY	16	10
Vol. Average dpa Rate	dpa/FPY	11	3.6
Cumulative Damage	dpa•ℓ	8045	4120*
Number of Specimens Tested			
Microscopy		17,200	12,200*
Mechanical Properties		11,630	9,150*
Physical Properties		720	1,220*
TOTAL		29,550	12,570*
Maximum dpa over life	dpa	85	78

*Design only completed for one of two modules.

shown in Figs. 3.3-3a and 3.3-3b. Because of geometrical considerations, temperature controlled capsules were placed around and perpendicular to the plasma axis. The number of individually controlled capsules ranged from 351 in one module of TASKA to 456 in two modules of TASKA-M. Coupled with the peak damage rates of 10 to 16 dpa/FPY the 20 cm long capsules produce a damage-volume quantity of 4120 dpa-ℓ to 8045 dpa-ℓ for TASKA-M and TASKA, respectively.

Perhaps the most important feature of the test modules is the large number of specimens which can be accommodated. The larger test module of TASKA can accommodate over 29,000 specimens while one of the TASKA-M modules can handle over 22,000 specimens over the operating life. Since the design was completed for only one of the two modules in TASKA-M, even more specimens could be studied.

In order to determine the adequacy of a TASKA level device to provide design data for a DEMO, we must review the general philosophy followed in this study. First we had to determine the general materials performance goal associated with a commercial reactor. At the present time a value of 20 MW-yr/m^2 (200 dpa) is considered economically attractive. Since it is commonplace in a research program to limit the extrapolation of the materials data base by no more than a factor of 2, the goal fluence in the DEMO should be 100-150 dpa. The same philosophy would indicate that a goal fluence of 50-75 dpa for the materials test facility should be adequate to design the DEMO. This is consistent with the 78 and 85 dpa value for TASKA-M and TASKA, respectively.

The general conclusion of the tandem mirror test facilities is that there is adequate test volume and neutron flux to supply the information needed to design a demonstration plant. The neutron fluence achieved over the lifetime of the machines is, however, somewhat less than that expected in a DEMO but is considered adequate to extrapolate to reasonable DEMO lifetimes.

3.3.6 Tritium and Exhaust

The continuous operation of the test facilities requires careful consideration of exhaust and fueling as well as of the specifics of the tritium circuit. These questions have been carefully analyzed in TASKA, TASKA-M and TDF. In all the devices, tritium fueling is accomplished by one of the neutral beam injection systems. This, together with the low burnup, leads to relatively large tritium flow rates as summarized in Table 3.3-10. The pump systems must be able to handle such particle streams.

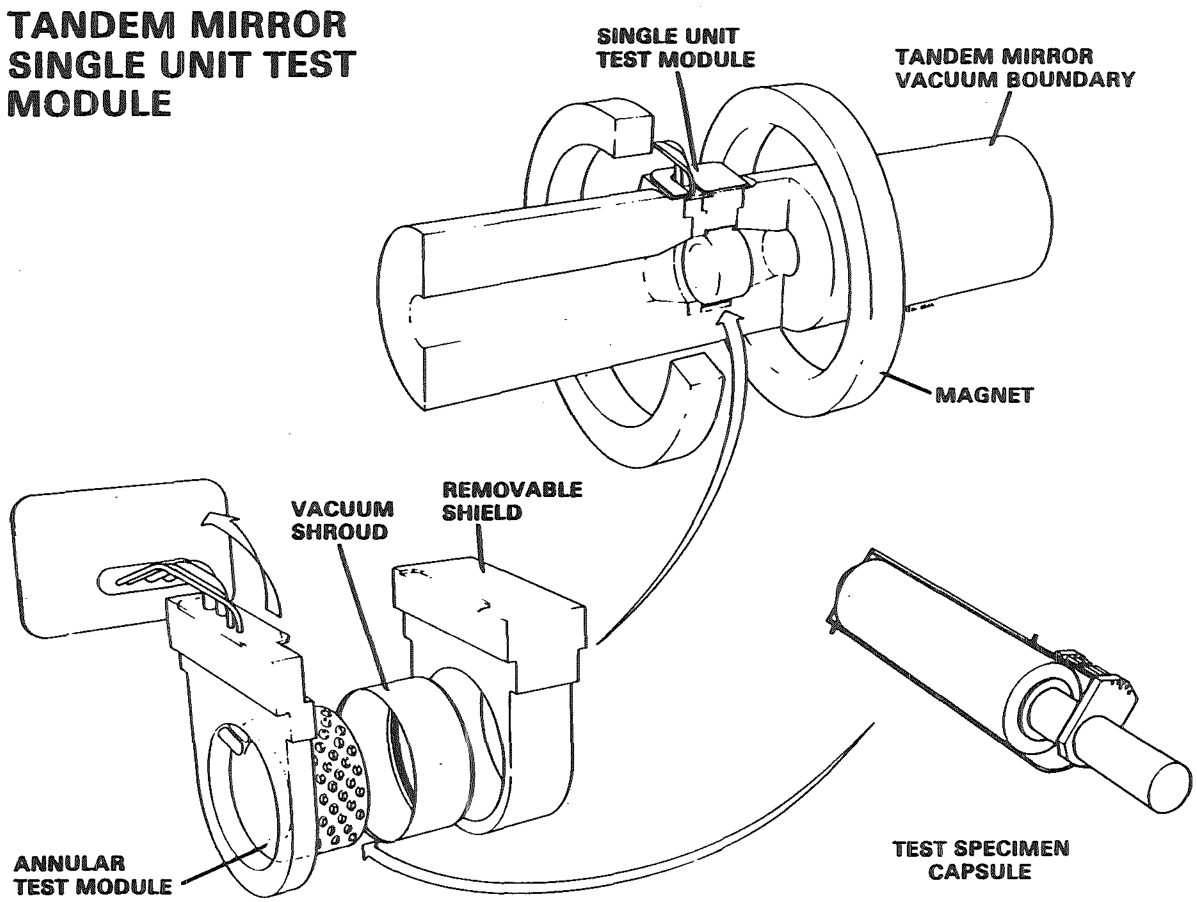


Fig. 3.3-3a. TASKA single unit test module.

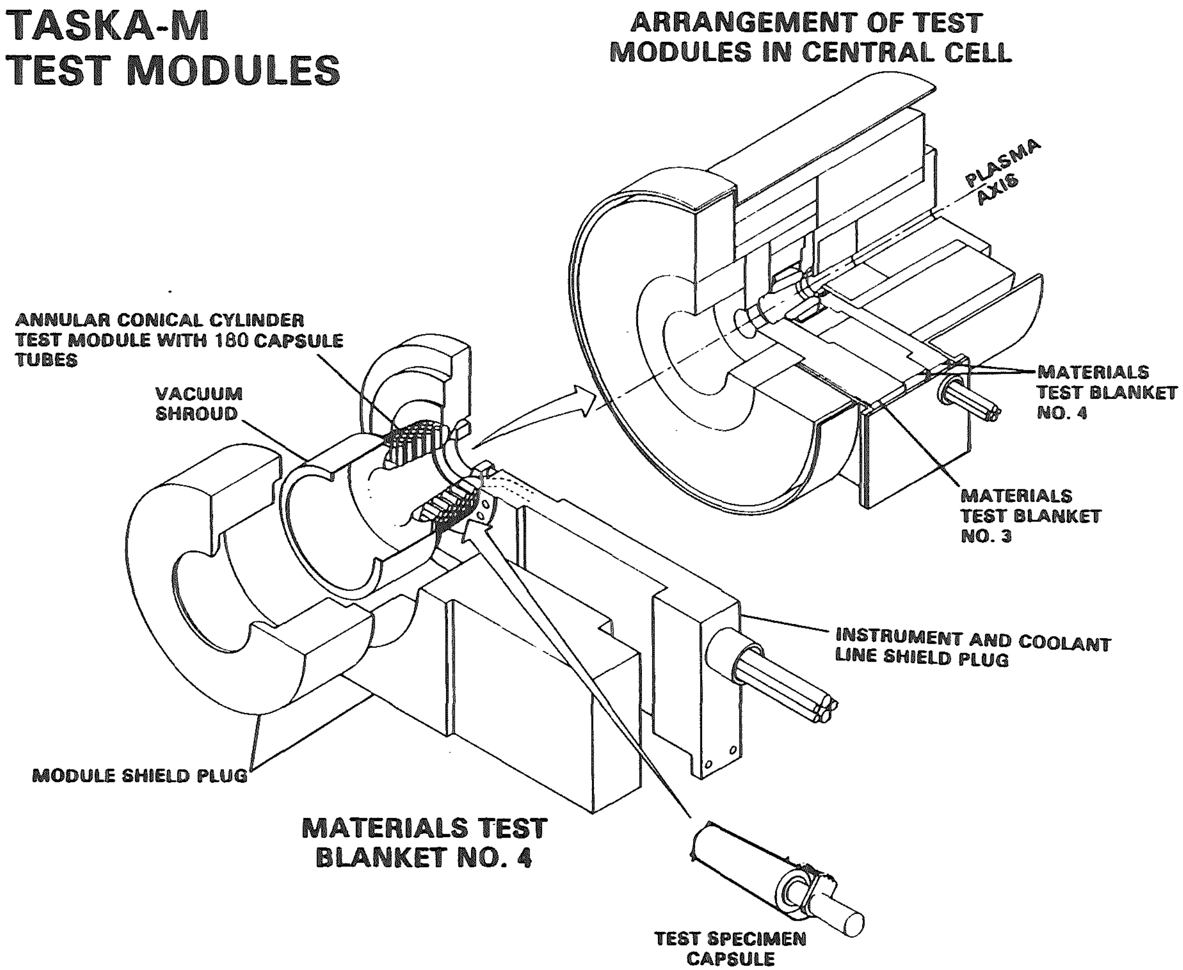


Fig. 3.3-3b. TASKA-M test modules.

Table 3.3-10. T-Flow Rates (g/FPd) for Several Mirror Fusion Test Facilities

	TASKA-M	TASKA	TDF
Injection into the plasma by neutral beams	347	1500	1187
NBI-Recycle (pumps)	1388	3500	5789

In TASKA and TASKA-M, pumping is provided in the end cells and the central cell while in TDF the pumping is accomplished in the end cells only. Of course, all neutral beam injectors have pumping systems as well. The tritium flow rate of 1 to 6 kg/d in these injectors is a great disadvantage of this T-fueling method. The preferred method for the primary system is the use of cryocondensation panels for D/T and cryotrapping with Ar for the He ash. Only the neutral beam injectors for TASKA use Zr-Al getter pumps. The size and requirements for all these pumps are rather large, with total cryopanel sizes of a few hundred to 100 m^2. Their successful operation would be an important demonstration of the availability of such "vacuum and exhaust" components for nuclear fusion devices.

The use of the cryopumping method requires that the cycle time be limited to values acceptable with respect to the tritium inventory. Thus, cycle times of several hours have been selected. The regeneration time can be kept to less than one quarter of the cycle time so that the additional pumping capacity to be installed is limited to about 25%. As can be seen from Table 3.3-11, the tritium amount accumulated in the cryopumps represents the largest fraction of active tritium inventory in all the devices considered.

The tritium circuit scheme is very similar in all devices; only the size has to be adapted to the specific flow rates. The absolute numbers are roughly an order of magnitude larger than for the currently operating TSTA and of the same order as for INTOR. As shown in Chapter 7, the information achievable is of direct relevance to all next generation nuclear fusion systems.

3.3.7 Costs

A difficult and sometimes delicate issue is to carry out and compare cost estimates for devices designed by different groups and at different times. In the present case, this comparison is facilitated due to the fact that the estimation procedure for all designs used the same or similar unit costs and cost algorithms. These are mainly the units used for INTOR based on a sug-

Table 3.3-11. T-Inventories (g) of Several Mirror Fusion Test Facilities

	TASKA-M	TASKA	TDF	MFTF-U*
NB-pumps (cryo or getter panels)	116	259	307	10
Central cell and beam dump pumps (cryopanels)	5	6	< 10	1
End cell pumps	55	119	66	5
Fuel cleanup unit	3	47	100	50
Isotope separation system	30	28	600	250
Coolant water* (end of life)	34	15	2	N.A.
Machine TOTAL	243	474	1085	316
Blanket test modules	6-25	18.3	N.A.	N.A.
Fuel storage	31	5000	N.A.	31

* Only rough estimates

gestion by the Fusion Engineering Design Center at ORNL(8) and those described by the Battelle Institute.(9) Thus, at least a fair comparison of the relative costs and their relation to tokamak devices can be undertaken.

Table 3.3-12 shows the direct costs as reported for the three devices TASKA, TASKA-M and TDF, divided into categories based on a format adopted for INTOR. The differences in the total direct costs reflect the increasing size of the facilities. Unusual differences in the estimations for some categories are probably due to a different level of detail in the design, but such differences have limited influence on the total cost. The comparison is graphically displayed in Fig. 3.3-4 also in relation to tokamak devices. The year that the cost estimates were made varied from 1981 to 1983 for the mirror devices; the TFCX costs are based on 1984 dollars. It can be seen that the tandem mirrors and tokamaks separate into two categories.

The first category represents the relatively low power (6 to 86 MW) tandem mirror facilities that range from 400 to 800 million dollars in direct capital costs. The second group represents the higher power tokamaks (~ 200-600 MW) which range from 1000 to 1400 million dollars in direct cost. It is

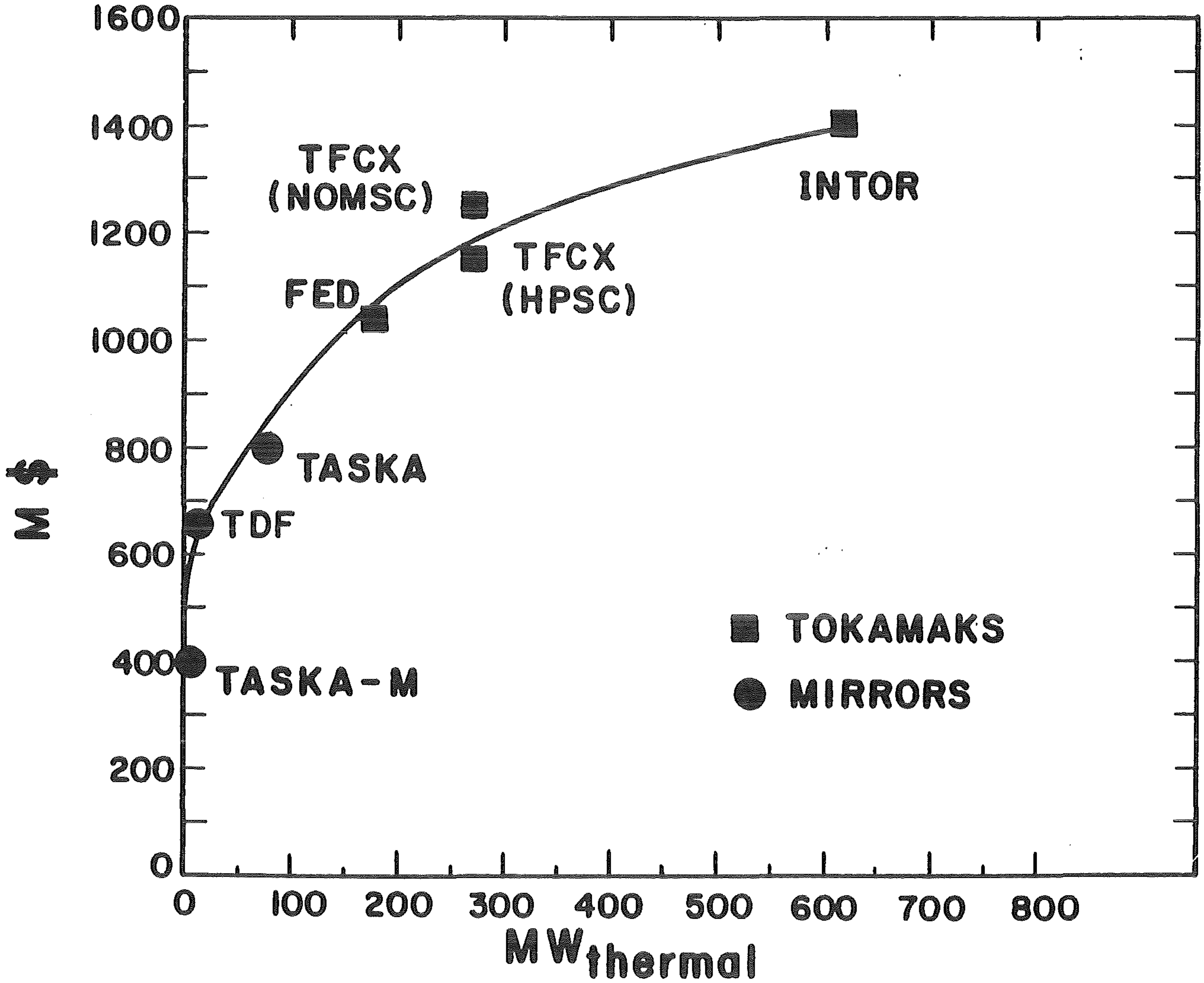


Fig. 3.3-4. Direct cost vs. power level for nuclear test facilities.

Table 3.3-12. Comparison of Direct Costs for the Test Devices TASKA, TDF and TASKA-M

	TASKA (1981) 10^6 \$	TDF (1982) 10^6 \$	TASKA-M (1981) 10^6 \$
Blanket/Shield/ Vacuum Chamber	46	21	2
Magnets	228	51	59
Plasma Heating	270	161	91
Electrical	25	21	31
Auxiliary Cooling	19	19	16
Instrumentation & Control	25	27	25
Fuel Handling	18	58	23
Maintenance Equipment	20	59	20
Primary Heat Transport	22	14	13
Secondary Heat Transport	6	---	---
Reactor Vacuum	16	17	16
Radwaste Treatment	1	3	3
Thermal Dumps	1	2	1
Reactor Support Structure	14	2	8
Special Materials	---	56	2
Miscellaneous Plant	---	8	10
Buildings	80	134	67
Total Direct Costs	780	653	406

Table 3.3-13. Comparison of Operating Costs for TASKA and TASKA-M

Year of Operation	Assumed Availability (%)	Operating Costs (10^6 $) TASKA	TASKA-M
1	10	35	24
2 and 3	15	39	27
4 to 7	25	46	34
8 to 20	50	65	50
levelized		54	43

not too surprising that the lower power tandem mirrors are less expensive than the higher power tokamaks and Fig. 3.3-4 shows that a "minimum" cost to generate neutrons in such facilities may be on the order of 400 M$ in direct capital costs. At the same time, the tandem mirror facilities could achieve much higher neutron fluence levels (6-8 MW-yr/m^2) than tokamaks (0.3 to 3 MW-yr/m^2) even though the mirror facilities may cost 1/2 to 1/3 of the tokamaks.

Difficulties in comparison occur if indirect costs, e.g. engineering, office costs, contingency etc., are included. Here different percentage levels are used, but in all cases the indirect costs could add as much as 50% to the direct costs of the project.

In addition the operating costs of test devices play an important role. Estimates have been carried out so far only for TASKA and TASKA-M. These are summarized in Table 3.3-13. The rather marginal differences in the values in spite of the factor of 12 in the thermonuclear power reflect clearly the limiting point. TASKA-M is highly driven so that the electricity costs remain high; the tritium purchasing costs do not play a significant role. The electricity demand in TDF is similar to that of TASKA, and consequently it can be stated that the operating costs will be of the same order of magnitude.

References for Chapter 3

1. "TFCX - Preconceptual Design Report," Princeton Plasma Physics Laboratory Report F-Axxx-8407-006 (1984).

2. B. Badger et al., "TASKA, A Tandem Mirror Fusion Engineering Facility," KfK-3311, 1982.

3. J.N. Doggett et al., "A Fusion Technology Demonstration Facility (TDF)," UCRL-90824, 1984.

4. B. Badger et al., "TASKA-M, A Low Cost, Near Term Tandem Mirror Device for Fusion Technology Testing," KfK-3680, 1983.

5. T.A. Kozman et al., "Magnets for the Mirror Fusion Test Facility: Testing of the First Yin-Yang and the Design and Development of Other Magnets," UCRL-88111 (1983) and IEEE-Trans. on Magnetics, Vol. MAG-19, pp. 859-66 (1983).

6. W. Herz et al., "Results of the Test of the European LCT-Coil in the TOSKA Facility," Paper CL-4, Appl. Supercond. Conf. 1984, San Diego, to be published at IEEE-Trans. on Magnetics.

7. M.A. Abdou et al., "FINESSE Interim Report," UCLA-ENG-84-30 (Oct. 1984).

8. C.A. Flanagan, D. Steiner and G.E. Smith, (Eds.), "Fusion Engineering Device Description," ORNL/TM-7948 (1981).

9. S.C. Schulte, T.L. Wilke, J.R. Young, "Fusion Reactor Design Studies - Standard Cost Estimating Rules," PNL-2987 (1978).

APPENDIX 3.A. PHYSICS ISSUES

3.A.1 Introduction and Overview

Testing of physics, per se, is not part of the objectives of TASKA, TDF, or TASKA-M. They aim at showing that a tandem mirror can serve as an excellent technology test facility with the implicit assumption that, by the time TASKA, TDF, or TASKA-M would be built, physics-oriented machines such as Phaedrus,(1) TMX-U,(2) TARA,(3) and MFTF-B(4) would have resolved the issues. Thus, this appendix takes a somewhat different tack than that of the rest of the study. Here, the designs will be analyzed with regard to how well the designs address and satisfy physics criteria within the present body of experimental and theoretical knowledge, rather than assessing which issues need further theoretical and experimental work.

Five major issues will be examined in subsequent sections: (1) MHD stability, (2) trapped particle modes, (3) microstability, (4) thermal barrier physics, and (5) startup. A brief overview will be presented first.

In general, the designs reflect the increasing state of physics knowledge with time. In TASKA, the earliest study, relatively well-known neutral beam physics was assumed for pumping the thermal barriers, and a hot trapped-electron population was not invoked. Under present theory, some modifications of the end cell configuration would have been required to achieve microstability, although they would probably not have had a major impact on the overall design. TASKA would have been stable to MHD and trapped particle modes -- despite the fact that the latter modes were conceived after TASKA had been published. TDF, the next study, used physics proposed for TMX-U, TARA, and MFTF-B. All theoretical criteria for stability to MHD modes, trapped particle modes, and microinstabilities were satisfied -- the primary uncertainty arising from the requirement of 400 keV electrons in the thermal barriers. Startup of both TASKA and TDF would be an intricate process. TASKA-M, the most recent effort, attempted to minimize device size and cost while relaxing physics constraints somewhat. In particular, the only microinstabilities which were design constraints were those which have been experimentally observed, although TASKA-M would possibly be unstable to some modes predicted by theory. On the other hand, TASKA-M was more conservative in other aspects of the physics which led to design simplification -- no thermal barrier was used and startup should be straightforward.

It is also worth noting that a number of new experimental and theoretical developments have occurred since these studies were completed. The direction of the developments is toward axisymmetry which would, presumably, allow even simpler and less expensive tandem mirror test facilities. One idea is to replace the yin-yang end magnets by octupole magnets.(5) Other ideas, allowing full axisymmetry, are RF stabilization,(1) electron ring stabilization,(6) and wall stabilization.(7) Intense work on all of these ideas is in progress, and their impact on tandem mirror design -- both for reactors and for test facilities -- will be substantial.

Table 3.A-1 gives a brief overview of the physics issues, which will be expanded upon in subsequent sections and tables. Readers wishing to delve more deeply into the issues should consult the references contained in any of these studies and, in particular Ref. 8.

The major point of Table 3.A-1 is the bottom line: From a **physics** perspective, TASKA and TDF were judged to have a relatively low overall degree of risk, while TASKA-M was judged to have a moderate degree of physics risk. The important implication of this is that TASKA, TDF, and TASKA-M demonstrate that a credible physics design of a tandem mirror-based technology test facility is possible.

3.A.2 MHD Stability

Two types of instabilities, interchange modes and ballooning modes, are important here. Both types of modes are driven by plasma pressure, with stability depending on details of the magnetic field curvature. Interchange modes are flute-like; they extend with constant amplitude along the whole axis of a device. Ballooning modes possess structure along the axis and are generally more difficult to stabilize. A fair body of experimental evidence exists relating to MHD instabilities. Interchange modes are much easier to analyze, and the stability criterion can be analyzed for most configurations with minor modifications to existing computer codes. TASKA, TDF, and TASKA-M were all analyzed numerically and found to be stable to interchange modes. Ballooning modes are much more difficult to treat. Only TDF stability was analyzed with the chief existing tool, the LLNL TEBASCO code, since it is set up to examine the similar geometry of MFTF-B. For TASKA and TASKA-M, ballooning modes were addressed by choosing an operating point far from the interchange mode stability boundary. Experience shows that such a point will generally also be ballooning mode stable.

Table 3.A-2 contains a summary of MHD stability issues. Since all three devices relied on yin-yang magnets to provide a minimum-B end cell anchor configuration, which is a standard method, all are judged to have a low degree of risk with respect to MHD issues.

3.A.3 Trapped Particle Modes

Trapped particle modes are of relatively recent discovery[9] -- in fact, they postdate the TASKA study. The modes are driven by plasma pressure but, unlike MHD modes, only possess axial electrostatic potential perturbations in regions of "bad" and "neutral" curvature. Only relatively rudimentary analytic techniques for assessing their stability exist, and no computer codes were available at the time the studies were done.

By the analytic criteria, TASKA and TDF are stable to the mode, while TASKA-M may be unstable to a mode with azimuthal mode number about five. A variety of physical effects are being examined in trapped particle mode theory, and a number of plausible stabilizing schemes have been proposed. Examples are finite Larmor radius effects, inward-directed electric fields, and collisions. There is no strong experimental evidence for the mode at present.

Trapped particle mode issues are summarized in Table 3.A-2. TASKA and TDF are judged to have low risk from the mode. TASKA-M is judged moderate, although the lack of experimental evidence and the early stage of theory efforts indicate that trapped particle modes should not cause undue worry for the device at this time. The TARA experiment will test the theory in the near future.

3.A.4 Microstability

Microinstabilities are modes driven unstable by plasma distributions which are not in thermodynamic equilibrium. This generally implies temperature anisotropy, an inverted energy population, or spatial density and temperature gradients. The modes of most concern for TASKA, TDF, and TASKA-M are those driven by ion anisotropy or an ion loss-cone distribution. They are the Alfvén ion-cyclotron mode (AIC), driven by temperature anisotropy, and the loss-cone modes: the drift-cyclotron loss-cone (DCLC) flute mode, its localized version, and the axial loss-cone (ALC) mode.[8] The three devices differ qualitatively in perhaps their greatest way with respect to how microinstabilities are stabilized in their central cells.

In TASKA, the end plug potential is somewhat higher than the ion temperature, so the central cell ions are essentially Maxwellian, and there is no concern over microinstabilities.

In the TDF reference case, the main ion population is created by neutral beams and is mirror-trapped. Microstability comes from a low density, warm ion "stream." This causes a power drain on the hot ions and an alternative TDF scenario, with greatly improved performance, may be available if the stream is not required. Preliminary evidence from TMX-U, although in a vastly different regime, indicates that the stream may be unnecessary. The main microstability concern for TDF is that the central cell neutral beams are injected at 65 degrees to the magnetic field, which creates significant temperature anisotropy. However, although near the boundary, TDF appears to fall within the stable region for the AIC mode, even disregarding stabilizing effects such as finite length.

The stabilizing mechanism for the TMX-U results discussed in the previous paragraph is thought to be similar to that invoked for TASKA-M: injecting neutral beams at an angle can create a "sloshing ion" population, where ions are mirror trapped and ion density peaks at the turning points with a consequent peaking of potential. Warm ions trapped in the potential dip can stabilize at least some of the modes. In order to achieve stability for the DCLC flute mode, TASKA-M required neutral beam injection at 45 degrees in the central cell. Since the warm ions in TASKA-M only exist at the central cell and plug midplanes, there is concern that modes may exist which are localized near or outboard of the sloshing ion peaks. Of the loss-cone modes, only the DCLC flute mode appears to have been seen in experiments. Evaluation of all loss-cone modes would require a computer code which does not exist for the TASKA-M configuration. Because of the 45 degree neutral beam injection, TASKA-M is well within the theoretically stable region for the AIC mode.

In the TASKA end plug, a 60 degree neutral beam injection angle was assumed, without detailed Fokker-Planck analysis, to give a significant sloshing ion population and consequent stabilization of loss-cone modes. The TASKA-M study showed that an angle on the order of 45 degrees is required. Thus, the TASKA end plug would require some modification to achieve microstability -- even for the DCLC flute mode.

Because the warm stream in TDF passes along the whole axis of the machine, the device should be stable to all of the loss-cone modes in the plug also.

For the TASKA-M end plug, the same microstability mechanism was invoked as for the central cell. Because of the lower density, a neutral beam injection angle of 50 degrees sufficed.

Tables 3.A-3 and 3.A-4 summarize these microstability considerations. TASKA would require some modification to achieve microstability. Nevertheless, the changes would probably not unduly impact the total machine design. Its overall degree of risk with respect to microstability is therefore judged to be moderate. TDF, utilizing essentially the same physics criteria as MFTF-B, would have a low degree of overall risk. TASKA-M, where the physics criteria were constrained only by those experimentally verified, would have a moderate risk.

3.A.5 Thermal Barrier Physics

The issues relating to thermal barrier physics are more qualitative in nature than those addressed so far. Both TASKA and TDF utilized neutral beams for thermal barrier pumping, while TASKA-M took the very conservative route of eliminating thermal barriers altogether. TASKA invoked so-called two-stage pumping, which relied only on atomic physics, but is unproven. TDF relied on a more standard neutral beam configuration, but required a hot, mirror-trapped electron population at 400 keV. The uncertainty there relates to the lack of a complete ECRF heating theory at relativistic energies. An interesting post-study development, however, is that TMX-U has shown the existence of thermal barriers working qualitatively as envisioned for TDF, albeit at only 40 keV.(10)

Thermal barrier physics issues are summarized in Table 3.A-5. Because of the complicated nature of the systems generating the thermal barriers in both TASKA and TDF, and because of the uncertain ECRF heating physics, both devices are judged to have a moderate degree of overall risk with respect to thermal barrier issues.

3.A.6 Startup

Without going into the details of the startup scenarios for TASKA, TDF, and TASKA-M, it should be noted that they rely on procedures qualitatively similar to those used for existing experiments -- such as stream guns and ECRF breakdown. The main uncertainty relates to the high complexity of the thermal barrier machines and the question of how to program the power inputs during the startup phase of operation. Both TASKA and TASKA-M used time-dependent computer codes to analyze the necessary startup procedure, with TASKA-M finding that a particularly simple programming of the power sources was required. TDF used a time independent computer code, finding acceptable operating parameters at a series of discrete times -- which also gave confidence that a reasonable startup scenario could be defined.

Table 3.A-6 summarizes startup issues. Because of the high device complexity introduced by the thermal barriers, both TASKA and TDF are judged to have a moderate degree of risk with regard to startup. TASKA-M appears to have a relatively low degree of risk.

References for Appendix 3A

1. N. Hershkowitz et al., "Plasma Potential Control and MHD Stability Experiments in the Phaedrus Tandem Mirror," Tenth International Conference on Plasma Physics and Controlled Nuclear Fusion Research, London, 12 - 19 Sept., 1984, Paper IAEA-CN/C-I-2 (1984).

2. T.C. Simonen et al., Phys. Rev. Lett. 50, 1668 (1983).

3. J. Kesner et al., "A Tandem Mirror With Axisymmetric Central Cell Ion Confinement," MIT Report PFC/JA81-11 (1981).

4. D.E. Baldwin and B.G. Logan, Editors, "Physics Basis for an Axicell Design for the End Plugs of MFTF-B," Lawrence Livermore National Laboratory Report UCID-19359 (1982).

5. B.G. Logan, private communication.

6. J.W. VanDam and Y.C. Lee, in EBT Ring Physics (Proceedings of the Workshop, Oak Ridge, TN, Dec. 1979), ORNL Report CONF-791228 (1980), p. 471.

7. H.L. Berk et al., "Stabilization of an Axisymmetric Tandem Mirror Cell by a Hot Plasma Component," University of Texas at Austin Report IFSR #130 (1984).

8. B.I. Cohen, Editor, "Status of Mirror Fusion Research 1980," Lawrence Livermore National Laboratory Report UCAR-10049-80-Rev. 1 (1980).

9. H.L. Berk et al., "Curvature-Driven Trapped Particle Modes in Tandem Mirrors," Plasma Physics and Controlled Nuclear Fusion Research 1982, STI/PUB/628, zugl. IAEA-CN-41, Vol. II, p. 175.

10. D.P. Grubb et al., Phys. Rev. Lett. 53, 783 (1984).

TABLE 3.A-1

OVERVIEW OF PHYSICS ISSUES

ISSUE	TASKA	TDF	TASKA-M
MHD Stability	Theoretically stable	Theoretically stable	Theoretically stable
a. Uncertainty	Ballooning modes No experimental test	No experimental test	Ballooning modes No experimental test
b. Degree of Risk	Low	Low	Low
Trapped Particle Modes	Theoretically stable	Theoretically stable	Possible m ~ 5-6 instability
a. Uncertainty	No experimental test Theory	No experimental test Theory	No experimental test Theory
b. Degree of Risk	Low	Low	Low
Microstability	End cell design requires modification for stability	Theoretically stable	Does not contradict experiment, but theoretical questions remain
a. Uncertainty	AIC mode ALC mode Localized DCLC	AIC mode	ALC mode Localized DCLC
b. Degree of Risk	Moderate	Low	Moderate
Thermal Barrier Location	Transition	Anchor	Not used
a. Uncertainty	Two-stage NB pumping	Hot electron physics, ECRH	---
b. Degree of Risk	Moderate	Moderate	---
Startup	Complicated--detailed analysis done	Complicated--rough analysis done	Moderate--detailed analysis done
Degree of Risk	Moderate	Moderate	Low
Overall Degree of Physics Risk	Low	Low	Moderate

TABLE 3.A-2

MHD AND TRAPPED PARTICLE MODE STABILITY ISSUES

MODE	TASKA	TDF	TASKA-M
Interchange			
Stability	Stable	Stable	Stable
Calculational Method	STAB code[a]	TEBASCO code	STAB code
Margin	Adequate	Large	Large
Degree of Risk	Low	Low	Low
Ballooning			
Stability	Assumed stable	Stable	Assumed stable
Calculational Method	Operating point is far from interchange boundary	TEBASCO code	Operating point is far from interchange boundary
Margin	Assumed adequate	Large	Assumed adequate
Degree of Risk	Low	Low	Low
Trapped Particle			
Stability	Stable	Stable	(Unstable to m ~ 4-5. E field may stabilize.)
Calculational Method	Analytic (post-study)	Analytic	Analytic
Margin	Adequate	Adequate	Adequate
Degree of Risk	Low	Low	Moderate (no experimental evidence for mode)[b]

(a) R.R. Peterson, "MHD Stability Analysis for TASKA," FPA Report FPA-83-1 (1983).
(b) To be tested in TARA (1985).

Table 3.A-3

MICROSTABILITY ISSUES - I

ISSUE	TASKA	TDF	TASKA-M
Central Cell Stabilizing Method	All ions are essentially Maxwellian	Cold stream	Warm ions trapped in potential dip
Plug Stabilizing Method	Warm ions trapped in potential dip	Cold stream	Warm ions trapped in potential dip
Regions of Concern	Plug	Central cell and plug	Central cell and plug
Overall Degree of Risk	Moderate	Low	Moderate
AIC Mode			
Stability	Probably stable	Probably stable	Stable
Calculational Method	Estimate from Smith's UCRL's (post-study)	Estimate from TMX-U predictions	Estimate from Smith's UCRL's
Margin	Low	Low	Adequate
Degree of Risk	Moderate	Moderate	Low

Table 3.A-4

MICROSTABILITY ISSUES - II

ISSUE	TASKA	TDF	TASKA-M
DCLC Flute Mode			
Stability	Possibly unstable in plug	Stable	Stable
Calculational Method	ND	Estimate from MFTF-B predictions	Analytic
Margin	May require 45° NB injection in plug	Adequate	Low
Degree of Risk	Moderate	Low	Low
DCLC-Localized			
Stability	Possibly unstable in plug	Stable	Possibly unstable beyond sloshing ion peaks
Calculational Method	ND	Estimate from MFTF-B predictions	Estimate from MFTF-B predictions
Margin	Low	Adequate	Low
Degree of Risk	Moderate	Low	Moderate
ALC Mode			
Stability	Possibly unstable in plug	Stable	Possibly unstable in central cell or plug
Calculational Method	ND	Estimate from MFTF-B predictions	Estimate from MFTF-B predictions
Margin	Low	Adequate	Low
Degree of Risk	Moderate (mode not identified in experiments)	Low	Moderate (mode not identified in experiments)

Table 3.A-5

THERMAL BARRIER PHYSICS ISSUES

ISSUE	TASKA	TDF	TASKA-M
System Components	Three neutral beams	Two neutral beams plus ECRH	No thermal barrier
Complexity	High	High	---
Physics Uncertainty	Two-stage NB pumping	Hot electron physics ECRH physics	---
Supporting Experiments	TMX-U	TMX-U	---
Degree of Risk	Moderate	Moderate	---

Table 3.A-6

STARTUP ISSUES

ISSUE	TASKA	TDF	TASKA-M
Complexity	High	High	Moderate
Calculational Method	Time-dependent computer code	Power balance code used at discrete times	Time-dependent computer code
Physics Uncertainty	Moderate	Moderate	Low
Supporting Experiments	TMX, TMX-U	TMX, TMX-U	TMX, Phaedrus, TMX-U
Degree of Risk	Moderate	Moderate	Low

4. PLASMA ENGINEERING

4.1 Introduction

The alternative scenario presented in Chapter 1 raised the question of whether NET can be replaced by the combination of a DT burning tokamak physics device, called NET-P, and a TASKA class tandem mirror facility which would provide testing of components under neutron irradiation. In this scenario, NET-P would provide the physics data required to commit to a demonstration tokamak reactor (DEMO); this would include confinement scaling, ignition and DT burn physics, plasma heating, impurity control, and disruption control. NET-P would presumably be designed for only low neutron fluence operation and therefore would provide little data concerning neutron fluence effects on RF launchers, neutral beam injectors, and other plasma engineering components. The other ingredients of the plasma environment (surface heat load, neutral and charged particle bombardment, electromagnetic effects, etc.) would still be present in NET-P. What would be missing would be extended neutron irradiation effects (i.e., neutron fluence). The question considered in this chapter is whether a TASKA class tandem mirror could provide the required neutron fluence data for plasma engineering components, and thereby, in connection with the plasma engineering data from NET-P, provide a sufficient basis for committing to a DEMO machine.

As sources of data for the plasma engineering parameters for the DEMO, we have used the Argonne Demonstration Reactor (ANL-DEMO)(1) and FINTOR-D.(2) In some cases where these studies did not provide the required data or seemed unrealistic or out-of-date because of newer developments in the field, INTOR(3) data was scaled up to a DEMO, or the STARFIRE(4) study data was scaled down. The data for NET-P was obtained by using the TFCX(5) study as a guide; where this was not adequate, INTOR data was used and scaled down. In this manner we attempted to obtain a picture of both the hypothetical DEMO and the hypothetical NET-P machines. The plasma engineering data for the TASKA class tandem mirror facility were taken from the three studies: TDF,(6) TASKA,(7) and TASKA-M.(8) Since this represents a composite picture taken from three different studies, it follows that the data represents possibilities for a TASKA class facility, but not all of the parameters would necessarily be achieved in a single facility.

4.2 Plasma Wave Heating

To prepare the choice of the optimum heating system for the DEMO an important aspect will be to minimize the number of different auxiliary systems needed for the different functions: heating to ignition, current drive, startup assist, and plasma current profile control. Table 4-1 compares the heating physics for radiofrequency heating and beams. The heating physics roles, wave modes and heating mechanisms together with current supporting experiments in which the various schemes can be tested are noted. There is some commonality in the use of ion cyclotron harmonic heating in central cell heating of current tandem mirror experiments such as TMX-U, TARA and Phaedrus and in the TASKA and TASKA-M tandem mirror reactor test facility designs. Note also that ion cyclotron harmonic heating is being pursued experimentally in numerous current tokamaks at multimegawatt levels[9] as well as in the JET program[10,11] and is considered the primary heating scheme in designs for future reactors such as TFCX and INTOR.

Current ion cyclotron frequency range experiments on PLT are aimed at heating with up to six (6) megawatts using helium-3 and deuterium to simulate D-T ion tail formation and bulk heating that could be accomplished in a reactor. Coupling at 1 megawatt/antenna has been achieved in the ion cyclotron frequency range and considerable attention has been given to minimizing impurity levels coming from the Faraday shield design. Experiments will soon begin on JET which are oriented towards efficient heating at up to 15 megawatts of "high grade" power deposition before the completion of the experimental program. This would indicate that detailed high power heating experiments would be available in the near future which will be adequate for most reactor or fusion test facility designs in the 1990's. It should be noted that both of these large ion cyclotron frequency range experiments can provide important coupling and heating physics information for the TASKA and TASKA-M tandem mirror reactor test facility designs. The use of waves in the ion cyclotron frequency range to heat electrons is proposed for the overdense central cell conditions of TASKA-M. Experiments on TFR and current PLT experiments can provide important information on this mechanism at elevated electron temperatures closer to the startup conditions of TASKA-M.

The use of electron cyclotron frequencies for heating in tandem mirrors is used extensively in TMX-U, TARA and MFTF-B, which should provide adequate information for the TASKA end plug heating design. Current experiments at up

TABLE 4-1. PLASMA HEATING

PARAMETER	UNIT	TASKA	TDF	TASKA-M	TFCX	INTOR
RF						
Role		Ion Heating (C.C.)/Plug Potential	Plug and Thermal Barrier Potential	Electron Heating	Current Drive/ Ion Heating to Ignition	Ion Heating to Ignition
Wave Mode		Fast Magnetosonic/ Ordinary Mode	Ordinary Mode	Fast Magnetosonic	Slow Lower Hybrid/Fast Magnetosonic	Fast Magnetosonic
Heating Mechanism		Deuterium Second Har-monic/Electron Cyclotron Damping	Electron Cyclotron Damping	Electron Landau Damping	Parallel Elec-tron Landau/ Second Ion Cyclotron Har-monic Damping	Second Harmonic Cyclotron Damping on Deuterium
Supporting Expts.		TMX-U, TFR, TARA, Phaedrus, MFTF-B	TMX-U, TARA, MFTF-B	TFR, TARA, Phaedrus, MFTF-B	PLT, ALCATOR-C/ PLT, TFR, JET, ASDEX, TEXTOR, ALCATOR, JFT-II	PLT, TFR, JET, ASDEX, TEXTOR, ALCATOR, JFT-II, ALCATOR-C
Beams						
Role		Potential, Microsta-bility, Beam Power	Potential and Barrier Formation	Potential, Microsta-bility, Beam Power	None	None

to 70 keV barrier electron energies provide useful information on relativistic heating and microstability. However, operation at much higher densities and energies will be required to experimentally verify the barrier operation for reactor-grade machines.

Although recent experiments on lower hybrid current drive on ALCATOR-C and PLT have been impressive, there is still a long way to go to demonstrate a viable reactor efficiency at higher density and electron temperatures. Since this scheme depends on the interaction between a tailored slow lower hybrid wave spectrum and electrons in the slideaway regime, it probably has a very limited role as a potential future electron heating scheme for tandem mirror reactors.

Table 4-2 indicates some of the technology parameters associated with the different radiofrequency heating schemes. It is noted that the ion cyclotron frequency ranges for the tandem mirror designs operate at somewhat lower frequencies than the tokamak designs, primarily due to the higher beta operation for the tandem central cells and the requirement that the wave be below all ion cyclotron resonances in a D-T plasma in order to obtain electron heating in the case of TASKA-M. This can be an advantage due to the efficiency of lower frequency generators but can lead to a more difficult antenna matching design problem than in the case of tokamaks due to the smaller radius of the central cell in the TASKA designs. It is anticipated that similar detailed matching, water cooling and launcher material designs could be employed for tandem mirror and tokamak reactors. It is further noted that antenna neutron flux in the TASKA-M design would provide a comparable test to that anticipated in the TFCX and INTOR designs. With respect to electron cyclotron heating, substantial tandem experiments are underway which should be capable of supplying sufficient information regarding scaling and technology improvements before the tandem reactor test facilities would be constructed. Although there is a current modest tokamak electron cyclotron heating program[12,13] oriented towards bulk heating at higher densities and lower temperatures than those required for tandem operation, they would not provide essential information regarding the fully relativistic heating regime. However, the design of megawatt steady-state tubes which might employ a quasi-optical cavity for output coupling in the TEM mode would provide a useful technological advance advantageous to tandem mirror reactor designs.

TABLE 4-2. PLASMA HEATING

PARAMETER	UNIT	TASKA	TDF	TASKA-M	TFCX	INTOR
Frequency	MHz	$30/5.6 \times 10^4$	$(6.0/3.5) \times 10^4$	15	1500/70	85
Source		Tetrode/ Gyrotron	Gyrotron	Tetrode	Klystron/ Tetrode	Tetrode
Feed		Coax/Quasi-Optical	Overmoded Guide	Coax	wg/Coax	Coax/wg
Launcher (Coolant)		Coils (H_2O)	Overmoded Guide (NAV*)	Coils (H_2O)	Grille/Ridged Guide (H_2O)	Coil/wg (H_2O)
Launcher Materials		Hard Cu/Hard Cu	NAV	Hard Cu/S.S.	Hard Cu/Hard Cu	Al/Cu//S.S.
Nuclear Hardening		Little	Little	Little	Some/Some	Some
Antenna Neutron Flux	MW/m^2	1.5	NAV	0.7/0.1	0.5-0.7	1.3 (1.05 MW Total Heat Load)
Source Efficiency	%	70	NAV	70	70	70
Coupling Efficiency	%	80	NAV	80	95	NAV
Heating Efficiency	%	80	NAV	80	100	100
Absorbed Power in Plasma	MW	40 (C.C.)	1.2	12.4	14/24	50 (10 S)
Barrier/Plug RF Parameters	GHz	56 Plug (15 MW)	60/35	NAP	NAP	NAP
Weak Points		Power handling capacity and neutron/plasma erosion data base.	Power handling capacity and neutron/plasma erosion data base.	Power handling capacity and neutron/plasma erosion data base.	Current drive at higher density & temperature operation.	Power handling capacity and neutron/plasma erosion data base.

* NAV = Not Available

4.3 ICRF Heating

Shown in Table 4-3 are some basic parameters associated with ICRF heating for the DEMO, NET-P, and TASKA class facilities. In this case the tandem mirror data is taken from the TASKA and TASKA-M studies. ICRF heating of the central cell ions is used in TASKA to reduce the required neutral beam power. In TASKA-M, ICRF power is used to heat electrons; the mechanism is Landau damping of the fast magnetosonic wave. In both tandem mirror cases, the technological environment is similar to that in a tokamak with the exception of plasma disruptions; the coil is in close proximity to the plasma, is subject to neutron irradiation and to energetic neutral and charged particle bombardment, and experiences a comparable surface heat load. The power level per launcher is comparable to that of the DEMO, but the frequency in the tandem mirror cases (~ 15-20 MHz) is somewhat lower than in the tokamak DEMO (~ 85 MHz). Because of the smaller plasma radius in the tandem mirror, the physical dimensions of the RF launchers are also considerably smaller. The waveguide option for tokamak launchers was not used for the tandem mirrors because of the lower frequency involved which would make the sizing of vacuum filled launchers prohibitive. However, if higher harmonic $\omega/\omega_{ci} \geq 5$ heating experiments show dominant electron heating, higher frequency waveguide launchers could also be considered for TASKA class facilities.

Note that a TASKA class facility could provide lifetime and transmutation information in a neutron environment for many basic materials used in ICRF heating. Examples would be copper/stainless steel antenna designs measuring basic high frequency conductivity, radiation damage and transmutation effects at high temperature operation (surface temperature $\lesssim$ 1000°C) in a neutron environment comparable to that of a DEMO. Although ICRF waveguide launchers used to heat the TASKA class central cell electrons would be difficult to accommodate in this machine at lower frequencies, a higher frequency compact waveguide launcher operating at 100 MHz suitable for ion heating on a DEMO device could be tested. Testing of first insulator materials for ICRF coaxial feeds where the neutron flux is an order of magnitude below that of the first wall could also readily be tested in a TASKA class device with a useful impact on a DEMO design.

4.4 Lower Hybrid Technology

Application of the power to the plasma using lower hybrid waves is not currently in favor for ion heating in tokamaks, but is under investigation for

Table 4-3. ICRF Parameters for the DEMO, NET-P, and TASKA Class Facilities

	DEMO Class	NET-P Class	TASKA Class
Power	50-100 MW	25	12/40
Frequency	~ 85 MHz	60-100	15/21
Pulse Length	~ 10 s	CW (10 s)	CW
Power/Launcher	~ 15 MW	6	3/5
Launcher Power Dens.	2 kW/cm^2	1	2/1
Surface Heat Load	~ 20 W/cm^2	~ 10	~ 10
Launcher Neutron Wall Loading	2 MW/m^2	0.5-0.7	0.7-1.5
Neutron Fluence	$\lesssim$ 10 $MW\text{-}yr/m^2$	< 0.01	3.5-7.5
Launcher Unit Size	$\gtrsim$ 0.75 m^2	0.6	0.15/0.5
Launcher	Coil or WG	Coil or WG	Coil
Launcher Material	Hardened Cu with SS backing	Hardened Cu with SS backing	Hardened Cu with SS backing

current drive. Since tandem mirrors have no need for current drive and ICRF heating appears better for ion heating, there has been no consideration of a lower hybrid system in a tandem mirror facility. Consequently, it does appear that the TASKA class machines, as currently envisaged, would not provide any data on lower hybrid systems for the DEMO. If this turned out to be a real need however, one could consider putting a lower hybrid system in a TASKA class facility for testing under neutron irradiation. A few basic parameters for lower hybrid systems are shown in Table 4-4. The neutron environment is the same as that for ICRF technology. Consequently, a TASKA class tandem mirror could provide data on the effect of neutron irradiation on lower hybrid components, but using smaller size launchers.

4.5 ECRF Heating

The tokamak machines are currently envisaged to utilize electron cyclotron resonance heating (ECRH, $\omega = \omega_{ce}$) to assist startup of the discharge and thereby reduce the voltage and volt-second requirements of the ohmic heating system. The estimated power required is 10-20 MW at a frequency of 140 GHz for a short pulse (~ 3 s). (See Table 4-5.) Tandem mirrors utilize ECRF (electron cyclotron range of frequencies, $\omega = \omega_{ce}$, $2\omega_{ce}$) to heat electrons and thereby produce the electrostatic potentials needed for confinement. The

Table 4-4. Lower Hybrid Technology Parameters

	DEMO Class	NET-P Class
Power	40-90 MW	14
Frequency	~ 1.7 GHz	1.5-3
Launcher	Grill	Grill
Launcher Power Density	8 kW/cm^2	4
Power/Launcher	5 MW	2.5
Launcher size	10 cm x 2 cm x 32 elements	10 cm x 2 cm x 32 elements

Table 4-5. ECRF Technology Parameters

	DEMO Class	NET-P Class	TASKA Class
Power	10-20 MW	1	1.2/15
Frequency	140 GHz	115	35-60/56
Pulse Length	~ 3 s	~ 3	CW
Launcher	WG/Reflector	WG/Reflector	Quasi-Optical
Launcher Neutron Wall Loading	2 MW/m^2	0.5-0.7	< 0.1 (plug region)

typical power levels in tandem mirrors are less (in TDF), or comparable (in TASKA), depending on the mode of operation, but at a considerably lower frequency (~ 35-60 GHz) and steady-state operation is required. The launcher is generally considered to be a quasi-optical system in the TASKA class machines and both overmoded waveguides and quasi-optical systems have been considered for tokamaks.

Note that the plug region where ECRF is employed in tandem designs such as TASKA or TDF has a much lower neutron flux than that of a tokamak. However, ECRF heating systems for a TASKA-M device could be envisioned for testing of components in the central cell region under comparable neutron flux to that of tokamak DEMO conditions. However, the overdense $\omega_{pe}/\omega_{ce} \gtrsim 1$ condition for the central cell region makes oblique accessibility for ECRF launch quite difficult.

4.6 Neutral Beam Heating

Although neutral beam injection is the primary heating method used in present tokamak experiments, RF heating is favored for reactor applications; neutral beam heating is considered as a backup option. Table 4-6 shows some basic data for the DEMO, NET-P, and TASKA class machines. The DEMO data is taken from FINTOR-D and the NET-P data from the INTOR study. In both cases, positive ion technology is considered, although this is at the upper end of the voltage range for positive ion source neutral beams. Positive ion source neutral beams (~ 80 kV) are used in the TASKA class machines to drive the central cell and/or pump the thermal barrier; the total power injected and the power per beam line are comparable to that in the DEMO and NET-P facilities. The beams driving the central cell are injected into the neutron producing region and hence experience neutron irradiation comparable to that in the DEMO. The neutron fluence required to simulate DEMO conditions is uncertain, since in an ignited tokamak, neutral beams are needed only for heating to ignition. Hence the beam ports can be closed with a shield during the burn phase in a long pulse machine. In this case, the relevant neutron irradiation of the neutral beam source is accumulated rapidly in a tandem mirror where the beams are subjected to constant irradiation.

The TASKA facility also utilized a small (5 MW) atomic hydrogen beam at 250 kV; this requires a negative ion source for efficient neutralization of the beam. This system has no counterpart at present on the tokamak side, but the development of sizable negative ion source neutral beams in the 200-500 kV range would be useful for tokamak reactors because of the improved penetration at higher energy and higher electrical efficiency. Negative ion source neutral beams have been suggested for current drive in tokamaks.

4.7 Particle Fueling and Impurity Control

Particle refueling appears to have received little attention in the DEMO studies and in the various studies used as guides for NET-P (see Table 4-5). The INTOR study specifies a pellet velocity of 2 km/s, but the required pellet size and injection repetition rate are unspecified. The ANL-DEMO and FINTOR-D studies are even less specific and merely state that the plasma will be refueled by either pellet injection or gas puffing. Pellet injection is utilized in one mode of operation of TDF to fuel the warm plasma in the central cell. The parameters are comparable to what one might utilize in a tokamak reactor.

Table 4-6. Neutral Beam Technology Parameters

	DEMO Class	NET-P Class	TASKA Class
Voltage	~ 160 kV	175	80/250
Power	100-300 MW	50-75	65/5
Power/Beam Line	18 MW	15	8/2.5
Pulse Length	~ 10-20 s	~ 5	CW
Neutron Wall Loading at Beam Port	2 MW/m^2	0.5-0.7	0.7-1.5

Table 4-7. Particle Refueling Parameters

	DEMO Class	NET-P Class	TASKA Class
Method	Gas Puff &/or Pellet	Gas Puff & Pellet	Pellet
Pellet Velocity	?	2 km/s	1.5
Pellet Repetition Rate	?	?	500 s^{-1}

Impurity control and first wall protection against disruptions are important plasma engineering concerns in tokamak reactors. Disruptions have no counterpart in tandem mirrors, but impurity control is considered in the form of a plasma halo (analogous to the scrape-off layer in tokamaks). The halo plasma dumps into a halo scraper which has a distinct similarity to a pumped limiter or divertor chamber in tokamaks. The halo scraper is located in the end region and therefore is subject to much less neutron irradiation than a tokamak pumped limiter or divertor. Consequently, it does not appear that a TASKA class machine would provide useful neutron irradiation data for impurity control components in a tokamak DEMO. The question for the DEMO becomes: Where does one get experience with impurity control options in a neutron irradiation environment before committing to a DEMO? NET-P would test components under appropriate surface heat load and plasma bombardment conditions, but only at low neutron fluence.

The neutral beam dumps and central cell region near where the neutral beams intersect the plasma in a tandem mirror experience high heat loads. This is because of the beam shine-through and the charge exchange flux emitted

Table 4-8. Contribution of NET-P and a TASKA Class Machine for the Heating Technology Needs of the DEMO

Needs for DEMO	NET-P	TASKA Class
Applicability of RF Systems to		
Plasma heating (ICRF, ECRF)	• possible	• possible
Startup assist	• possible	• possible
Profile control	• possible	• possible
Launchers		
ICRF-Loop antennas: power density, lifetime (n,γ; sputtering), impurity generation, phased waveguides	• comparable conditions but low n fluence	• comparable power and n fluence • smaller size • 1/4 frequency (testing of higher frequency component possible)
LHH-Grill structures	• comparable frequency, 1/4 power, low n fluence	• fluence testing of components conceivable
ECRF: Waveguides or quasi-optical guides; gyrotrons (~ 1 MW,CW)	• comparable frequency, 1/10 power	• comparable power 1/2 frequency
Power Units for CW Operation	• testing possible	• testing feasible
RF-Windows and Insulators (loss tangent, dielectric breakdown, lifetime)	• required for operation	• testing at high high power under neutron irradiation conceivable
RF-Components (transmission lines, etc.)	• testing possible	• testing possible
RF-System in Reactor Environment	• not possible	• possible
Neutral Beam Heating	• comparable power and energy, but low n fluence	• comparable power and n fluence, 1/2 energy

from the plasma in this region. This heat flux has been evaluated for TASKA-M. The heat flux on the beam dumps is 250 W/cm^2 (oblique incidence) and on the central cell insert is 70 W/cm^2 in a region where the neutron wall loading is 0.7 MW/m^2. The local heat flux can be as high as 400 W/cm^2 on surfaces closer to the plasma. This provides an opportunity for testing high heat flux components in a neutron environment. In this case the surface heating is provided by energetic neutral atoms (energy ~ 50-100 keV). This region could be used for testing tokamak limiter/divertor target plates at reduced sizes with simultaneous surface heating and neutron irradiation. The surface erosion rate, however, is not representative of tokamak conditions because of the different energy spectrum of the incident atom flux.

4.8 Conclusions

Table 4-8 summarizes the contributions of a NET-P facility and a TASKA class facility to the heating needs of the DEMO. It indicates that this combination can provide most of the heating technology data base for the DEMO. The NET-P facility best simulates the physics environment, but only at low neutron fluence. The TASKA class machine utilizes comparable heating technology in a somewhat different physics environment, but similar neutron environment.

The smaller size and different geometry of a TASKA class facility, compared to a tokamak DEMO, means that full-size testing of plasma engineering components is not possible. Size scaling must be used in extrapolating from a TASKA class test to a DEMO design, but the experience in the NET-P device reduces the uncertainties in this extrapolation.

Because of the different physics requirements of tandem mirrors, tokamak relevant impurity control testing in a TASKA class facility does not appear to be feasible. Information about the behavior of high heat flux components, such as beam dumps and central cell inserts, can be obtained in a TASKA class facility; this may provide useful information for the design of a limiter or divertor target plate for a DEMO. The energy spectrum of the particles incident of the high heat flux surfaces is peaked, however, at much higher energy than that expected in a tokamak DEMO.

References for Chapter 4

1. M. Abdou et al., "A Demonstration Tokamak Power Plant Study (DEMO)," Argonne National Laboratory Report ANL/FPP/82-1 (1982).

2. "FINTOR-D, A Demonstration Tokamak Power Reactor," edited by W. Izzo and G. Realini, CEC Report EUR-7322-EN (1981).

3. INTOR Group, "International Tokamak Reactor, Phase Two A, Part 1," International Atomic Energy Agency, Vienna (1983).

4. C.C. Baker et al., "STARFIRE - A Commercial Tokamak Fusion Power Plant Study," Argonne National Laboratory Report ANL/FPP/80-1 (1980).

5. "TFCX Preconceptual Design Report," Princeton Plasma Physics Laboratory Report F-Axxx-8407-006 (1984).

6. T.H. Batzer et al., "A Tandem Mirror Technology Demonstration Facility," Lawrence Livermore National Lab. Report UCID-19328 (1983).

7. B. Badger, F. Arendt, H.M. Attaya, et al., "TASKA - A Tandem Mirror Fusion Engineering Test Facility," Kernforschungszentrum Karlsruhe Report KfK-3311 (1982).

8. B. Badger, F. Arendt, H.M. Attaya, et al., "TASKA-M, A Low Cost, Near Term Tandem Mirror Device for Fusion Technology Testing," Kernforschungszentrum Karlsruhe Report KfK-3680 (1984).

9. E. Mazzucato, M. Bell, B. Bittler, et al., "Ion Cyclotron Heating Experiments in PLT," 10th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research, London, 12-19 Sept. 1984, paper F1-2.

10. V.P. Bhatnagal et al., "Optimization of ICRH Power-Deposition Profiles in INTOR Tokamak by $k_{\parallel}$-Shaping," Proc. 4th Int. Symp. on Heating in Toroidal Plasmas, Rome, March 1984, Vol. I, p. 291.

11. P.O. Lallia et al., "Possible ICRF Heating Modes of Operation in Non-active JET Plasma," Proc. 4th Int. Symp. on Heating in Toroidal Plasmas, Rome, March 1984, Vol. I, p. 393.

12. J. Thumm et al., "ECRH Systems for Linearly Polarized Plasma Irradiation in the HEII Mode at 28 and 70 GHz," Proc. 4th Int. Symp. on Heating in Toroidal Plasmas, Rome, March 1984, Vol. II, p. 1461.

13. R. Cano, "Introduction and Summary of the Panel on ECRH Power Sources and Applications," Proc. 4th Int. Symp. on Heating in Toroidal Plasmas, Rome, March 1984, Vol. II, p. 1413.

5. BLANKETS

5.1 Introduction

There are many blanket technology issues that need to be examined carefully before proceeding to build a fusion reactor demo. Some of these issues are critical to the blanket design as they might impact the feasibility of a particular blanket concept as well as affect the overall blanket performance with subsequent effects on cost and environment. Proper testing of these issues will require careful simulation of the parameters that affect them. Since most of these issues depend on more than one environmental condition, the proper simultaneous duplication of the conditions that impact the blanket issue to be tested, is essential.

Several blanket concepts or concept variations have been proposed. Each one has somewhat different critical issues and test needs. However, in a broad sense one can divide the blanket concepts into two categories; ceramic breeder blankets and liquid metal breeder blankets. In the next two sections, the critical issues for each of these blanket categories are identified and the environmental conditions that impact each issue are determined. These conditions must be properly simulated in a meaningful test of that particular issue. The ability of different test facilities to meet the test needs for each issue is assessed. Both fusion and non-fusion test facilities are considered. Non-fusion test facilities include non-nuclear test stands, accelerator-based point source facilities, and fission reactors.

A comparison between the testing capability of the different tandem mirror fusion technology test facilities is given in Section 3.3. The blanket parameters obtained in "NET-P" and "TASKA" class facilities are compared to the parameters required in a tokamak demo in Section 5.4. Conclusions related to the capability of different test facilities for integrated blanket testing are given at the end of this chapter.

5.2. Testing of Ceramic Breeder Blankets

A large number of ceramic breeder materials and solid breeder blanket concepts has been proposed. But the material data base for the ceramics is far from being sufficient for a feasibility assessment and concept selection. Before this can be done several points have to be assured including the following:

1. Enough tritium is produced in the blanket to compensate tritium consumption and losses.
2. Tritium release from the blanket and in particular from the ceramic material is fast enough to keep the inventory small.
3. Tritium losses from the plant are below safety limits.
4. Heat can be removed at the design temperatures and all material temperatures stay within the design limits.
5. The breeder material is physically and chemically stable.
6. The breeder material is compatible with coolant and structure.
7. Blanket performance is not jeopardized by radiation damage.

In view of the limited resources and the large numbers of options material and concept selection has to follow a step by step procedure starting from the most fundmental issues and small scale screening tests to go after candidate selection to large scale interactive effects tests under conditions of a specific design concept. In the course of such a program different test facilities may be needed at different steps.

Good understanding of the basic phenomena is extremly important for such a procedure in order to avoid wrong selection of candidate materials and concepts from early screening tests which may not be done under proper environmental conditions.

5.2.1. Evaluation of Possible Testing Facilities

The possible testing facilities can be categorized into:

1. non-nuclear test stands
2. point neutron sources
3. fission reactors
4. fusion test reactors

In case of fission reactors one has to distinguish between existing facilities and facilities which have to be modified or specially constructed for the fusion reactor tests.

Non-nuclear test stands will be needed for issues such as
- ceramic material property data base
- ceramic material fabricability problems
- thermal hydraulic tests
- tritium migration and tightness of circuits

The degree of simulation is essentially limited by the lack of tritium production in the ceramic, radiation effects and a volumetric heat source.

These effects can be simulated quite well for small samples in fission reactor irradiations. In larger samples such as blanket submodules of about 15 cm diameter the power density and neutron flux gradients in the blanket are quite important. In fission reactor irradiations they can only be simulated when a test assembly is located at the plane surface of a special slab-type reactor. The degree of simulation is limited.

Insertion of a test module between the coils of a NET-P facility would give the most realistic simulation of the geometrical and radiation field conditions of a tokamak-DEMO. But the low power density and short operation periods will not lead to a correct temperature distribution in the test module and no radiation damage effects will occur.

In this respect a blanket test module at a TASKA type facility would be superior. But the difference in geometry makes it necessary to carefully design the test module in order to get Tokamak relevant conditions on a submodule scale.

In the next sections the role of different test facilities for nuclear tests to solve critical issues of ceramic breeder blanket concepts will be examined.

5.2.2. Tritium Production

In view of the large number of nuclear reactions involved in a fusion reactor blanket, the strong space, angular, and energy dependence of the neutron spectrum, the very large and complex geometry, and the many materials not used in fission reactors, integral tests have to be examined to check the neutronic calculations. While meaningful code verification experiments can be proposed, it is very difficult to conduct experiments to verify tritium self sufficiency.

In fission reactors the situation is much simpler: A self-sustaining neutron chain reaction allows easily to adjust the neutron flux level to the desired value. The nearly homogeneous fission source distribution makes the neutron flux and spectrum only weakly space dependent. Nevertheless, about 1400 manyears and facilities of about 70 M$ were spent in Europe for LMFBR neutronic experiments with supposedly a similar effort in the USA.

The need for a large surface strong 14 MeV neutron source together with the above mentioned problem makes the situation in fusion much more difficult. A first analysis at KfK showed that even the very simple experiment of a point source in the center of a beryllium sphere shell and measurement of neutron multiplication as a function of shell thickness does not lead to results which can easily be interpreted. The situation will be much more involved for complex blanket type arrangements. Thus, instead of error identification one can only compare measurement and calculation in a blanket mock up arrangement.

The experiments presently under way or planned at Tokai Mura, Osaka Universitiy, and Lausanne with a 14 MeV point source in front of a 1m x 1m slab arrangement will certainly improve our knowledge but supposedly will not lead to the required accuracy in tritium breeding rate prediction.

With NET-P a large surface 14 MeV neutron source of sufficient intensity for tritium breeding rate measurement and of typical tokamak geometry would be available. However, neutronic calculations have shown that at the blanket surface the intensity of neutrons scattered back from the walls of the tokamak is several times higher than the original 14 MeV neutron flux, depending on blanket material and first wall geometry. Thus, the quality of the experiment will depend on the first wall design of NET-P and the capability to correct for the difference to a situation where the whole surface is covered by blankets. Because breeding rate experiments need no high fluence, a TASKA-class machine would not provide much more information than NET-P.

5.2.3. Tritium Release

Tritium release determination calls for an irradiation test with flowing purge gas and in situ tritium measurement.
All blanket concepts with ceramic breeder foresee periodic arrangements of breeder material in rods, plates, or shpere pac in relatively small units. Tritium release from these units can be measured in fission reactor irradiations. In thermal reactors the ^{6}Li concentration should be kept low enough to avoid neutron flux depression in the probe. Cadmium filtering flattens the power distribution but may not always be possible because of its large reactivity reduction. More important than the correct neutron spectrum are

- clean and well defined experimental conditions
- good instrumentation
- a broad enough program to identify the best solution (ceramic material, its structure, fabrication process, purity requirements etc.)
- improvements in model and computer code developement to describe the tritium migration and interaction effects.

The program can and will be conducted in existing facilities.

5.2.4. Radiation Damage

In order to qualify materials for the use in a DEMO, they have to be irradiated to fluences which are expected to occur in the DEMO in a neutron spectrum which simulates sufficiently well the DEMO conditions.

The first question to adress in this context is: How well can existing fission reactors do the job ? To answer this question a study has been conducted to compare the irradiation conditions of ceramic breeder materials in a fusion reactor blanket with those obtainable in test sample irradiations in fission reactors. The most important parameters to be simulated in the irradiation are:

1. Tritium production rate with a corresponding damage rate caused by the tritium and α particles.
2. Radiation damage resulting from fast neutron elastic collisions,
3. power density in the ceramic material,
4. temperature in the test sample

The proper sample temperature can generally be adjusted by the coolant conditions. In fast reactor irradiations it may sometimes be difficult to get sufficiently low temperatures.

Because of the higher gamma radiation contribution the power density is higher in fission reactor irradiations than it would be in a fusion reactor blanket of the same tritium production rate. This effect is of the order of 20 % and not important for small sample irradiations. Thus, the problem is reduced to achieving the same damage rates and tritium production rates.

Damage rate calculations were based on the NRT-model /1/ which was modified to treat compositions of different kinds of atoms. Calculations were made for various blankets, test reactors, and ceramic materials. Fig. 5.2-1 shows for illustration the damage rates in Li_2SiO_3 in dpa per full power year (FPY). Plotted is the damage rate caused by the fast neutron elastic collisions versus the damage rate caused by the tritium and alpha recoil particles. The latter is strictly proportional to the tritium production rate in atoms/s per gram of ceramic material with

$$\mathrm{TBR/g\ Li_2SiO_3} = 6.6 \times 10^{12} \left(\frac{\mathrm{dpa}}{\mathrm{FPY}}\right)_{t+\alpha}$$

Each point in the plot corresponds to an irradiation position or position in a blanket. The areas labeled NET-P and DEMO cover the different blankets and different positions in the blankets which were considered and refer to NET (1.3 MW/m^2 wall load) or a similar DEMO with 3 MW/m^2 wall load. The upper left corner refers to the front part of a blanket, the lower right corner refers to the back side of a blanket with relatively high hydrogen content to improve the breeding ratio.

When changing the ^{6}Li-enrichment of a sample in a fast fission reactor irradiation, the $(t+\alpha)$-generated dpa-rate can be varied without changing the elastic collision dpa-rate. In thermal reactors this possibility is more limited due to neutron self shielding effects. The end points in the graphs of Fig. 5.2-1 indicate the values of natural and fully ^{6}Li-enriched lithium. In this way various areas for irradiation tests in fission reactors are obtained.

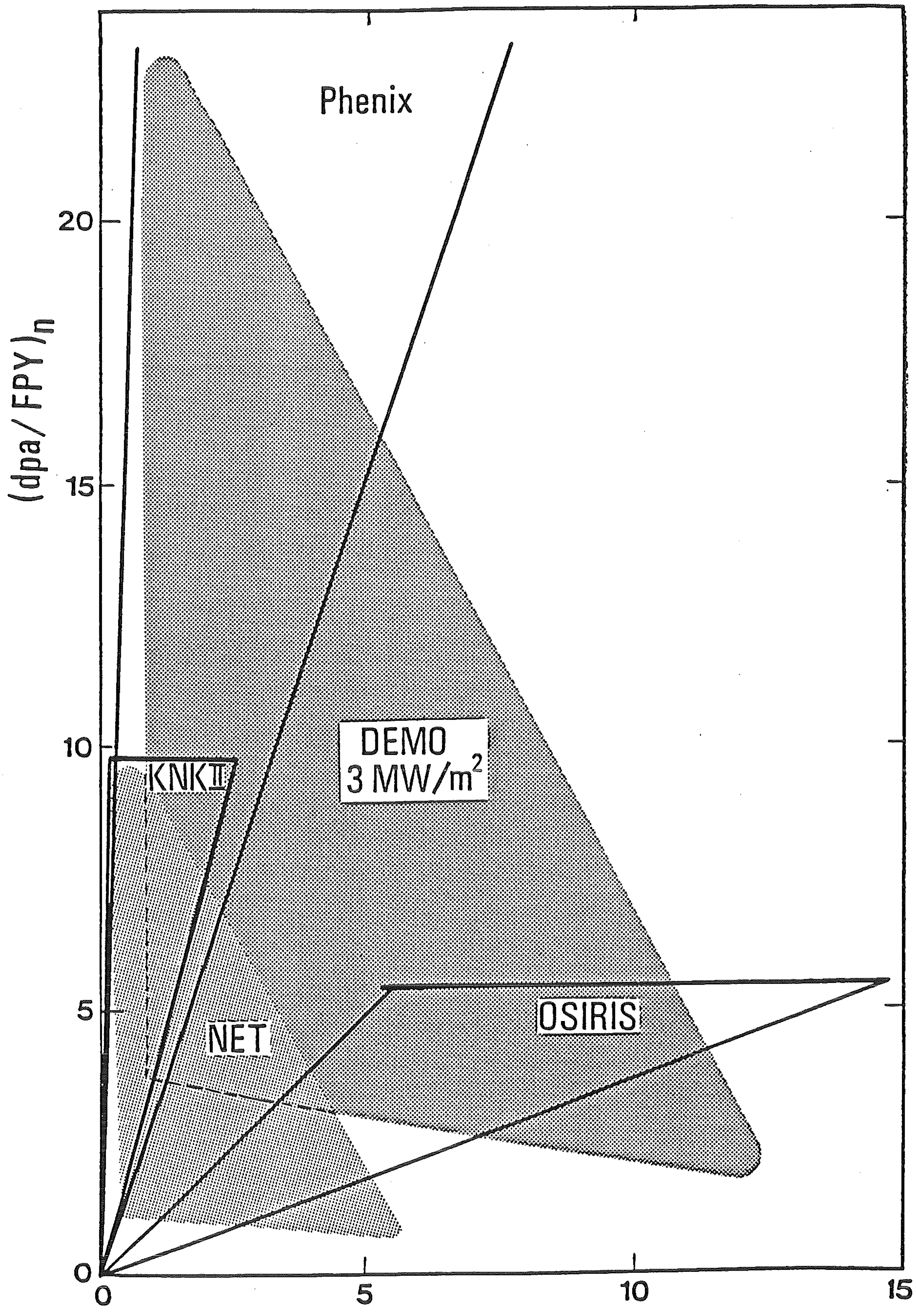


Fig. 5.2-1

Radiation damage rates in Li_2SiO_3 caused by elastic neutron collision vs. those caused by the $^6Li(n,\alpha)T$ process. Comparison of test reactor conditions with fusion reactor blanket.

The reactors should be considered as examples of a class: Phenix for fast breeder, KNK II for a small fast experimental reactor, OSIRIS for a high performance thermal reactor. Although the study is far from being complete one gets the impression that small sample test conditions representative for blankets in NET or DEMO can in principle be obtained with existing fission reactors. In practice, however, it may be quite difficult to install irradiation loops in fast reactors.

The range of the primary knock on atom from 14 MeV neutron scattering on lithium is similar to the range of the alpha and tritium recoils from the ^{6}Li (n,α)T-reaction. Thus, radiation damage from the two processes is supposedly also similar. In view of this, simultaneous adjustment of both fast neutron scattering and ^{6}Li (n,α)T-reaction rates is a rather puristic requirement. In most cases generation of the same total damage rate would be sufficient. If this turns out to be correct thermal reactor irradiations could satisfy all needs for NET. Fast reactor irradiations will be required for a DEMO.

Atomic transmutations in ceramic breeder material which occur in the fusion reactor but not in a fission reactor are (n,α) and (n,p) processes on silicium, aluminium, and oxigen. The (n,p)-reactions always lead to β-active nuclides which go back to the original species very fast so that they can be neglected. An estimate of the (n,α)-reactions on O,Si, Al in a fusion reactor blanket showed that they are always less than 1 % of the ^{6}Li (n,α)-reaction. This dominating transmutation process is well simulated in a fission reactor spectrum.

The fact that in principle fission reactor irradiations can simulate fusion reactor conditions quite well, does not mean that they are simple, cheap, of adequate test volume, flexible in boundary conditions and immediately available. Whereas the irradiation of small probes in capsules should generally be possible, larger samples with realistic coolant and purge gas conditions - including clad compatibility tests - are not so easy to introduce in the existing fast reactors. This refers especially to water cooled or relatively low temperature conditions.

Therefore, tests which go to high damage levels should primarily be done with small probes in fast reactors. Interactive and integral tests should be done in thermal reactors and may sometimes require a modification of the reactor.

It is not clear at the moment how important the radiation damage effects really are. There are some indications that the diffusion coefficient in ceramic materials is much larger than previously assumed /2/. Then one could use a sphere pac of ceramic material of nearly theoretical density. In that case open porosity is not used to extract the tritium, and closing of pores by radiation damage coupled with thermal transport - what is presently the primary concern - would be obsolete. In some years from now we problably will have a clearer picture of the test requirements for radiation damage in ceramic materials and the test needs should be reconsidered.

Radiation damage in multipliers also is of great importance for the blanket design. The dominating effect for beryllium is the helium production by (n,2n)-reactions. In spite of the reaction threshold at 1.8. MeV the neutron flux in fission reactors is high enough to get adequate beryllium (n.2n) reaction rates.

In a fast reactor like Phenix, the helium production rate in beryllium is about the same as in the front part of a fusion reactor blanket with 1.3 MW/m^2 wall load. In a thermal reactor such as OSIRIS it is half that value.

For lead the (n,α)- and (n,p) cross-section at 14 MeV are in the millibarn range. A simulation of the radiation damage effects in a fission reactor is not possible. But the low melting point of lead makes its use in solid form questionable anyway.

5.2.5. Interactive Effects

There are some interactive effects such as clad breeder interaction which can be investigated in irradiation probes of a few cm diameter. Others require submodule testing. These are generally the problems where the strong neutron flux and power gradients of a blanket play an important role. Therfore, we have to look for irradiation tests where these can be simulated and geometrical dimensions are at least large enough to irridiate a typical blanket structure element (a few rods or plates). The objectives and needs of such tests are less clear than those for basic data and radiation damage effects. In addition, they are strongly design dependent.

To get a first impression of the possibilities, some one-dimensional neutron transport calculations were made with a slab type test assembly in front of a fission reactor surface. A helium cooled ceramic breeder with beryllium multiplier mixed to the ceramic was used as the blanket references case /3/.

The flat surface of a thermal Materials Testing Reactor (MTR-type) was used as a neutron source. With 20 cm core thickness, 0,8 m active height and 0,8 m width the reactor was critical with the test assembly in front of it. To avoid a steep decay of the power density in the first few centimeters of the blanket mock-up, a 1cm B_4C + 10cm Pb decoupler was placed between reactor and blanket.

The reactor power was adjusted to give at the test assembly surface the same tritium production rate as a fusion reactor with 1.3 MW/m^2 wall loading. This was 122 MW and an average power density of 953 kW/l . A reactor with these specifications does not exist but should be technically feasible.

The tritium production profile in the test assembly was over a distance of 20 cm quite similar to that of the corresponding fusion reactor blanket. This may partly be due to the one-dimensional calculation and could be worse in reality. Moreover the power density in the test assembly was lower by a factor of two when compared with the blanket. This results from the lack of 14 MeV neutrons in the fission neutron spectrum so that less energy is transmitted in the slowing down process.

In NET-P all conditions can be perfectly simulated but the wall loading is too low by a factor of three compared with a DEMO. In addition, the short periods of operation may not lead to thermal equilibrium.

TASKA has about twice the wall loading of NET-P but the geometry is quite different from a tokamak. This difference has the surprising effect that tritium production rate and power density in the blanket are lower by nearly a factor of two when compared with a system in tokamak geometry and the same wall loading.

The main advantages of TASKA compared with NET-P would be the higher availability, the easier access and facility operation primarily for such tests.

All three test-options (fission reactors, TASKA-class, NET-P class) suffer from insufficient power density and tritium production rate. However, in fusion reactor blankets the power density is quite low and heat transfer problems do not play the important role they did for fission reactors. Therefore power density simulation may not be needed and the correct temperature can be adjusted by a lower flow rate. It is also evident that the correct tritium rate is necessary for the tests. Other aspects such as flexibility, cost, access, volume, control of test conditions may be more important. Thus, the objectives of interactive tests have to be defined much better before a meaningful assessment of the best suited test facility can be made.

5.2.6. Summary for Ceramic Breeder Blankets

In contrast to steel and other materials ceramic breeder irradiation tests can be performed quite well in fission reactors with good simulation of the fusion reactor conditions up to fluences relevant for DEMO. This will allow to develop with existing test facilities the data base for the candidate solid breeder materials so that appropriate breeder concepts can be developed.

Performance tests of blanket concepts require larger test volumes and steep flux gradients which to a certain extent can be obtained with slab type fission reactors, with NET-P and with a TASKA-type facility. Although the test objectives are quite vague at present there seems to be a good chance that these facilities together with an extensive non-nuclear test program will deliver enough information to go to a DEMO as the next step.

5.3 Testing of Liquid Metal Blankets

Understanding the complex behavior of liquid metal blankets for fusion reactors is made more difficult by the fact that the various phenomena and environmental conditions which they will experience are highly interactive, contributing to large uncertainties in their performance. Useful tests must reproduce the environmental conditions of a real reactor such as geometry, size, surface area, magnetic field, surface heat flux and nuclear bulk heating. Full scale testing of DEMO blankets is not possible and, therefore, scaled down models with reduced parameters will have to be used. Careful scaling of a test module is essential to ensure that it will act like a full scale reactor blanket module. Detailed studies are presently under way in the FINESSE[4] program to define the approach for appropriate scaling of test modules.

As far as the breeding materials themselves are concerned, liquid metals are not subject to radiation damage and nuclear tests requirements are less stringent than for solid breeding materials. The critical issues which need resolution for liquid metal blankets are:

1. Corrosion and structural behavior.
2. Magnetohydrodynamic (MHD) effects.
3. Thermal hydraulics.
4. Tritium production, diffusion and extraction.

The facilities in which some or all of these critical issues can be tested are:

1. Non-neutron test stands.
2. Point neutron sources.
3. Fission reactors.
4. Fusion test reactors.

In the next section a brief description and evaluation of these facilities will be made.

5.3.1 Evaluation of Possible Testing Facilities

There is no question that a large amount of very relevant information on the development of liquid metal blankets can be obtained from non-fusion facilities and test stands. Non-neutron facilities can provide data on single effects and some possible multiple effects which can be valuable in initial material screening and the determination of some operational limits at a relatively low cost. These experiments would primarily be in the area of corrosion, MHD effects, thermal hydraulics and possibly, tritium extraction.

Point neutron sources are severely handicapped by a low test volume, low fluence, incorrect spectra and the operation of complicated devices such as accelerators. In all truthfulness, they can only provide single effect materials and neutronic testing in a limited way. Point neutron sources can be virtually ruled out for any kind of integrated testing of blanket assemblies or subassemblies.

It is interesting to examine the possibility of testing in fission reactors. While it is true that fission neutrons and gammas can produce bulk heating in structures and breeding media, other limitations severely curtail truly integrated testing of blankets. The most severe limitation is the test volume available in a fission reactor and the relatively low flux in many of them. For example, there are no US reactors and only 16 worldwide which can provide an in-core space of 15cm diameter with a flux of 5×10^{14} n/cm^2-s[(1)] (which for a water moderated plate fueled test reactor is roughly equivalent to 1 MW/m^2 in a fusion reactor). Other limitations such as incorrect spectra, negative reactivity effects, the lack of surface wall heating and magnetic fields attendant in fusion blankets also makes the testing more difficult.

Finally there is testing in fusion test reactors. A TASKA class facility is expected to provide a fully integrated test capability for appropriately scaled down blanket modules suitable for use in a tandem mirror DEMO. Such a facility, however, falls short of providing all the requirements of a tokamak DEMO which has a different geometry blanket, a different magnetic field profile and higher surface wall heating. A NET-P facility, on the other hand, can provide an integrated test bed for a tokamak DEMO blanket for tests where flux and not fluence is the important criterion. This would include tests on thermal hydraulics, MHD and some limited tritium experience. A TASKA class facility, however, provides fluence which can give extensive experience on corrosion and tritium handling for both tokamak and mirror DEMOs.

In the following sections we will go through each of the critical issues, namely corrosion and structural behavior, MHD, thermal hydraulics and tritium considerations, and will attempt to identify which facility can contribute toward partial or full resolution of the issue.

5.3.2 Liquid Metal Corrosion and Structural Behavior

Corrosion is a severe problem in reactors in general and is of paramount importance for liquid metal systems. The problem manifests itself in three ways:

1. Thinning of structural elements.
2. Corrosion transport of radioactive material.
3. Potential plugging of valves, orifices, etc.

The first problem involves more than just uniform dissolution of any structural material which comes in contact with a liquid metal. It can lead to alloying, intergranular penetration, interstitial impurity transfer to or from the liquid metal and preferential leaching out of certain elements in the structure, thus changing its overall composition.

Because the blanket structure becomes activated when exposed to a fusion environment, the transport and redeposition of radioactive corrosion products becomes a problem from the standpoint of maintenance. Usually, the corrosion product is deposited in the colder region of the liquid metal loop such as in the steam generator. This brings up the final problem of plugging. Redeposition of corrosion products depends on many factors and takes different forms. For example, chromium deposits as needle-like metallic crystals which because of their high surface to volume ratio can begin to impede the flow of liquid metal.

Although experience with liquid metal research for the LMFBR program is very valuable, it is generally agreed that the presence of a magnetic field in a fusion reactor and a high thermal gradient at the first wall can change the corrosion mechanism sufficiently to make this information of limited use. Further, the bulk of data from the LMFBR program is on materials which are not suitable for use in fusion.

Although there are several liquid metal corrosion loops presently operating in the world, most of them address single or at most, double effects only. For a meaningful integrated effects test, a loop must have a magnetic field of 2-4 tesla with flow in both parallel and transverse field directions, surface heating to simulate radiant heat transfer to the blanket and bulk heating with an established thermal gradient to simulate nuclear heating. The channel length must be long enough to ensure that fully developed flow is reached. The requirements of a magnetic field and surface heating can, in principle, be satisfied without neutrons. Although bulk heating can be simulated by inductive means it would be very difficult to implement on a test module particularly with correct thermal gradients.

The effect of a magnetic field is to laminarize the flow and to reduce the boundary layer thickness. The first effect tends to inhibit corrosion, while the latter, to enhance it. Diffusion and transport of dissolved material depends on the solubility of the corrosion product, which is heavily temperature dependent. Which of these effects will dominate and to what extent are there mutual interactive influences can only be determined in a test which can simulate all of these conditions simultaneously, and can be operated for periods of up to several thousand hours.

Although some very meaningful results can be obtained from corrosion loops without neutrons, it is felt that the establishment of a correct temperature profile is sufficiently important as to require ultimate testing of blanket subassemblies in the presence of neutrons.

Point neutron sources, due to the small size of the test volume, are useless in this case. Fission reactors can provide bulk heating (albeit with a wrong spectrum) if sufficient test volume can be found in the core. Providing a magnetic field in a fission core, with sufficient length to reach fully developed flow, would require a space of at least 50 x 40 cm. Even if it could be done there are presently no operating reactors in the world which can provide that. Some reactors can provide slab type locations on the side of the core, with 5 in the world[(1)] which can accommodate a 50 cm slab. Here the flux is considerably depressed and will be even further depressed by the needed magnet structure. The usefulness of such an experiment is, therefore, limited. Further, there are safety issues related to having stray magnetic fields and high power leads so close to a fission core.

Only a TASKA class facility can provide the requirements for integrated corrosion testing of a liquid metal blanket with all the synergistic effects of a fusion environment, and most importantly, fluence. Although the size of the module will have to be smaller, an appropriately scaled down module can be used for corrosion tests.

Radiation induced structural changes such as swelling, gas production, atom displacement and transmutation can be studied in point neutron sources. However, realistic forces from magnetic fields and other sources can only be obtained under actual operation. Results of radiation, such as its effect on yield strength, creep, ductility, embrittlement and fracture toughness and how they relate to the end of life of a blanket will have to be evaluated under such loads. Again in this case, only a TASKA class facility can provide appropriate conditions even if the magnetic field profile differs from that of a tokamak.

5.3.3 MHD Effects

Magnetohydrodynamic (MHD) problems exist in self-cooled liquid metal blankets and translate into the following:

1. Excessive pressure drop resulting in a high system pressure and high pumping power or increased blanket stresses.
2. Uneven flow distribution with possible resulting "hot spots" or preferential corrosion sites.

Present capability for modeling MHD is very limited and the problem is extremely complicated because it involves the solution of three-dimensional coupled electromagnetic and fluid dynamics equations. Thus far, experiments have been limited to flow in straight pipes in relatively low uniform magnetic fields. As part of the US DOE Blanket Technology Program a facility is currently being built at ANL consisting of a NaK loop and a 2 tesla split frame normal magnet with a pole face of 0.8 m x 1.9 m and a separation of 0.2 m. The facility will operate in early 1985 and provide a flow of 1000-1200 ℓ/m and can therefore test MHD effects in fairly large subsystems. However, this test stand will not be capable of duplicating the magnetic field profile in a tokamak which can have both toroidal and poloidal field components. Further, it is not clear what effects bulk heating and temperature gradients may have on MHD. Since the electrical resistivity of the liquid metal varies with temperature, it can in principle affect the eddy currents generated and in turn, influence MHD.

The approach to this problem should proceed on several fronts: development of MHD codes which will help understand the basic phenomena with input from test stands like the one being built at ANL, and the implementation of new test loops which are better capable of simulating realistic field profiles.

Fission reactors cannot make a contribution here for the same reason as in the case of corrosion. There simply are no fission reactors which can accommodate a large magnet.

A TASKA class facility can duplicate most of the conditions needed to test MHD effects in a liquid metal blanket. The limitations of such a facility with respect to providing data for a tokamak DEMO are that the geometry and the magnetic field profiles are different. However, bulk heating by neutrons will lead to realistic thermal gradients and thermally equillibrated flow. Since a large neutron fluence is not needed for MHD tests, a NET-P class facility will be able to fully test these effects if a liquid metal blanket module can be incorporated into it.

5.3.4 Thermal Hydraulics

One of the most important parameters in a liquid metal fusion blanket is the temperature of the structure and breeding material. While it is important to maximize the temperature to improve the power cycle efficiency, this must be done within the material operating limits. The structural integrity of the blanket, its corrosion, tritium diffusion and radiation creep are all strong functions of temperature and temperature gradients. Another consideration which was discovered early in the (LMFBR) breeder program is that separate liquid metal streams with even small temperature differences (< 15°C) when they mix at a junction can produce shocking thermal stress on structures in close proximity, with disastrous consequences. All this points to the need for a very comprehensive and accurate knowledge of the thermal hydraulics.

Thermal hydraulics and MHD are closely coupled in that the velocity profile and the boundary layer thickness determine the heat transfer from the blanket walls. Bulk heating both from neutrons and MHD eddy currents affect temperature profiles and influence heat transfer coefficients. Further, because blanket geometric configurations vary a great deal, it is important to determine for each blanket, which part lies in the thermal entry region and where it is in fully developed flow. Heat transfer coefficients vary substantially for these different flow conditions.

Non-neutron test stands such as the one being built at ANL can shed a great deal of light on the problem. Limited bulk heating effects can be simulated with electric heaters; however, establishing a temperature gradient that will persist for more than a minute complicated by the differences for entry regions and fully developed flow is difficult if at all possible.

As in the case of corrosion and MHD, fission reactors cannot simulate magnetic effects and would be of limited use. Point source test volumes are too small to be useful. The burden again falls on fusion test facilities.

In the case of thermal hydraulics, because of the short time constants needed to establish steady state conditions both TASKA and NET-P facilities can be used. Again in the case of TASKA, the true conditions of a tokamak DEMO with respect to magnetic field profiles and surface wall heating cannot be reproduced. However, it can go a long way in answering some fundamental questions of heat transfer in the presence of magnetic fields and bulk heating.

5.3.5 Tritium Production, Diffusion and Extraction

Tritium issues such as production, confinement and extraction have tremendous implications for fusion. Breeding of tritium depends on the breeder material, neutron multipliers, neutron spectrum, geometric configuration and the material in the regions surrounding the blanket. It is a highly complex and very interactive process which is virtually impossible to duplicate in any test facility. Verification of tritium self-sufficiency cannot be obtained even in a fusion test facility where a scaled down blanket test module with different boundary conditions would have to be used. It can, however, be used for calculational method verification. Such information can also be obtained from point neutron source facilities.

A great deal of information on T_2 diffusion through structural materials with various barriers can be obtained in non-nuclear test stands. Such information would be more relevant to tritium containment in elements not subjected to a nuclear environment, such as distribution pipes and steam generators. It would be marginally useful for diffusion through first wall and blanket components which are subjected to corrosion, neutron and ion sputtering, implantation and transmutation. Other synergistic effects of a fusion environment may also have a role. This type of information can only be obtained from a fusion test facility. Unlike MHD and thermal hydraulics, however, such a test requires long time constants to reach steady state equilibrium conditions. In that respect, a TASKA class facility would have to be the primary test bed with NET-P playing only a supporting role.

With regard to T_2 extraction techniques which reduce the tritium pressure and concentration in liquid alloys, much of the testing can be performed in non-nuclear test stands. Questions would remain, however, on the synergistic effects of a fusion environment on a continuous tritium extraction scheme where sputtering generated debris and corrosion products may have a role. Many fission reactors would have sufficient volume for such an experiment and may be of some help, albeit without a magnetic field and with a much softer spectrum. Such a test would also have to be of long duration to ensure steady state equilibrium. Here again, a TASKA class facility would be the primary choice.

5.4 Summary and Conclusions

The major blanket parameters required for a tokamak demonstration power reactor (DEMO) are given in Table 5.4-1. The design values for these parameters in the ANL DEMO are considered. The corresponding values in a "NET-P" class facility are also included. Since such a facility is not aimed at testing technology issues, no breeding blanket is included. The general nuclear environment and first wall parameters for TFCX, which is considered as a representative of this class of facilities, are shown. The range for the parameters obtained in a "TASKA" class facility is also given in Table 5.4-1. This represents the different tandem mirror technology test facilities such as TASKA, TDF, TASKA-M, and MFTF-α+T. A detailed list of the blanket parameters for these facilities is given in Appendix 5.A.

Proper simulation of the DEMO blanket and shield conditions in a test facility is essential for meaningful integrated blanket testing. The simultaneous duplication of the temperature, magnetic field and nuclear environment is necessary. The ability to provide a large enough testing volume that allows for duplicating the blanket geometry is also essential. The ability of the different test facilities to properly simulate the DEMO conditions is illustrated in the chart given in Fig. 5.4-1. The "NET-P" and "TASKA" class facilities are considered together with the simulation facilities. RTNS-II and FMIT are representative of the high neutron energy non-fusion test facilities. The fission reactors are represented by HFIR (thermal reactor) and FFTF (fast reactor). A full black square indicates that the DEMO condition can be fully simulated in the test facility. The full white square is indicative of the inability of the test facility to simulate the DEMO condition. A partially black square indicates that the test facility cannot fully duplicate the DEMO condition.

The blanket temperature environment can be properly simulated in the different test facilities with the exception of FFTF where the temperature cannot be reduced below ~ 300°C. While large magnetic fields, similar to those in a DEMO, can be obtained in fusion test facilities, very low (< 1 T) or no magnetic field is obtained in the non-fusion test facilities. Using energetic ion beams to produce the high energy neutron source does not permit using magnetic fields in RTNS-II and FMIT. A "TASKA" class facility does not fully

Table 5.4-1. Comparison of Relevant Blanket Parameters in the Demo, "NET-P" class and "TASKA" Class Facilities

	TOROIDAL "DEMO"	"TASKA" CLASS FACILITY	"NET-P" CLASS FACILITY
General			
Neutron wall loading	2 MW/m^2	0.25-1.5	0.7
Integrated wall load	20 $MW\text{-}y/m^2$	2-10	0.005
Average/peak surface heat flux	25-45/1000 W/cm^2	5/2400	7
Overall TBR	1.05	1.04 (TASKA)	NAP*
Energy multiplication	1.26	1.3-1.37	NAV**
FW surface area	400 m^2	11-33	~ 250
First Wall			
Materials	Steels	Steels	Steels
Temperature	< 400°C	100-440	NAV
Coolant	H_2O	H_2O	H_2O
Coolant op. temp.	260-300°C	30-60	~ 100
Number of cycles	< 100/y	< 100/y	~ 500/y
Peak damage rate	20 dpa/FPY 200 He appm/FPY	7-16 70-160	7 70

* Not applicable ** Not available

Table 5.4-1. (Continued)

	TOROIDAL "DEMO"	"TASKA" CLASS FACILITY
Solid Breeder Blanket		
Breeder	Li_2O	Li_2O
Neutron multiplier	Be	None
Coolant	H_2O	H_2O
Structure	Steel	Steel
Thickness	0.7 m	0.15
Local TBR	1.23-1.41 (with Be)	0.71
Peak power density	13 W/cm^3	6.4
Li_2O Temp. range	410-660°C	NAV
Inlet/outlet coolant temperature	260/330°C	150/200
Coolant pressure	100 bar	50
Liquid Metal Breeder Blanket		
Breeder/coolant	$Li_{17}Pb_{83}$	$Li/Li_{17}Pb_{83}$
Structure	Steel/V	Steel
Thickness	0.7 m	0.45-0.9
Local TBR	1.5-1.6	1.15-1.32
Peak power density	15 W/cm^3	3-7
Inlet/outlet coolant temperature	300/450°C	300/450
Coolant pressure	20 bar	3.5-5
Max. field	Inboard 6-8 T Outboard < 3 T	4-6

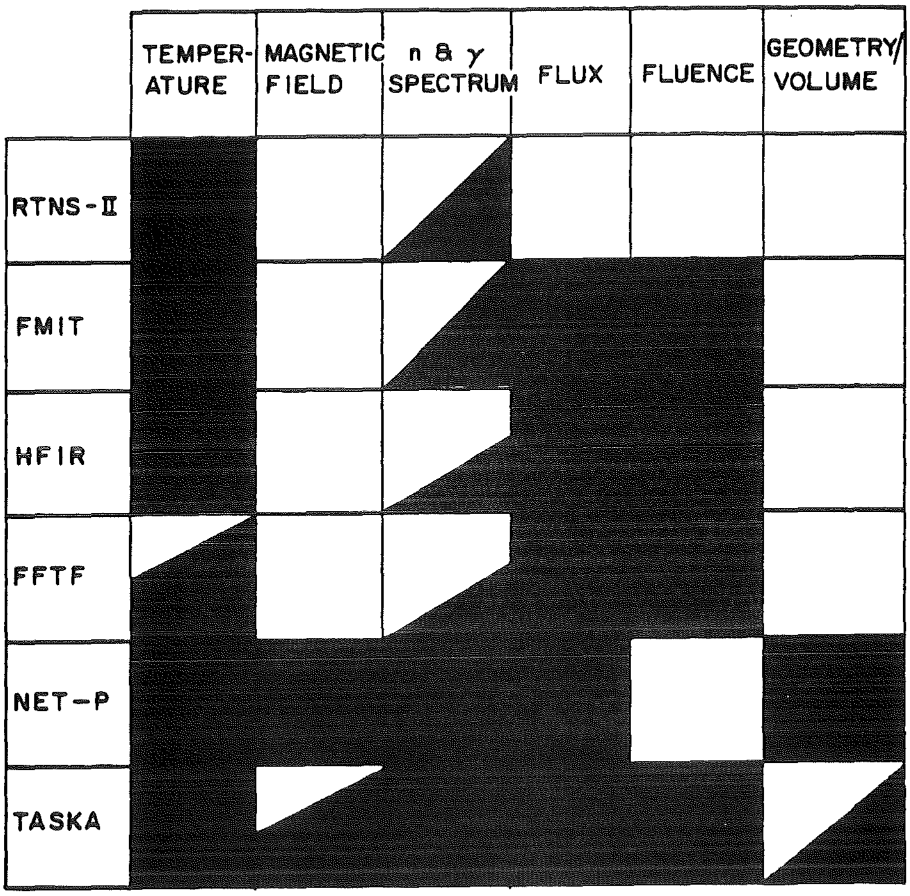


Fig. 5.4-1. Blanket test characteristics for different test facilities for DEMO blanket development.

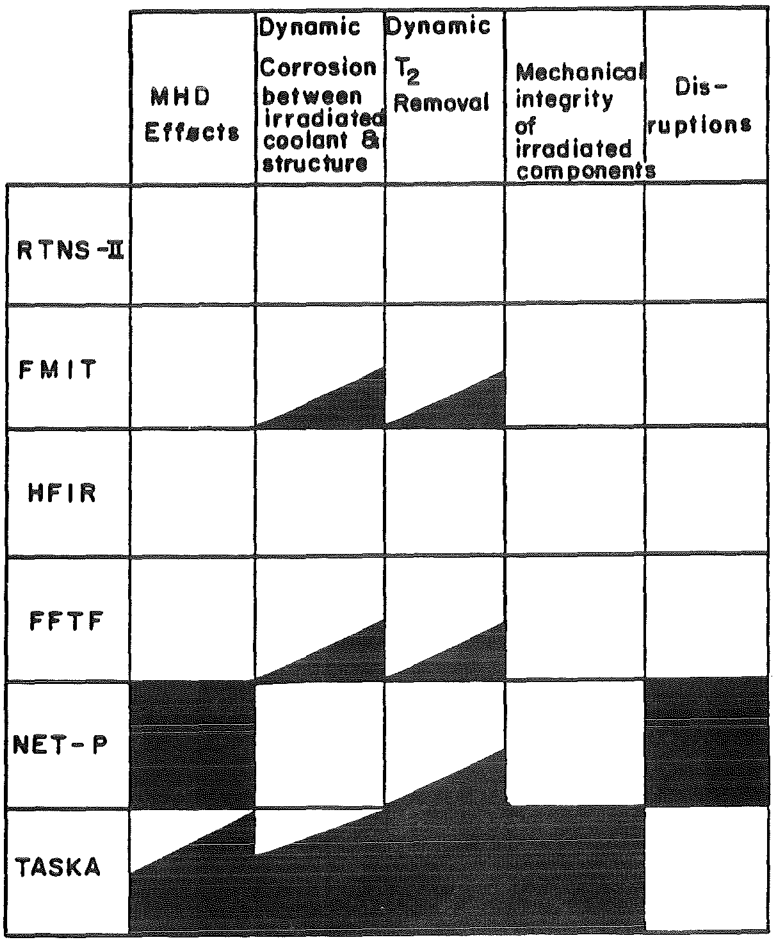


Fig. 5.4-2. Tokamak blanket issues tested in different test facilities.

simulate the magnetic field environment of a tokamak due to the different field profiles.

The blanket nuclear environment needs to be properly simulated in a test facility. This includes the neutron flux or neutron wall load, the fluence or integrated neutron wall load, as well as the neutron and gamma spectra. While the DEMO nuclear environment is fully simulated in a "TASKA" class facility, NET-P will not be capable of simulating the DEMO fluence. A physics test facility is a low availability machine. While TFCX has a neutron wall loading of 0.7 MW/m^2, its accumulated burn time of 2×10^5 s results in a very low fluence of ~ 0.005 $MW\text{-}y/m^2$ which is about four orders of magnitude lower than that needed in a DEMO. None of the non-fusion simulation facilities is capable of fully duplicating the DEMO nuclear environment. RTNS-II has very low flux and fluence besides having a neutron spectrum that is much harder than that in a fusion blanket. While FMIT has flux and fluence higher than in a DEMO as shown in Table 5.4-1, the much harder neutron spectrum does not render it capable of fully simulating the DEMO nuclear environment. While large neutron flux and fluence can be achieved in fission test reactors, the large gamma flux and the different neutron spectrum implies that fission reactors do not fully simulate the DEMO nuclear environment.

The testing volume in the simulation facilities is limited. The test volumes in RTNS-II, FMIT, HFIR and FFTF are 0.00016, 0.17, 1, and 2.5 ℓ, respectively. Tens of liters of test volume is needed for meaningful integrated blanket testing. While both "TASKA" and "NET-P" class test facilities can accommodate full blanket test modules, "TASKA" class facilities cannot fully simulate the geometry of the large radius D-shaped blanket modules of a tokamak DEMO.

The chart in Fig. 5.4-1 can be used to determine the blanket issues that can be tested in the different test facilities. The results are given in Fig. 5.4-2 for liquid metal blankets as an example. For a particular issue to be properly tested the test facility must be capable of properly duplicating the parameters that influence the blanket issue under consideration. For example, in order to test the MHD effects, the temperature, magnetic field, and neutron flux conditions of the DEMO need to be properly simulated. This implies that the non-fusion test facilities, where the magnetic field is not properly duplicated, cannot be used to test the MHD effects in a liquid metal blanket. The issue of dynamic corrosion between irradiated coolant and structure can be properly tested in a "TASKA" class facility with the only limitation of not adequately simulating the magnetic field profiles of a tokamak. This issue cannot be tested in a "NET-P" class facility or RTNS-II because of the limited neutron fluence. The corrosion issue can only be partially tested in the other simulation facilities due to the limited test volume, lack of magnetic field and the different neutron and gamma spectra. The issue of dynamic tritium removal under reactor relevant conditions can be fully tested only in a "TASKA" class facility. The limited fluence for NET-P and RTNS-II and the limited test volume in the non-fusion simulation facilities imply that other test facilities are only of limited use in testing this issue. Proper testing of the mechanical integrity of irradiated components requires an adequate testing volume and neutron fluence. This can be achieved only in a "TASKA" class facility. Another important issue that needs to be tested for a tokamak DEMO is the blanket response to plasma disruptions. Since plasma disruptions are postulated to occur only in a tokamak device, this issue can be tested only in a "NET-P" class facility. It is concluded that a "TASKA" class facility complemented by a "NET-P" class facility can adequately test the different tokamak DEMO blanket issues.

We conclude that many blanket and shield conditions of a tokamak DEMO can be properly simulated in a "TASKA" class facility. Meaningful integrated blanket testing can only be performed in such facilities where the combined nuclear, thermal, chemical and magnetic field environment is duplicated. Because of the small size of the blanket in a TASKA class facility proper scaling has to be used to interpret test results for a tokamak DEMO blanket. The issue of blanket response to plasma disruptions can be tested in a tokamak physics test facility of the "NET-P" class.

References for Chapter 5

1. M. Norget, M.T. Robinson, I.M. Torrens, Nucl. Eng. Des. 37 (1975),5o
2. D. Guggi et. al.
" Diffusion of tritium in single crystal Li_2O" Journ. Nucl. Mat. 18, 100 (1983)
3. M. Dalle Donne et. al.
Conceptual design of two helium cooled fusion blankets for INTOR, KfK 3584 (1983)
4. M. Abdou et. al.,"FINESSE, A Study of the Issues, Experiments and Facilities for Fusion Nuclear Technology Research and Development," Interim Report UCLA-EBG-84-3o, Oct. 1984.

APPENDIX 5.A. BLANKET TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR*
First Wall					
Independent or Integral		Int.	Ind.	Int., Ind.	Ind.
Radius	m	0.46	0.25	0.18-0.7[b]	1.4
Thickness	m	0.003[a]	> 0.01	0.02[b]	0.012[c], 0.014[d]
Non-Testing Surface Area	m^2	25.5	3	7.3	NAV
Testing Zone Surface Area	m^2	7.8	~ 8	3.6	NAV
Material / vol/o		HT-9/NAP[e]	304 SS/ND[e]	HT-9/50[b]	316 SS/50
Coolant / vol/o		$Li_{17}Pb_{83}$/NAP	H_2O/ND	H_2O/50[b]	D_2O[c], H_2O[d]/50
Peak/Av. Neutron Wall Loading	MW/m^2	1.52/1.52	1.4/1.4	1.34/0.63	NAV[e]/1.3
Peak Heat Flux	W/cm^2	5	2400	65	44
Temperature	°C	440	144	117[b]	350

---

* Reference: "International Tokamak Reactor," Phase 2A, Part 1 (1983).

a Thickness of first row of blanket tubes.

b For central cell shield insert.

c Outboard

d Inboard

e NAP - not applicable, ND - not determined, NAV - not available.

APPENDIX 5.A. BLANKET TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR
Total Power	MW	1.05	ND	0.23[a]	NAV
Max. Power Density	W/cm^3	10.4	25	4.3[a]	NAV
Max. dpa Rate	dpa/FPY	22.5	ND	2.5[a]	NAV
Max. He Production Rate	appm/FPY	162.8	ND	25[a]	NAV
Blanket[b]		I/II	II/III	II/III[c]	I/II/III[c]
First Wall Radius	m	0.463/0.4[d]	0.25	0.22-0.29/0.25	1.4
Average Wall Loading	MW/m^2	1.52	1.4	0.85/0.73	1.3
Width of Breeding Zone	m	8.7/1.01	1	0.83/0.43	60% coverage/3.5[e]
Thickness	m	1/0.96	~ 0.5	0.45, 0.64/0.15	0.5/0.5/NAV
Volume	m^3	51.7/3.6	~ 1.57/NAV	1.01, 1.77/0.13	114[f]/NAV/NAV

[a] For central cell shield insert.

[b] I - breeding blanket, II - liquid breeder test module, III - solid breeder test module.

[c] Two test modules.

[d] Due to different assumption.

[e] 1/12 of major circumference.

[f] 380 x 0.6 x 0.5

APPENDIX 5.A. BLANKET TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR
Breeder		$Li_{17}Pb_{83}$/Li	LiPb, Li/ceramics	$Li_{17}Pb_{83}$, Li/Li_2O	Li_2O[a]/$Li_{17}Pb_{83}$/ Li_2SiO_3, Li_2O
Vol/o		73	ND	73/20	59/NAV/NAV
Enrichment	% ^{6}Li	90/7.5	ND	90, 7.42/30	30/7.5/30, 7.5
Structure		HT-9/SS	ND	HT-9/316 SS	316 SS/SS/SS
Vol/o		7	ND	7/32	11/NAV/NAV
Multiplier		LiPb/---	ND	LiPb, ---/---	Pb/LiPb/Be
Coolant		$Li_{17}Pb_{83}$/Li	ND	LiPb, Li/H_2O	H_2O[b]/CO_2/He, H_2O
Vol/o		73	ND	73/28	5/NAV/NAV
Local TBR		1.04[c]/1.32	ND	1.15, 1.19/0.713	0.6[c]/1.2/1.28, 1.25[d]
Peak Power Density	W/cm^3	11.24/3	ND	6.95, 4.51/6.4	20[e]/NAV/NAV
Total Power	MW	42.42/3.2	ND	0.95, 1.02/0.38	26/13.5/20.5, 31.2

[a] Pellets

[b] and 25% He purge gas

[c] overall

[d] D_2O cooled F/W

[e] W/cm

APPENDIX 5.A. BLANKET TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR
Local Energy Multiplication		1.06[a]/1.02	ND	1.34, 1.37/1.31	NAV
Max. Structure Temp.	°C	420/NAV	ND	512, 500/230	150/650/NAV
Min. Structure Temp.	°C	300/NAV	ND	350, 345/NAV	NAV
Coolant Conditions:			ND		
Inlet Temperature	°C	300		300/150	50/220/390, 280
Outlet Temperature	°C	400/450		450/200	100/500/580, 320
Max. Velocity	cm/s	9.7/7		2.5, 1.8/NAV	450/NAV/NAV
Max. Pressure	bar	10.2		4.2, 3.5/NAV	40/40/50, 150
Max. Pressure Drop	bar	10.8/1[b]		5.7, 4.5/20	3/NAV/NAV
Total Coolant Flow Rate	kg/s	2917/6		38, 1.7/1.8	NAV/14/30, 116
Pumping Power	kWe	500/NAV		2.3, 1.4/NAV	NAV

[a] Overall

[b] Without MHD losses

6. NEUTRON DAMAGE IN MATERIALS

6.1 Introduction

One of the main driving forces behind the need for technology test facilities has always been the lack of high fluence, meaningful neutron irradiation data for fusion structural materials. As was discussed in Chapters 3 and 5, the current perception of the toroidal "DEMO" reactor includes steel first walls and blanket structure operating as high as 500^{0}C, under intense heat fluxes, while experiencing on the order of 100 to 150 dpa or more before being replaced. The hard 14 MeV component of the neutron spectrum will also produce several thousand appm of helium and a significant amount of solid transmutation products. In addition to the structural materials, coatings and heat sink materials will also be subjected to the unique fusion environment (see Table 6-1), and radiation damage in magnets must be considered.

Many studies in the past have attempted to address one or two of the key environmental parameters but there exists no facility at the present time to simultaneously produce the appropriate temperatures, neutron spectrum and adequate neutron fluence within 10 Calendar Years (CY) on a large volume ($\sim$ 100 liters) specimen. The capabilities of existing and proposed neutron facilities are listed in Appendix 6 A.

6.2 Test Volumes

An illustration of the test volume limitations of various neutron facilities is given in Fig. 6-1. Using the DEMO as the facility which we wish to simulate, we find that damage levels of at least 10 dpa/CY for ten years in total test volumes of roughly 100 liters would be necessary to extrapolate the effects to the 100 - 150 dpa level. The TASKA and TASKA-M (and even NET-EP) devices produce damage at a slightly lower rate but have more than adequate test volumes. On the other hand, the NET-P class of devices has the necessary volume but the accumulated damage rate is at least a factor of 100 too low. The use of a NET-EP device would give both test volume and damage levels of a magnitude similar to the TASKA-class.

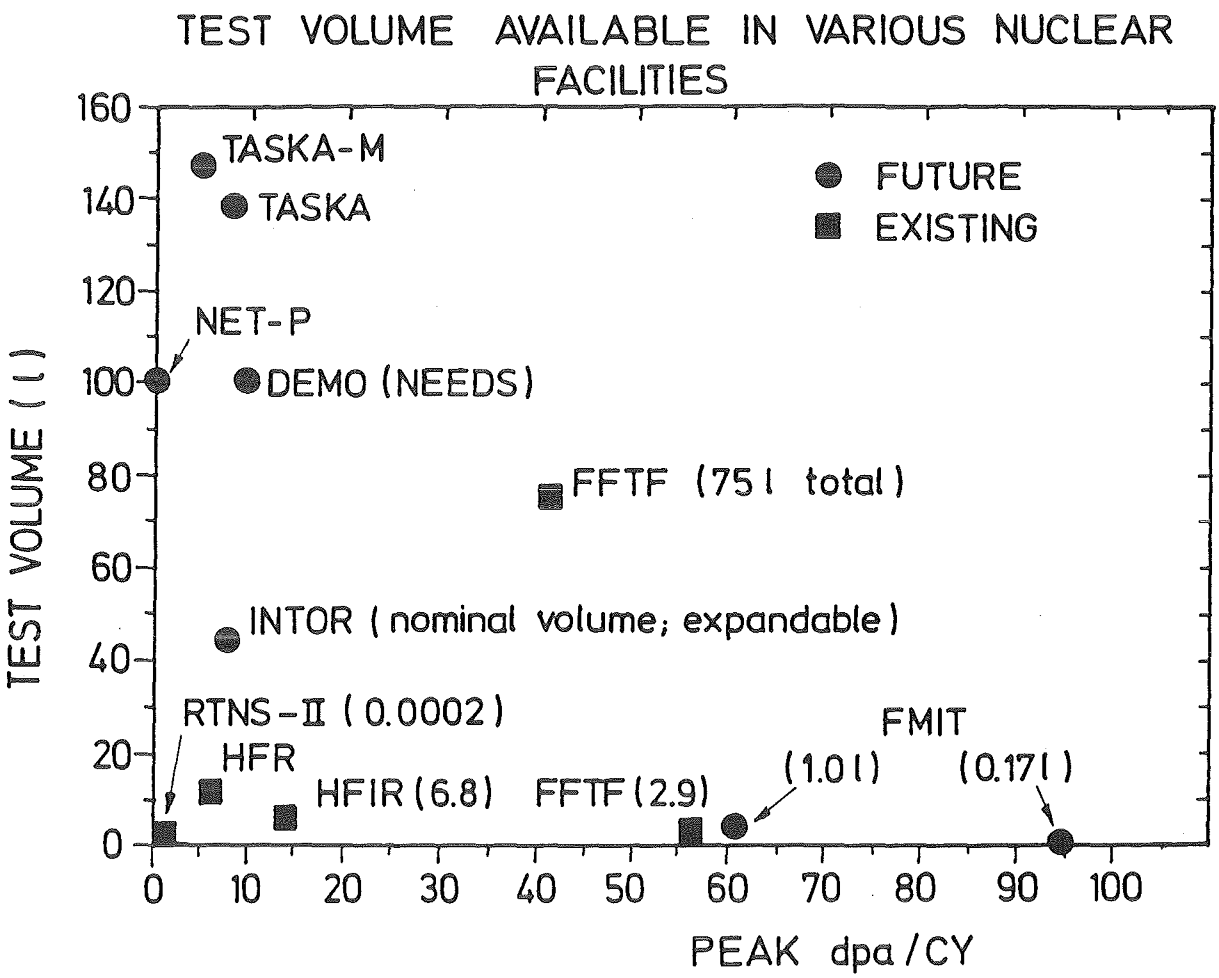
The useful test volume in a fisssion reactor is difficult to state because it usually is made up of several smaller test holes with a very elongated cylindrical geometry. Diameters of approximately 10 to 15 cm are typical

Table 6-1: Characteristic Nuclear Environment in Fusion Facilities

	Needs for Toroidal "DEMO"	"NET-P" Class Facility	"TASKA" Class Facility	Non Fusion Simulation Facilities
Coating (Be, W, TiC, etc.)				
Heat flux	500 W/cm^2	100-500	10-100	ASURF/ESURF
Wall Load	2-4 MW/m^2	0.7	1.5	Fission Reactors
Lifetime *	10 $MW\text{-}yr/m^2$	0.005	1-5	Fission Reactors
Heat Sink				
Wall Loading	2-4 MW/m^2	0.7	1.5	Fission Reactors
Lifetime *	5 $MW\text{-}yr/m^2$	0.005	1-5	Fission Reactors
Damage	70 dpa	0.07	14-70	Fission Reactors
Helium	500 appm	0.5	100-500	Fission Reactors
Operating Temperature	$300^{o}C$	$\sim 200^{o}C$	~ 100	Fission Reactors

*Components will probably be changed several times during reactor lifetime.

Fig. 6-1


TEST VOLUME AVAILABLE IN VARIOUS NUCLEAR FACILITIES
TASKA-M
TASKA
NET-P
DEMO (NEEDS)
FFTF (75 l total)
INTOR (nominal volume; expandable)
RTNS-II (0.0002)
HFR
HFIR (6.8)
FFTF (2.9)
FMIT
(1.0 l)
(0.17 l)
FUTURE
EXISTING
TEST VOLUME (l)
PEAK dpa / CY
0
20
40
60
80
100
120
140
160
0
10
20
30
40
50
60
70
80
90
100

of thermal reactors while fast reactor test elements have an even smaller diameter. The active test length also varies from 30 to 100 cm with the fast fission flux varying by sometimes more than a factor of 2 over this length. Similar test reactors are available in Europe but not repeated in Fig. 6-1. We see that while high displacement damage levels may be attained, the useful individual test volume is usually 10 liters or less. It should also be emphasized again that the neutron spectrum in the fission reactor will not give the correct transmutation rate but it will simulate the displacement damage reasonably well.

The pure 14 MeV neutron facility, RTNS-II, has the disadvantage of very low test volume ($\sim$ 0.0002 ℓ) and low dpa rates while thermal and fast neutron test facilities have high damage rates but low test volumes. The proposed FMIT facility has perhaps the highest test fluence but again a very low test volume capability.

The problem is illustrated in a slightly different way in Fig. 6-2 where we have plotted the product of damage times volume or dpa-ℓ/CY. This latter number also includes availability. From Fig. 6-2 we can see that NET-P and RTNS-II fall very short in their testing capabilities. The fission test facilities still fall short of the desirable values of several hundred dpa-ℓ/CY but the FFTF facility looks more attractive in spite of its lack of helium producing capability. Finally, the FMIT facility, despite its high damage level, has such a small test volume that it also falls far short of the combined damage-volume value deemed necessary.

6.3 Neutron Spectrum

One can get a perspective on the importance of neutron spectra from Fig. 6-3. The damage capability of the same neutron facilities as discussed above is plotted with respect to the appm He/dpa ratio in iron. The He/dpa ratio is basically a unique function of the neutron spectrum; i.e., there is a fixed ratio for a pure 14 MeV neutron flux (15 for Fe) and a unique number for every mixed spectrum. In the case of Fe, appm He/dpa ratios of 9 - 15 are typical of reactor facilities that contain Li or some tritium breeding compound.

It is clear from Fig. 6-3 that fast reactor facilities such as FFTF, and

Fig. 6-2

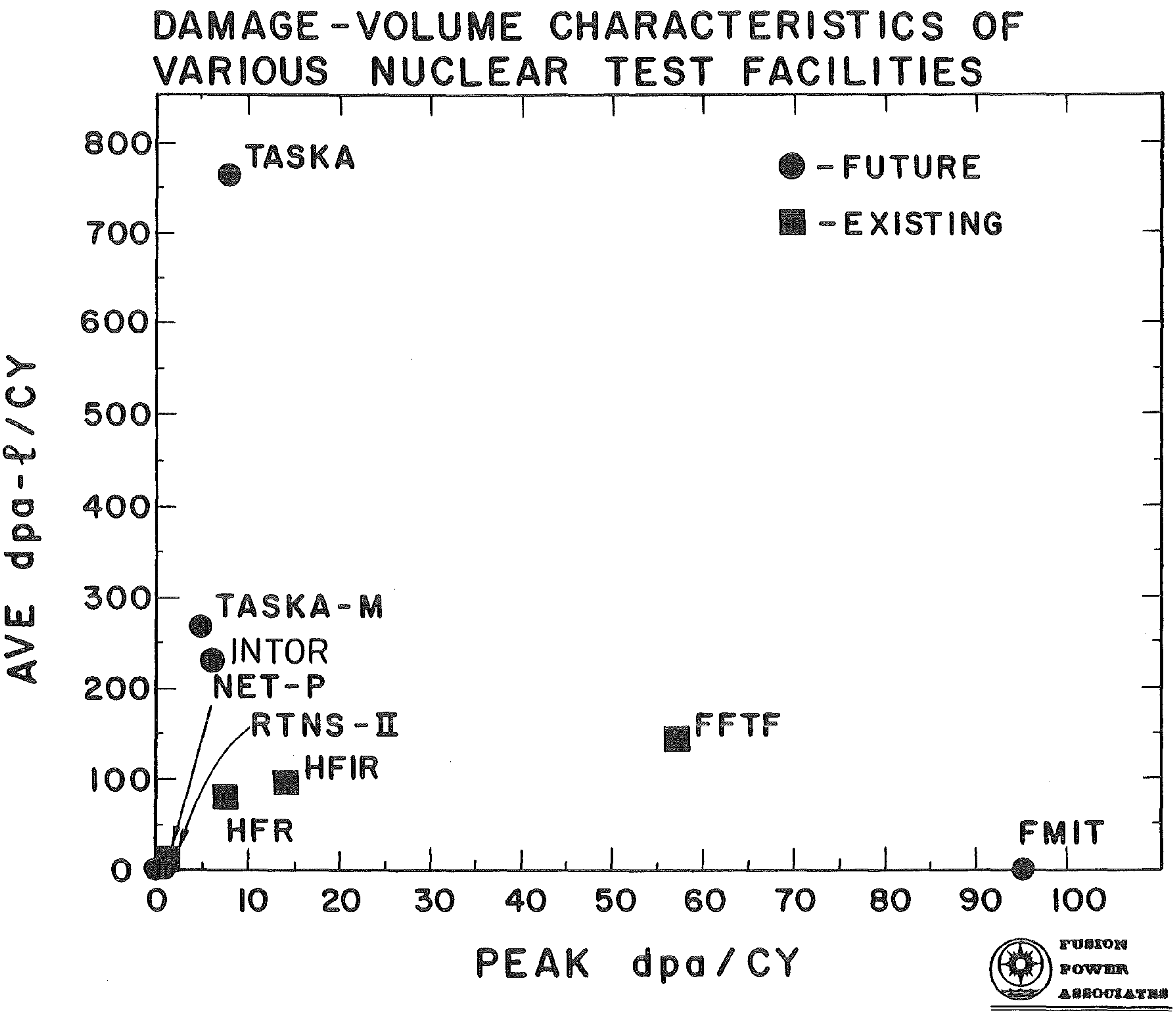
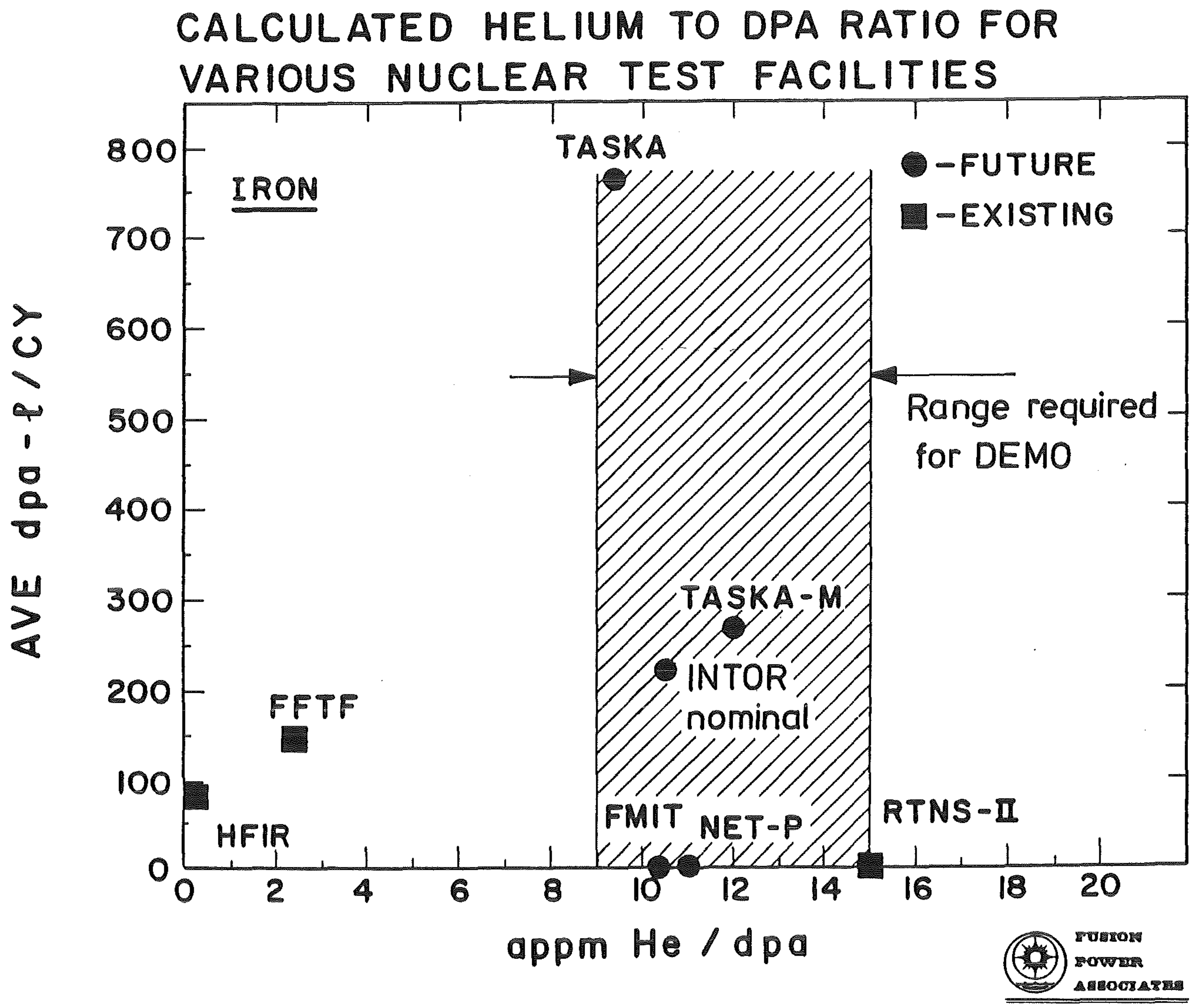

DAMAGE-VOLUME CHARACTERISTICS OF VARIOUS NUCLEAR TEST FACILITIES
AVE dpa-ℓ/CY
PEAK dpa/CY
0
100
200
300
400
500
600
700
800
0
10
20
30
40
50
60
70
80
90
100
TASKA
● -FUTURE
■ -EXISTING
TASKA-M
INTOR
NET-P
RTNS-II
HFIR
HFR
FFTF
FMIT
FUSION
POWER
ASSOCIATES

Fig. 6-3


CALCULATED HELIUM TO DPA RATIO FOR
VARIOUS NUCLEAR TEST FACILITIES
IRON
TASKA
●-FUTURE
■-EXISTING
Range required
for DEMO
TASKA-M
INTOR
nominal
FFTF
HFIR
FMIT
NET-P
RTNS-II
AVE dpa-ℓ/CY
appm He / dpa
0
100
200
300
400
500
600
700
800
0
2
4
6
8
10
12
14
16
18
20
FUSION
POWER
ASSOCIATES

thermal reactor facilities such as HFIR, have He/dpa ratios which fall far outside the values typical of fusion reactor facilities. Some damage processes such as void swelling, embrittlement, and creep are sensitive to the He/dpa ratio. It is therefore very important to correctly duplicate the spectral effects in any meaningful test.

6.4 Test Temperature

Another aspect of materials testing is the relationship beween the test volume available and the minimum temperature at which tests can be conducted (see Fig. 6-4). For various reasons, the minimum temperature test in a fast reactor is approximately 300^{o}C. Such facilities also tend to have quite small high flux individual test volumes (which usually are less than 10 cm in diameter and 30 cm in length). Obviously many test capsules could be used to irradiate a large number of small specimens, but it is usually impossible to place a very large (several liters in volume) test specimen of appropriate shape in the core of a fast reactor.

The FMIT facilities also suffer from small individual test volumes as do the thermal reactors such as HFIR and BSR. RNTS-II and BSR have liquid helium facilities but the HFIR is limited to water cooling and FMIT is limited to liquid Li cooling.

6.5 Magnet Irradiation

Fission reactors and facilities such as RTNS-II are able to provide some small specimen irradiation data for magnets. However, it is very difficult if not impossible to perform integral tests of superconducting magnets (i.e., including the insulator, stabilizer, and the superconductor in one unit) at relevant temperatures in FFTF, FMIT, HFIR, BSR, or even RTNS-II. Only large facilities such as the TASKA-class of devices can do such integral tests even though individual tests of some of the components can be done in fission reactors.

Finally, one can summarize the advantages and disadvantages of the neutron testing facilities for superconducting magnets in a qualitative way such as in Fig. 6-5. The five variables of importance are temperature, neutron and gamma spectra, test volume, neutron flux and fluence. The RTNS-II can adequately reproduce the temperature, damage rate and total fluence for

Fig. 6-4

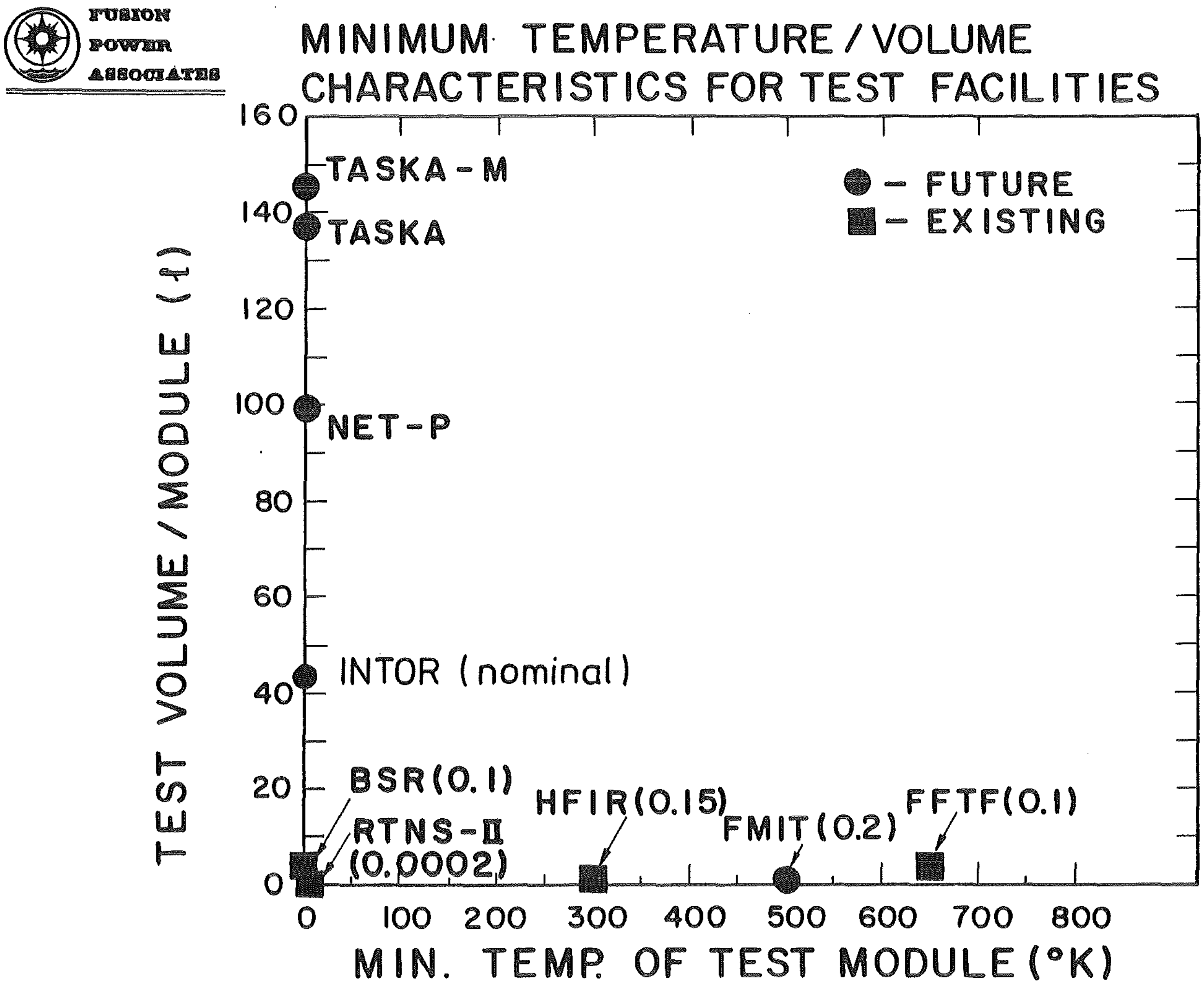
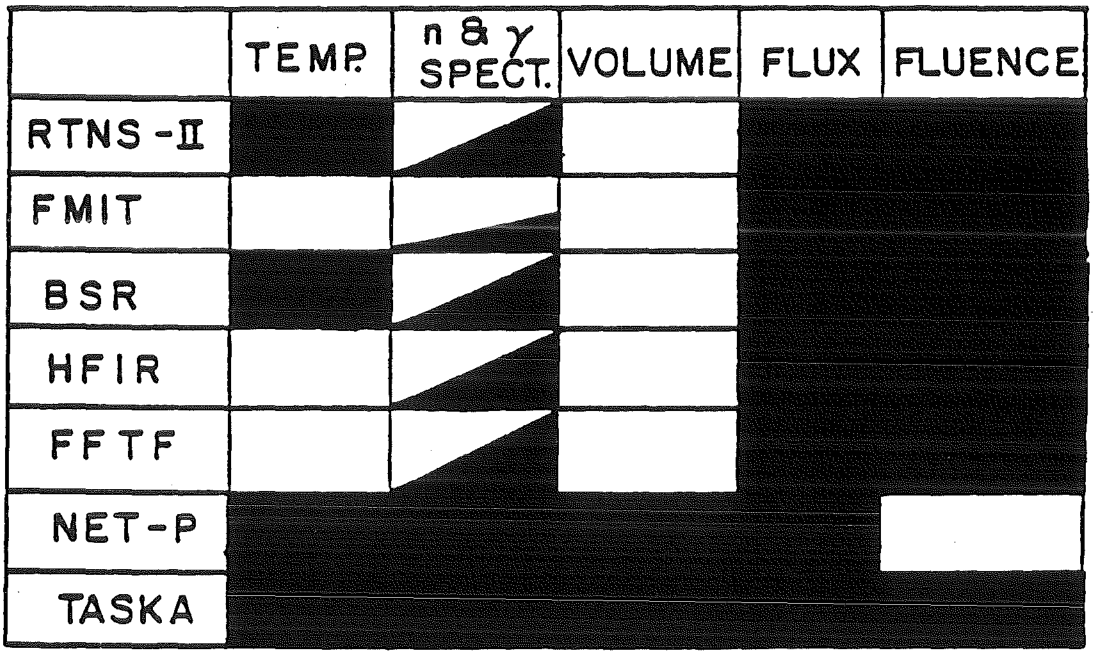

FUSION
POWER
ASSOCIATES
MINIMUM TEMPERATURE / VOLUME
CHARACTERISTICS FOR TEST FACILITIES
TASKA-M
TASKA
NET-P
INTOR (nominal)
BSR(0.1)
RTNS-II
(0.0002)
HFIR(0.15)
FMIT(0.2)
FFTF(0.1)
● - FUTURE
■ - EXISTING
160
140
120
100
80
60
40
20
0
0
100
200
300
400
500
600
700
800
TEST VOLUME / MODULE (ℓ)
MIN. TEMP. OF TEST MODULE (°K)

Fig. 6-5

ABILITY OF NUCLEAR FACILITIES TO TEST SUPERCONDUCTING MAGNETS FOR TOKAMAK DEMO



FUSION
POWER
ASSOCIATES

magnets and it can do a reasonable job on the spectral effects even though the spectra are harder than in a fusion device. However, the one big drawback of RTNS-II is the limited test volume. Individual specimens can be irradiated but composites are not readily accommodated in the facility.

The FMIT facility can do a good job with respect to flux and fluence but the higher energy neutrons (35 MeV) may not adequately simulate the damage to transmutation ratios found in a fusion reactor magnet. Again the test volume of FMIT is too small and cryogenic temperatures are not in the facility as presently envisioned.

The two thermal reactors, HFIR and BSR, can duplicate the flux and fluence parameters but have the drawback of not duplicating the exact neutron spectra. The test volume and geometrical configuration (long cylinders) in both facilities is such that only small specimens (usually not under stress) can be examined. Integral tests of a composite magnet are not possible in present fission reactors, even though valuable insight into some specific effects on the components has been gained. Fortunately the BSR can be used to irradiate at cryogenic temperatures but the HFIR cannot be used in that way.

The main drawbacks of FFTF, besides the different neutron spectra, are the high temperatures and lack of large test volumes. Fast reactors, to our knowledge, have never been used to successfully irradiate any components at cryogenic temperatures.

The NET-P device can satisfy all the requirements for in-situ testing of large superconducting magnet components except one. The total neutron fluence from the limited number of shots is far below that needed to test end of life behavior of DEMO magnets and less shielding is limited due to problems with respect to nuclear heating.

A TASKA level device does satisfy all the major irradiation criteria in a way which will be acceptable to design engineers. The temperature is certainly appropriate because the magnets are an integral part of the confinement scheme. The correct flux, fluence, and neutron spectra are there and since whole magnets will be subjected to the radiation, the tests in TASKA level devices are quite meaningful (see Fig. 6-5).

6.6 Conclusions

In summary, it should be quite apparent that auxiliary neutron facilities can be helpful, but are not adequate, in providing data for materials in future fusion DEMO reactors. The construction of TASKA level devices in parallel with physics facilities could tackle this problem as well as a NET-EP at a reasonable cost and probably even on a shorter time scale In both approaches the damage in the specimens, accumulated only after 10 to 20 years of operation, will require roughly factors of 2 extrapolation to the anticipated operation of the DEMO.

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR
Materials Test Modules					
Number of Modules		1	3	2	1
First Wall Radius	m	0.46	0.25	0.25-0.325/ 0.18-0.22[a]	1.4
Width	m	0.7/0.5[b]	2 x 0.5 + 1.2	0.55/0.53	1 x 0.9
Thickness	m	0.2	~ 0.5	0.125-0.2/0.2	0.15
Surface Area	m^2	2.02/1.45	2 x 0.78 + 1.88	1.01/0.69	0.9
Peak Neutron Wall Loading	MW/m^2	1.5	1.4	0.78/1.34	NAV
Average Neutron Wall Loading	MW/m^2	1.5	1.4	0.7/1.23	1.3
Operation Time	FPY	5.3[d]	~ 3.6[c]	7.8[f]	4.24[e]
Module Volume	ℓ	493/352	ND	186/227	135
Capsule Volume	ℓ	138/98	ND	75/71	45

[a] Module #3/Module #4

[b] He cooled/water cooled

[c] Based on 5 MW y/m^2 and 1.4 MW/m^2

[d] 15 calendar years

[e] 0.8 duty cycle and 15 calendar years

[f] 20 calendar years

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR
Max. # of Capsules		351/252	ND	276/180	153
Capsule Diameter	m	0.05	ND	0.05	0.05
Capsule Length	m	0.2	ND	0.125-0.2/0.2	0.15
# of Specimens in Test Matrix		29,548	ND	ND/22,576	29,548
# of Capsules Needed to Accommodate Test Matrix		313	ND	ND/116	300[a]
Volume of Largest Specimen	mm^3	11120[b]	ND	11120[b]	11120[b]
Volume of Smallest Specimen	mm^3	0.7[c]	ND	0.7[c]	0.7[c]
Depth of Highest Fluence Specimen Measured from Front of F.W.	mm	13	ND	13	32.6
Module Structure vol %	%	Ti-6Al-4V/ 316 SS 20	ND	HT-9 20	316 SS 24

[a] Only 130 capsules are used in the reactor at any time.

[b] Fracture toughness compact tension specimen.

[c] Swelling and microstructural specimen.

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR
Capsule Material vol %	 %	316 SS 20	ND	316 SS 25/20	316 SS 17
Coolant vol %	 %	He/H_2O 40/15	ND	He 30/40	H_2O 36
Thermal Contact Material vol %	 %	NaK 20	ND	NaK 25/20	NaK 17
Coolant Flow Rate	ℓ/min	7.3 x 10^4/ND	ND	3.4 x 10^4	0-760
Coolant Pressure	bar	34.5/6.9	ND	34.5	3.45
Inlet Coolant Temperature	°C	100/ND	ND	100	10-50
Outlet Coolant Temperature	°C	150/ND	ND	150	100-150
Specimen Temp. Range	°C	328-650/ND	ND	350-650	50-700
Peak Power Density	W/cm^3	5.8/11.6	ND	2.1/4	13.9[a]
Average Power Density	W/cm^3	3.2/7.3	ND	1.3	8.2[a]
Peak dpa rate	dpa/FPY	16/14	ND	7.5/10	8.5[a]

[a] Results of neutronics calculations performed at UW.

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	TASKA	TDF	TASKA-M	INTOR
Average dpa Rate	dpa/FPY	11/7.7	ND	3.4/3.9	5.07[a]
Peak He Production Rate	appm/FPY	150/140	ND	70/120	85[a]
Average He Production Rate	appm/FPY	73.5/66	ND	30/35	52[a]
Peak Accumulated dpa	dpa	85/74	ND	59/78	36[a]
Peak Accumulated He Production	appm	800/740	ND	546/936	360[a]
Cumulative dpa•ℓ	dpa•ℓ	8045/4000	ND	1960/2160	965[a]
Cumulative He appm•ℓ	appm•ℓ	53740/34300	ND	17140/19330	9922[a]

[a] Results of neutronics calculations performed at UW.

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	RTNS-II	FMIT
Materials Test Modules			
Number of Modules		1	2[a]
First Wall Radius	m	NAP	NAP
Width	m	0.01[b]	0.07 x 0.03/ 0.18 x 0.07[c]
Thickness	m	0.002	0.08/0.17
Surface Area	m^2	$7.8x10^{-5}$	0.002/0.013
Peak Neutron Wall Loading	MW/m^2	0.1[d]/0.3	12[d]
Average Neutron Wall Loading	$MW/^2$	0.08[d]	NAV
Operation Time	FPY	14[e]	13[f]
Module Volume	ℓ	0.00016[g]	0.17/2.25

[a] High flux test assemblies.

[b] Disc diameter.

[c] VTA-1/VTA-2

[d] Equivalent first wall neutron load.

[e] Based on 20 calendar years and 70% availability.

[f] 65% availability and 20 calendar years.

[g] Primary irradiation volume is 2 mm thick by 10 mm diameter.

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	RTNS-II	FMIT
Capsule Volume	ℓ	0.00016	NAP
Max. # of Capsules		1	3[a]
Capsule Diameter	m	0.01	NAP
Capsule Length	m	0.002	NAP
# of Specimens in Test Matrix		ND	14575[b]
# of Capsules Needed to Accommodate Test Matrix		ND	ND
Volume of Largest Specimen	mm^3	ND	2750[c]
Volume of Smallest Specimen	mm^3	ND	2.1[d]
Depth of Highest Fluence Specimen Measured From Front of F.W.	mm	3.5	20-30

[a] Three channels in each test assembly.

[b] High flux test matrix.

[c] Fracture toughness charpy specimen.

[d] Swelling and microstructural specimen.

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	RTNS-II	FMIT
Module Structure vol %	 %	ND	316 SS NAV
Capsule Material vol %	 %	ND	316 SS 50[a]/NAV
Coolant vol %	 %	ND	NaK/He or H_2O NAV
Thermal Contact Material vol %	 % 	ND	NaK/NaK, He or H_2O 50[a]/NAV
Coolant Flow Rate	ℓ/min	ND	NAV
Coolant Pressure	bar	ND	< 3.45/NAV
Inlet Coolant Temperature	°C	ND	NAV
Outlet Coolant Temperature	°C	ND	NAV
Specimen Temp. Range	°C	ND	60-650/ 100-600
Peak Power Density	W/cm^3	NAV	70/44[b]

[a] VTA-1 assumed to have 50 vol % SS and 50 vol % NaK.

[b] Power per unit volume of SS.

APPENDIX 6.A. MATERIALS TESTING PARAMETERS FOR THE DIFFERENT TEST FACILITIES

PARAMETER	UNIT	RTNS-II	FMIT
Average Power Density	W/cm^3	NAV	NAV/NAV
Peak dpa Rate	dpa/FPY	0.4/1.2[c]	145/5
Average dpa Rate	dpa/FPY	0.3/0.9	37.5/NAV
Peak He Production Rate	appm/FPY	6/18	1480/62
Average He Production Rate	appm/FPY	4.5/13.5	383/NAV
Peak Accumulated dpa	dpa	5.6/16.8	1885/65
Peak Accumulated He Production	appm	84/252	19240/806
Cumulative dpa.•ℓ	dpa.•ℓ	0.0007/0.002	83/NAV
Cumulative He appm.•ℓ	appm.•ℓ	0.01/0.03	850/NAV

[c] Based on achieved peak flux of 4×10^{12} n/cm^2s/Based on design peak flux of 1.2×10^{13} n/cm^2s.

7. TRITIUM CYCLE

7.1 Introduction

The tritium technology needs for the tokamak DEMO have been reviewed. Based upon the present conceptual designs, the tritium systems of all magnetically confined fusion reactor experiments are independent of the method of plasma confinement. The tritium systems generally recognized are divided into three functional groups, namely: (a) the Fuel Circuit, which includes fuel delivery to the plasma, plasma-wall interactions, unburned fuel and ash removal, fuel recycle (purification and isotopic separation), and fuel storage; (b) Tritium Containment, which includes the primary, secondary (gloveboxes) and tertiary (building) containment facilities, air detritiation systems, monitoring and control instrumentation, and barriers to prevent tritium contamination of the power cycle; and (c) Tritium Breeding, which includes breeder materials, tritium release, removal, and purification.

The planning for a fusion power DEMO will require that each component of the tritium cycle be examined in more detail than has been done in the past. Such an examination is especially needed because a DEMO will have to undergo a licensing process and the inventory and confinement of tritium at every point in the cycle will be carefully scrutinized. Several radiological studies[1,2] regarding the environmental impact of proposed fusion power plants have calculated that tolerable doses to the population occur when the tritium airborne releases are in the range of 10-100 Ci/d. Release of tritium to surface waters has the effect of decreasing the maximum dose to an individual by a factor of 100 and only slightly increasing the local-area dose.[2] Current conceptual reactor designers have accepted these tritium effluent limits and proposed conceptual containment schemes to meet the goal. In light of the large amount of tritium handled in the fuel cycle and stored on the site, realistic release rates will need to be carefully evaluated for the DEMO.

This study represents an attempt to quantify the tritium cycle needs for the DEMO and compare them with the information which will be obtained from pre-DEMO experimental facilities. From such a comparison, some preliminary conclusions are made regarding the adequacy of the planned experiments to support the needs of the DEMO. Information for the pre-DEMO facilities was obtained from the conceptual designs of machines in the TASKA-class and NET-P class. Information is included regarding the tritium burn experiments planned for the current plasma test facilities, the Tokamak Fusion Test Reactor (TFTR) and the Joint European Torus (JET), as well as the capabilities of a non-plasma facility, the Tritium System Test Assembly (TSTA),[6] which is dedicated to the development of the fusion fuel cycle. At present, plans for the construction of tritium test facilities similar to TSTA are being discussed in Japan and Europe. Data and remarks referring to TSTA in the following chapters should therefore be understood as typical for this class of facilities. It is also recognized that numerous experimental studies on components of the tritium fuel cycle are in progress and have been summarized in various reports[3,4] and proceedings;[5] however, these studies were not included in this limited survey because they have not been integrated as part of a fusion demonstration system. Data on the DEMO and the pre-DEMO experimental facilities are recorded in Table 7.1-1 and discussed in the following sections.

7.2 Fuel Cycle Systems

7.2.1 Fuel Injection

The fueling rate of the TASKA class facility device nearly approaches that of the DEMO (see Table 7.1-1 and Fig. 7.2-1), because the fractional burnup in the DEMO is 10% while in the mirror facility it is only 1%. The total handling rate in TASKA is, therefore, about 80% of that in the DEMO. It should be noted that there is a difference in fueling technique, pellets in DEMO and NET-P, but neutral beams in the mirror. The fueling of JET and TFTR will probably be by gas injection which does not simulate the pellet fueling for the DEMO. The fueling rate for NET-P during a short operational period will be only 30% of the DEMO rate. TSTA can simulate gaseous fuel handling but because no plasma exists in the system, experiments to achieve reactor relevant fueling procedures cannot be accomplished.

7.2.2 Plasma Wall Interactions

With tritium in the plasma, several chemical and physical plasma wall interactions are predicted. The extent of these interactions represents one of the most critical unresolved issues for tritium fuel management; hence, experimental information is sorely needed. Some of the potential interactions can be inferred from current experiments. Chemical reactions were first considered. These reactions are caused by energetic tritium atoms impinging on the carbon, nitrogen and oxygen impurities on the first wall surface to form tritiated methane, ammonia and water. The chemical species and quantities of such impurities have a direct impact upon the exhaust fuel cleanup requirements.[6]

Table 7.1-1. Comparison of Tritium Cycle Needs for DEMO

	Toroidal "DEMO"	Pre-DEMO Facilities		Plasma Test Facilities		Non-Plasma Facilities
Integral Tests	Needs	NET-P Class	TASKA Class	JET	TFTR	TSTA-Type
A. Fuel Cycle						
• Fueling (mg/s)	Pellets: 23 T, 16 D	Pellets: 7 T, 5 D (intermittent)	NBI: 4-20 T, 3-16 D Pellets: 17 D	NBI: ~ 3 D Gas-Puffing: ~ 13 T ~ 10 D	Gas Injection: 40 T mg/Test	Gas Injection: 13 T, 8 D
• Plasma						
T-Burn/Pass (%)	10	~ 5	0.3 - 1	< 1	< 1	No Plasma
T-Wall-Interactions	Chemical, Sputtering and Implantation	Low Fluence for Data Required	Low Ion Flux to Walls (except certain inserts)	Very Low Fluence	Very Low Fluence	No Plasma or Neutrons
• Exhaust from Plasma-Chamber						
Flow Rate (mg/s)	21 T, 14 D, 3 He	7 T, 5 D, 0.5 He (intermittent)	4-17 T, 6-24 D 0.02-0.2 He	0.1-6 T, 0.06-4 D 0.2 He + NBI-Exhaust		13 T, 8 D, 0.3-3 He
Impurities Flow Rate (mg/s)	0.09 (H), 0.6 (C), 0.05 (N), 0.03 (O) 1 (I_2)	Very Small	0.04-0.4 (H), 0.1 (C), 0.1 (N), 0.7 (O)	0.01 (H), 1% of D,T		0.08 (H), 0.02-0.1 (C), 0.1 (N), 0.7 (O), 0.2 (Ar)
• Vacuum Pumps	Cryogenic	Compound Cryogenic	Compound Cryogenic	Cryogenic Turbomolecular	Cryogenic Turbomolecular	Cryogenic Adsorptive
• Fuel Cleanup						
Flow Rate (mg/s)	21 T, 14 D, 3 He + Impurities	Accumulate for Batch Process	4-80 T, 18-90 D 0.02-0.2 He + Impurities	0.02 T, 0.07 D + Impurities Batch Process	No Recycle	13 T, 8 D, 0.3-3 He + Impurities
Processes	He-Separation Gettering Cryosorption	He-Separation Gettering Cryosorption	He-Separation Gettering Cryosorption	He-Separation Cryosorption Diffusion	Gettering	He-Separation Gettering Cryosorption
• Isotope Separation	Cryogenic Distillation	Cryogenic Distillation	Cryogenic Distillation	Chromatographic	No Isotope Separation	Cryogenic Distillation
• Fuel Storage	Metal Tritide	Depleted Uranium	Depleted Uranium	Gaseous Storage	Depleted Uranium	Depleted Uranium

Table 7.1-1. Comparison of Tritium Cycle Needs for DEMO
(Continued)

Integral Tests	Toroidal "DEMO" Needs	Pre-DEMO Facilities: NET-P Class	Pre-DEMO Facilities: TASKA Class	Plasma Test Facilities: JET	Plasma Test Facilities: TFTR	Non-Plasma Facilities: TSTA-Type
B. T-Containment						
• T-Inventory, (g)						
Fuel Cycle	443	62	459	2 x (1-3)		115
Breeder	40 (LiPb) - 1100 (Li_2O)		33 (LiPb) - 80 (Li_2O)			
Storage	2000	155	1500	max. 8	max. 5	max. 150
• T-Retention System						
Glovebox (vol)m^3	900	250	900		30	~ 100
Cleanup	Cataly. Recomb. Mol. Sieve Dryer	Cataly. Recomb. Mol. Sieve Dryer		Cataly. Recomb. Mol. Sieve Dryer	Cataly. Recomb. Mol. Sieve Dryer	Cataly. Recomb. Mol. Sieve Dryer
Bldg. (vol)m^3	3 X 10^5	7.6 X 10^4	7 X 10^4		2.6 x 10^4	3 x 10^3
Cleanup	Cataly. Recomb. Mol. Sieve Dryer	Cataly. Recomb. Mol. Sieve Dryer		Cataly. Recomb. Mol. Sieve Dryer	Cataly. Recomb. Mol. Sieve Dryer	Cataly. Recomb. Mol. Sieve Dryer
• Waste Treatment						
Liquid						
Coolant (H_2O) Contamination [Ci/d]	1600	Low Permeation	23 - 3300	Very Small	Very Small	Very Small
Water Flow to Detritiation Unit (ℓ/d)	1600	Not Needed	~ 1600	Not Needed Controlled Disposal	Not Needed Controlled Disposal	Not Needed Controlled Disposal
Gas						
Flow Rate (m^3/min)	15	1.7	15	Very Small	Very Small	1.7
Solid						
Volume Generated	Undefined (T + γ)	Undefined	Undefined (T + γ)	Used Cells: Chromatographic, Diffusion	Used Beds: Getter, Adsorber	Used Beds: Getter, Adsorber
Disposal Method	Undefined	Undefined	Undefined	Undefined	Package and Recover Off-Site	Will Determine
• T-Monitoring	Needed - Not Defined	Needed - Not Defined	Needed - Not Defined	Needed - Not Defined	In Place	In Place/ No γ or Neutrons

Table 7.1-1. Comparison of Tritium Cycle Needs for DEMO (Continued)

	Toroidal "DEMO"	Pre-DEMO Facilities		Plasma Test Facilities		Non-Plasma Facilities
Integral Tests	Needs	NET-P Class	TASKA Class	JET	TFTR	TSTA-Type
C. Breeder Blanket						
• Liquid Breeder/ T-Extraction	$Li_{17}Pb_{83}$/ Vacuum, Getter or Molten Salt	Small Test Module	Li/ Cold Trap $Li_{17}Pb_{83}$/Vacuum	No Breeder	Small Test Module	No Breeder
• Solid Breeder/ Coolant	Li_2O/H_2O	Small Test Module	Li_2O/H_2O	No Breeder	Small Test Module	
T-Extraction	In-Situ Continuous Sweep Gas		Batch Process			No Breeder
• T-Separation from Carrier Gas	Oxidize, Adsorb Desorb, Electrolyze	Static Test	Test Systems		Static Test	No Breeder
• T-Reduction in Steam Generator	Liquid Metal - Double-Walled HX, or Intermediate Na Loop. H_2O - 1% to Tritium Water Recovery Unit.	No Power Cycle	Liquid Metal - Double-Walled HX, or Intermediate Loop of Na, He, or Organic Terphenyl. H_2O - Side Stream to Tritium Water Recovery Unit	No Power Cycle	No Power Cycle	No Power Cycle

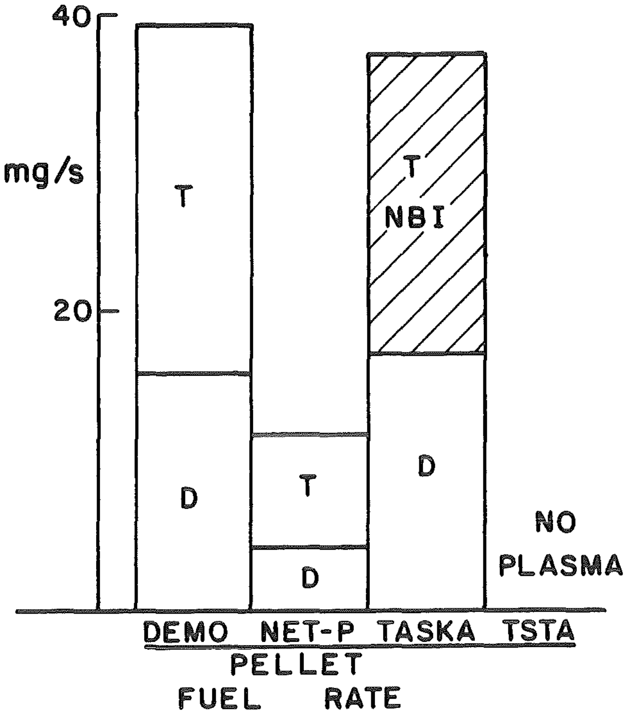


Fig. 7.2-1

TRITIUM FUEL CYCLE

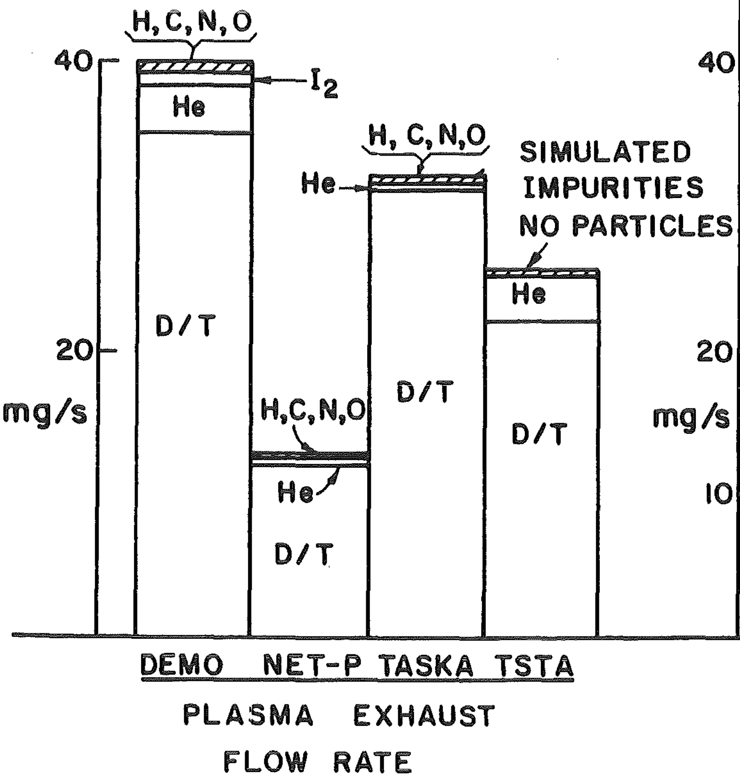


Fig. 7,2-2

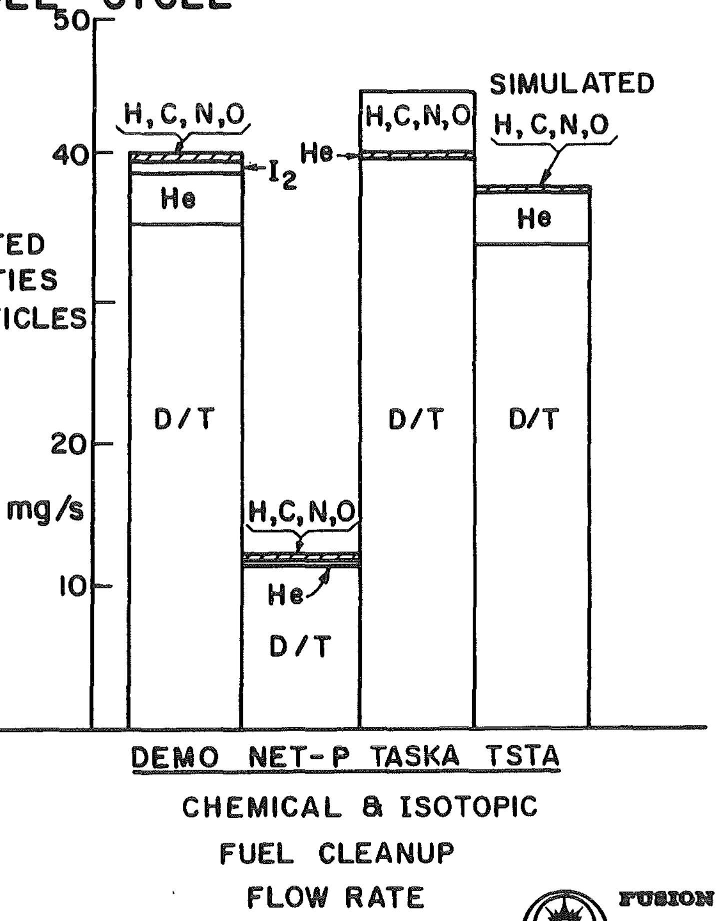


Fig. 7.2-3

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In the case of a tokamak DEMO the high fluxes of D/T ions and neutral atoms impinging against the first wall and limiter have serious implications for tritium containment. Initially, the impinging particles cause sputtering of the wall and limiter. The physical form of these redeposited sputtered metallic particles is of concern. If they adhere to the wall, these surfaces may become rough and porous. Such surfaces will adsorb molecular DT and other chemical species, which will be only slowly desorbed during subsequent evacuation of the torus. Conversely, the metallic particles may not adhere to the wall but form micro-dust particles which adsorb gaseous tritium species. Such particles may become airborne during the evacuation of the chamber, and special techniques may be required to remove such particles from the exhaust stream.

The third area of plasma wall interactions involves energetic tritium atoms penetrating into the first wall.[(7)] Most of these atoms migrate to the plasma side of the wall, form molecular tritium and are recycled. Some of the tritium atoms diffuse, however, to the coolant side of the first wall and contaminate the coolant. During the diffusion process the first wall is also subjected to a neutron flux from the plasma which may displace atoms in the metal. These displaced atoms may form vacancies which trap the diffusing tritium atoms, thereby retarding the tritium diffusion and thus increasing the tritium inventory in the first wall and limiter.

In regard to which pre-DEMO devices may yield information, only those devices which contain tritiated plasmas can be considered. The present day experiments, JET and TFTR, will not have sufficient operational time in order to study these effects. The mirror device, TASKA, should yield information on the new chemical species formed in the reactor because the flow rate of tritium is nearly 80% of that required for the DEMO. Normally, a mirror machine does not have sufficient particle flux to the first wall to make it a useful tool to study plasma wall interactions. In the experimental mirror device, TASKA-M, however, the impingement of the neutral D/T beam onto the plasma causes a high flux of charge-exchange atoms impinging upon the surrounding wall insert. Studies of plasma-wall interactions in this region and examination of the neutral beam dumps and end dumps after use may yield valuable information, as discussed in Chapter 4.

7.2.3 Plasma Exhaust

The plasma exhaust rate for a TASKA class facility, Fig. 7.2-2, approaches that of the DEMO for reasons described previously. TSTA can provide some experience in this technological area but again the lack of a plasma makes it impossible to satisfactorily simulate the chemical composition of the exhaust gases.

7.2.4 Fuel Cleanup

TSTA provides a substantial amount of information on the chemical and isotopic fuel cleanup, Fig. 7.2-3, at a rate comparable to the DEMO. The TASKA class facility provides a fuel cleanup rate comparable to DEMO but does not provide much experience with the separation of helium because of its low fusion power. The D/T recycle in TASKA is particularly high because in addition to the low fractional burnup in the reactor, the neutral beam fuel recycle into the Fuel Cleanup Unit is also high.

7.3 Tritium Containment

7.3.1 Amount and Location

The potential for tritium release to the environment is determined by the quantity, location and containment design. For comparison purposes, the inventories in the tritium subsystems of each facility have been determined, Fig. 7.3-1. Such a classification does not assess the containment design and the potential for tritium release from each subsystem. Note that the reactor site tritium inventory for the DEMO is estimated to be twice that of the TASKA mirror facility. This is due to the fact that the comparison is between a 1000 MW device and a 86 MW device. Note also that the inventory for the solid breeder is much greater than for the liquid alloy $Li_{17}Pb_{83}$. At steady-state, tritium is extracted from the breeder at the same rate that it is consumed in the fusion process; therefore, because of the higher retention of tritium in the solid breeder, the solid breeder has a higher tritium inventory than a liquid breeder.

7.3.2 Detritiation Systems

Most of the gaseous tritium decontamination required in these facilities will be achieved by the use of air or glovebox detritiation units usually involving oxidation followed by the adsorption of tritiated water on a desiccant. The comparison of tertiary (building) volumes and secondary (glovebox) volumes, plotted in Fig. 7.3-2 on a logarithmic scale, is an appropriate measure of the system requirements. Estimates of several facilities were made

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TRITIUM RADIOLOGICAL ISSUES

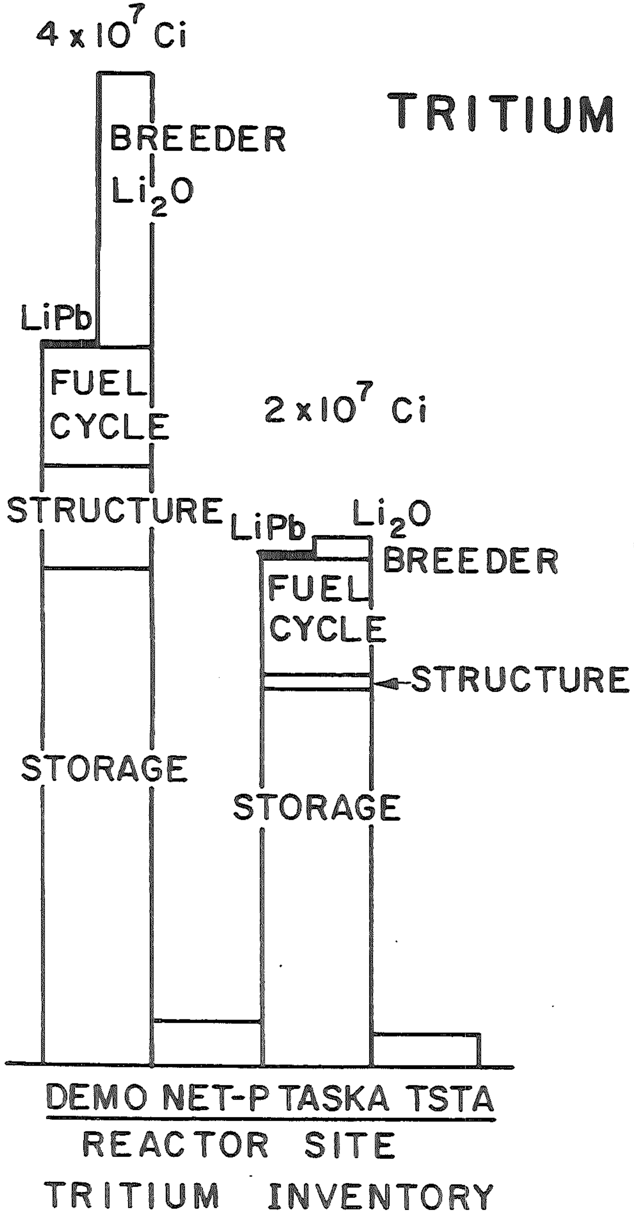


Fig. 7.3-1 .

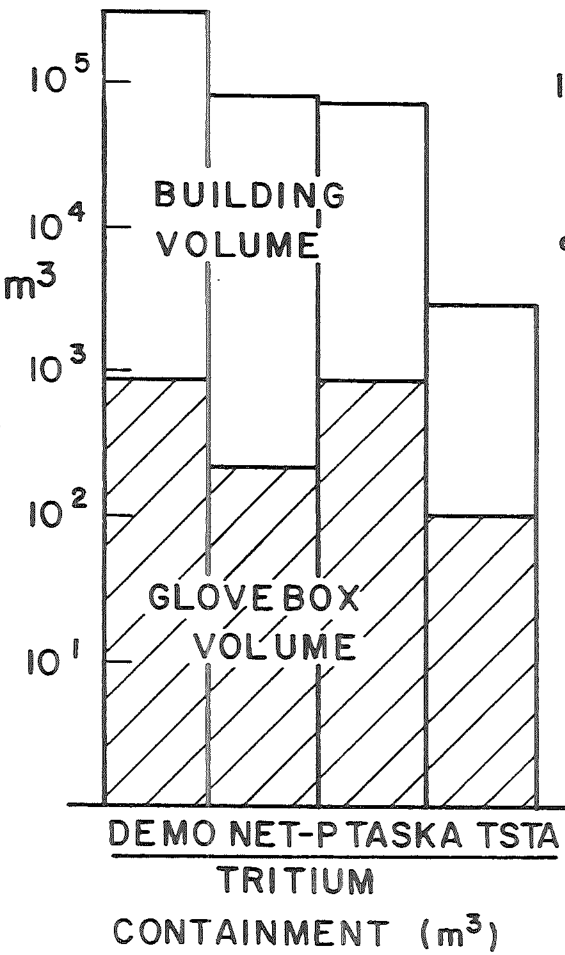


Fig. 7.3-2

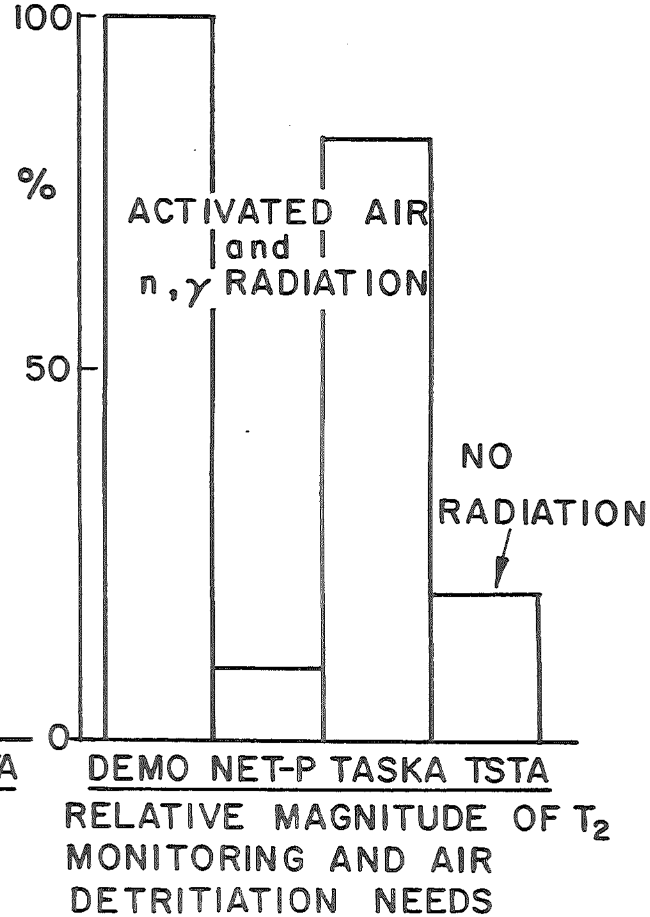


Fig. 7.3-3

in order to make this comparison. The DEMO reactor containment building was given in the DEMO report but the size of the secondary containment structures required an estimate. It was assumed to be similar to the 900 m^3 for the INTOR design.[8] The preconceptual design presented data for the Ignition Experiment,[9] TFCX. The TFTR volumes were obtained from the site design[10] and tritium delivery system module.[11] No facilities were designed for the TASKA mirror facility; consequently, the systems were scaled down from the Mirror Advanced Reactor Study.[12] Based upon the capacity of the Tritium Waste Treatment system[6] for TSTA, it was estimated that up to 100 m^3 of secondary enclosures could be installed.

All pre-DEMO facilities will provide considerable experience because standardized units for air and glovebox detritiation will probably be linked together to achieve the capacity needed. Air detritiation systems are required both on a routine basis and on an emergency basis. Detritiation systems are expensive because of the large volumes of air to be handled, and the time delay acceptable before human entry is regained to the facility following an accidental tritium release.

Water detritiation units are planned for installation in most of the pre-DEMO facilities. Experience gained with these water detritiation units in the pre-DEMO and in non-fusion facilities will provide the data necessary for the DEMO.

7.3.3 Tritium Barriers

Tritium losses to the environment via the power conversion system are difficult to control especially when a steam-driven turbine generator is utilized. Prevention of tritium permeation (and leakage) from the reactor coolant through the steam generator requires either maintenance of the tritium concentration in the reactor coolants to exceedingly low levels, utilization of tritium barriers in the steam generator (SG) which reduce permeation by a factor of 10^5, or the introduction of an intermediate heat transfer loop. Tritium barriers for use in the SG have been studied in non-reactor experiments. These include oxide coatings on the water-side of the steam generator,[13] laminated duplex tubing[14] and double-walled SG tubing with tritium gettering between the walls.[15] The loop designed for TASKA for the continuous testing of tritium barriers has the advantage of examining the tritium barrier concept in concert with synergistic effects, such as corrosion.

Coolants such as water or helium have been suggested for DEMO designs with non-mobile breeders. The tritium concentration in water could be controlled by the use of a water detritiation unit as mentioned previously. Conceptual design studies in helium gas coolants have assumed that the tritium partial pressure can be reduced by the introduction of oxygen to form tritiated water. If the reaction occurs at a significant rate, the tritium partial pressure would only be about $\sim 10^{-14}$ torr. No experiments have been performed to date which have tested the kinetics of the formation of tritiated water under conditions which simulate the regime of a fusion reactor coolant system. A non-reactor experiment suggests that the formation of tritium oxide occurs in the metallic oxide coating on the SG tubes and does not depend upon the slower gas phase oxidation reaction.(16) A test loop has been designed for TASKA type devices which would serve to test these systems.

7.3.4 Tritium Monitoring

The estimate of the relative magnitude of tritium monitoring requirements for each pre-DEMO device compared to the DEMO, Fig. 7.3-3, has been related to the total inventory in the fuel cycle. Such monitoring, both within the facility and the surrounding environs, must be accomplished in the presence of neutron and gamma radiation from the reactor and interference from activated air radionuclides. The TASKA facility monitoring requirement more nearly matches that estimated for the DEMO because of the large amount of tritium in the recycle from the neutral beam injectors. TSTA suffers in this comparison because interfering radiation is not present.

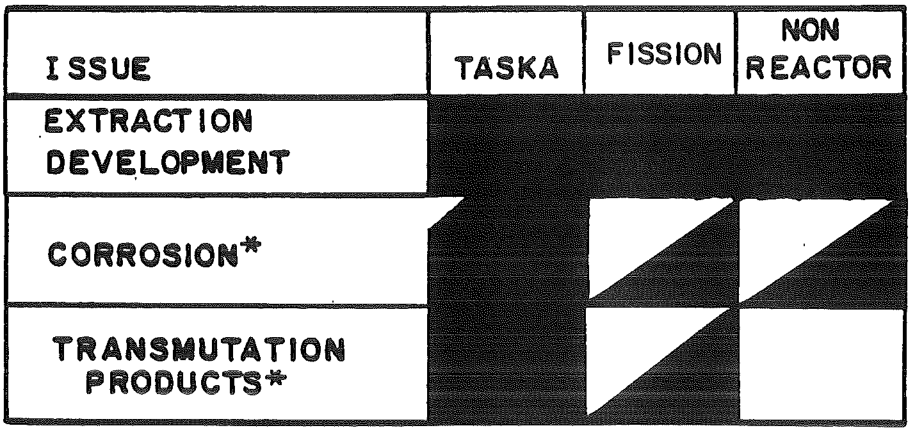
7.4 Tritium Breeding Systems

The ability of the pre-DEMO devices to provide sufficient information for tritium breeding and recovery in the DEMO is a subject of vital concern. Two types of lithium-containing breeder materials have been considered for the DEMO, namely, (a) self-cooled liquid metal breeders, such as Li or the liquid alloy, $Li_{17}Pb_{83}$, in which the liquid breeder flows directly to the steam generator, and (b) non-mobile ceramics, such as Li_2O, $LiAlO_2$ and Li_4SiO_4, which utilize inert coolants, water or helium, to transfer heat from stationary breeder blankets to the steam generator. Each type of breeder blanket concept requires a determination of (a) the ability of the blanket to achieve a tritium breeding ratio > 1.0, as discussed in Chapter 5, (b) an efficient technique to extract the tritium from the breeder so that a low tritium inventory is achieved, and (c) a method to maintain a low tritium pressure in the heat transfer fluid, to prevent excessive tritium losses in the steam generator (see Section 7.3.3.). In this section, we are principally concerned with the efficient extraction of tritium from the breeder and endogenous effects in the blanket modules, caused by radiation and the magnetic fields, which may interfere with the extraction process. Such information is urgently needed from the pre-DEMO experimental facilities. Detailed studies(17) are being conducted in order to assess the type of information which can be obtained and the most expeditious use of each experimental facility; see Chapter 5. At this time, therefore, we can compare these facilities only qualitatively.

7.4.1 Liquid Metal Breeders

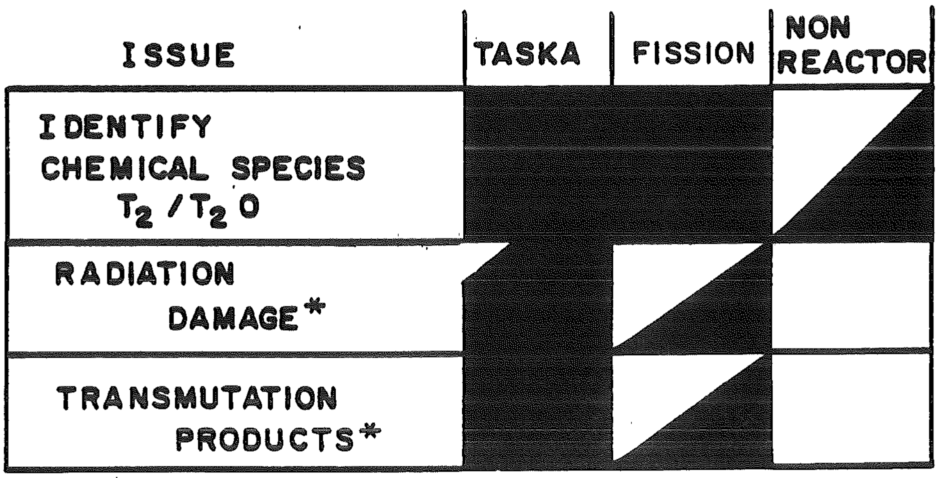
A tritium partial pressure of approximately 10^{-9} torr must be maintained in liquid metal breeder/coolants in order to minimize tritium permeation across the SG. Such a low tritium partial pressure can be accomplished in liquid lithium by the use of molten-salt extraction or yttrium getters.(18) Liquid lithium suffers because of a high tritium inventory and a high energy release during accidental mixing with water. Liquid $Li_{17}Pb_{83}$ has a much lower tritium solubility and lower rate of reactivity with water; however, the minimum tritium partial pressure which can be practically achieved is $\sim 10^{-4}$ torr; therefore, a tritium barrier is needed in the SG. Tritium extraction techniques for liquid lithium or liquid alloy breeders can be tested initially in non-reactor experiments followed by appropriate tests in fission and fusion reactors, as symbolized by the solid squares in Fig. 7.4-1. Tritium generation and extraction in the liquid breeder of an operational fusion reactor are subjected to additional complications caused by the neutron and gamma radiation and the presence of the magnetic field. These additional forces may lead to changes in the corrosion rate and transmutation products.

The effect of corrosion upon tritium extraction in a magnetically confined fusion device is not only a function of temperature but also of the presence of the magnetic field. The presence of the magnetic field causes "slug-flow"(19) which is neither laminar nor turbulent. TASKA mirror devices can generate such a flow pattern during corrosion tests; however, the geometry of the magnetic field in a mirror device does not duplicate some of the extreme variations in the angular forces present in a tokamak blanket. Corrosion tests in fission reactors would require the presence of a magnetic field, a difficult requirement. Non-reactor experiments in a magnetic field could mimic a fusion reactor blanket; however, the nuclear heating is not the



*IN PRESENCE OF NEUTRON
AND γ RADIATION AND
MAGNETIC FIELD

Fig. 7.4-1 Tritium breeding system/liquid metals (Li, $Li_{17}Pb_{83}$)



*IN PRESENCE OF NEUTRON
AND γ RADIATION AND
MAGNETIC FIELD

Fig. 7.4-2 Tritium breeding system/ceramic breeders (Li_2O, $LiAlO_2$)

same. Based upon these differences only partial benefits are shown for these latter two experimental devices, Fig. 7.4-1, while nearly full credit is given to TASKA.

Nuclear transmutation products may also interfere with the tritium extraction system, depending upon the chemical elements formed.[20] Because some transmutation products may be different in a fusion reactor than in a fission reactor, such studies can only be fully explored in fusion devices, such as TASKA. Again only partial benefit is given for the non-fusion devices, Fig. 7.4-1.

7.4.2 Solid Breeders

Solid breeder materials will require more extensive studies than for liquid breeders. This is principally due to that fact that tritium generation and extraction reaches a steady-state condition much more rapidly in a liquid breeder than in a solid breeder.

Recovery experiments of tritium from solid breeders have measured the tritium volatilization to a vacuum or a sweep gas. Present experiments have not conclusively established the chemical form of the tritium as either T_2 or T_2O, as it is released from a solid oxide breeder. Studies which unambiguously establish the chemical composition of the species leaving the surface of the solid breeder must be designed. Non-reactor experiments can only be accomplished in samples which were previously irradiated in a reactor; hence, only partial credit is given for non-reactors in Fig. 7.4-2.

In regard to in-reactor tritium generation and release from solid breeders several synergistic effects must be considered, namely, atomic lattice damage, sintering caused by thermal gradients and radiation,[21] swelling, and the effect of the transmutation products from the neutron radiation. The effects of sintering and radiation damage upon tritium release are crucial issues because such damage can significantly retard tritium migration. Until such effects are understood, the equilibrium tritium inventory and rate of release cannot be adequately evaluated. For instance, various predictions of tritium inventory[22] in a lithium oxide blanket for a 1000 MW fusion reactor range between 25-50,000 g. TASKA facilities are adequate, therefore, to test most of the radiation effects for the ceramic DEMO breeder, although the exact geometry of the tokamak blanket cannot be duplicated. The n-γ spectra for fusion and fission reactors are different; however, in Chapter 5 it is shown that the radiation damage caused by the bred T and He atoms is very similar in either reactor. A number of fission test reactors can be used, therefore, to test tritium release if comparable thermal gradient and sweep gas configurations are achieved.

Transmutation products may also retard tritium release by reacting with tritium atoms or blocking the diffusion path for tritium release. This effect is only prevalent for several ternary oxides, such as $LiAlO_2$ in which the Al forms transmutation products. Because some of the transmutation products are different in fusion and fission reactors, such studies are preferably accomplished in TASKA type devices, although much information can be obtained from fission reactors. Therefore, nearly total credit is given for TASKA while approximately 50% credit is given for fission reactors in regard to both for radiation damage and transmutation studies in Fig. 7.4-2.

7.5 Conclusions

1. A TSTA-type facility will demonstrate (a) plasma exhaust and fuel reprocessing (chemical and isotopic) at a rate equal to 50-60% of that required for the DEMO; (b) tritium containment facilities within 2 orders of magnitude of the DEMO, but of the same essential type; (c) tritium monitoring and control capabilities within 30% of the DEMO, although T_2 monitoring in the presence of gamma and neutron radiation will not be present.
2. A TSTA-type facility will not provide, because of the absence of a plasma, information on (a) fuel delivery to the plasma, (b) plasma wall interactions, (c) composition of impurities in the plasma exhaust and, (d) any knowledge of tritium breeding.
3. The experimental plasma facilities TFTR, JET, and TFCX will provide little information regarding the tritium fuel cycle because of the short burn time. Tritium breeding experiments will be of limited value because of the low fluence; however, verification of tritium release models at low burnups may be possible, see Chapter 5.2. Preliminary information regarding the plasma exhaust composition and plasma wall interactions may be obtained.
4. If the TASKA facilities are constructed and provide the necessary information before the DEMO design is completed, the following benefits are obtained:
   a) The tritium systems required for the DEMO can be safely extrapolated from those of TASKA, except for plasma-wall interactions.

b) A TSTA upgrade for study of the fuel cycle technology will not be required. Certain large valves, etc., may have to be tested in TSTA before TASKA can be built.

c) Specialized laboratory experiments involving tritium will be required to study plasma wall interactions such as tritium implantation, reemission and diffusion. Also, laboratory experiments need to be continued to assess the use of barriers to prevent tritium contamination of the heat transfer fluids. Preliminary information regarding these phenomena is needed for TASKA designs.

d) Additional tritium breeder characterization studies must be initiated and breeding experiments planned for fission reactors. Fusion relevant tritium breeding and release studies will require the use of TASKA or NET-EP facilities.

Such conclusions, of course, must be updated as new experimental information is obtained and as the priority develops for the introduction of fusion power into the world's economy.

References for Chapter 7

1. U.S. DOE: priv. communication, "Generic Environmental Impact Statement", Magnetic Fusion Energy Program, Draft 1983

2. S.J. Piet and M.S. Kazimi, "Implications of Uncertainties in Modelling of Tritium Releases from Fusion Reactors," Proc. ANS Mtg. Tritium Technology in Fission, Fusion and Isotopic Applications, April 1980, Dayton, Ohio, pp. 25-31.

3. T.B. Rhinehammer and P.H. Lamberger, "Tritium Control Technology," USAEC Report, WASH-1269 (Dec. 1973).

4. K.Y. Wong, et al., "Canadian Tritium Experience," Canadian Fusion Fuels Technology Project, Mississauga, Ontario, 1984

5. L.J. Wittenberg (Ed.), Proc. of ANS Meeting on Tritium Technology in Fission, Fusion and Isotopic Applications, Dayton, Ohio, April 1980.

6. J.L. Anderson, "Design and Construction of the Tritium System Test Assembly", Proc. ANS Mtg. Tritium Technology in Fission, Fusion and Isotopic Applications", April 1980, Dayton, Ohio, pp. 341-346.

7. R.A. Kerst and W.A. Swansiger, "Plasma Driven Permeation of Tritium in Fusion Reactors", J. Nucl. Mat. 122 & 123 (1984) 1499-1510.

8. W.M. Stacey, et al., "U.S. Contribution to INTOR, Phase-I Workshop", June 1981, Chapter XI. (unpublished)

9. "TFCX Preconceptual Design Report - Vol. 1", Princeton Plasma Physics Lab. - F-Axxx-8407-006, 30 Sept. 1984.

10. K.E. Lind et al.: "The Preliminary Safety Analysis Report for the TFTR", Proc. 3rd Top. Meetg. Techn. Control. Nucl. Fus., Santa Fe, May 1978, p. 924

11. H. Phillips et al., "Tritium Storage and Delivery System for TFTR", Proc. ANS Mtg. Tritium Technology for Fission, Fusion and Isotopic Applications", Dayton, Ohio, April 1980, pp. 353-359.

12. B.G. Logan et al., "Mirror Advanced Reactor Study", Lawrence Livermore National Laboratory, UCRL-53480, July 1984, Section 22.

13. J.T. Bell, J.D. Redman and H.F. Bittner, "Tritium Permeability of Structural Materials and Surface Effects on Permeation Rates", Proc. ANS Mtg. Tritium Technology for Fission, Fusion and Isotopic Applications, Dayton, Ohio, April 1980, pp. 48-53.

14. V.A. Maroni, E.H. Van Deventer, T.A. Tenner, R.H. Pelto and C.J. Wurdale, "Experimental Investigations of Tritium Barrier Concepts for Fusion Reactors", Gatlinburg, Tennessee, October 1975, CONF-750989, pp. 329-361.

15. D.C. Schluderberg, J.H. Huang, L. Pong and D.K. Sze, "Thermal and Mechanical Design of a Double-Walled Steam Generator", Nuclear Tech./Fusion 4 (1983) 1509-1513.

16. D.F. Holland, Idaho Nuclear Eng. Lab., private communication, 1984.

17. M.A. Abdou et al., "FINESSEE: A Study of the Issues, Experiments and Facilities for Fusion Nuclear Technology Research and Development", University of California, Los Angeles, PPG-821 (October 1984).

18. M.A. Abdou et al., "Blanket Comparison and Selection Study", Argonne National Lab., ANL/FPP-83-1 (October 1983) pp. VII-90 to 105.

19. M.A. Abdou et al., "FINESSEE", op cit., pp. 6-13 to 6-48.

20. M.A. Abdou et al., "Blanket Comparison and Selection Study", Argonne National Lab., ANL-FPP-83-1 (October 1983) pp. XI-15 to 20.

21. M.A. Abdou et al., "Blanket Comparison and Selection Study", Argonne National Lab., ANL-FPP-83-1 (October 1983) pp. VIII-1 to 21.

22. M.A. Abdou et al., "A Demonstration Tokamak Power Plant Study (DEMO)", Argonne National Lab., ANL/FPP-82-1 (September 1982) p. 5-152.

8. Vacuum, Exhaust and High Heat Flux Components

8.0 Introduction

Impurity control, ash removal and the related aspects of the vacuum and exhaust system are of great importance for reliable operation and economic success of all types of fusion reactors. With the presently available experimental devices only some basic questions within this scenario are addressed, mainly in the field of plasma physics. Also the two new larger facilities, TFTR and JET, will provide only limited contributions, even if they can demonstrate the ignition of D-T plasma (e.g. they do not have a pumped limiter or a divertor). Some valuable information about the characteristics of the divertor can be expected from the operation of ASDEX-Up /1/, which is under construction at present. Unfortunately, this device is not designed for tritium operation. To answer the great number of technology questions in the field of vacuum and exhaust therefore requires new test facilities.

However, the specification of the testing needs is handicaped by the fact, that up to now no uniform and generally accepted working concept for these processes can be outlined. Actually, a number of options are only discussed. In many key areas of them the physical feasibility is not even proven.

One drastic example for instance is the question, whether steady-state operation can be realized for tokamak-plants, as it is postulated in the latest US DEMO-Study /2/. In this case all material research topics related to thermal cycling, fatigue, crack propagation etc. and some questions of cyclic hydrogen adsorption/desorption on the plasma chamber wall structures would be simplified.

Another example is the question to what extent and for which geometric conditions a cold and dense plasma can be realized under actual operating conditions in the vicinity of the main plasma-material interacting surfaces. If this cold plasma were to exist, the sputtering problems and the associated problems created by the sputtered material are relieved and the pumping speed required is reduced. At present it is not possible to choose intelligently among limiter or divertor design concepts for impurity control. In view of this situation the following text does not distinguish between these two solutions. Only the common aspects valid for both concepts are discussed.

In this context, the material selection for the plasma interacting surfaces should be mentioned. For the bulk of the limiter surface even with relatively low edge plasma temperatures of < 100 eV only low Z-materials (e.g. Beryllium) with its high sputtering/redeposition rates may be acceptable because of the close vicinity of these materials to the main plasma. In contrast, for the divertor, which is located further away from the fusion plasma, a cold divertor plasma regime would possibly permit the use of high Z-materials such as Tungsten or Molybdenum and there seems to be a realistic chance to keep the mean ion energy in front of these plates below the threshold of the sputtering process (< 30 eV), so that long lifetimes could be expected.

The vacuum and exhaust system of a fusion plant must generate the conditions required within the plasma chamber (1) for the first startup after completion of construction and (2) for all subsequent startups after component replacement or repair (startup gas pressure $\lesssim 10^{-5}$mbar). In normal plant operation, it must (3) exhaust the helium (the fusion ash) and the impurities emanating from the first wall surface. Even if in this case the pressure of the high temperature plasma reaches some bars, the neutral gas pressure within the scrape-off layer close to the surrounding first wall surfaces must be kept at the level of approximately 10^{-4} to 10^{-5} mbar by appropriate vacuum pumping. Otherwise, the thermal load on these surfaces would reach unacceptable levels and the main plasma would become excessively poisened by atoms sputtered from the material surfaces. Finally, in case of a pulsed device the vacuum and exhaust system has (4) to recondition the plasma chamber after each burn (during the dwell time) for the next pulse.

To accomplish these tasks, new requirements must be satisfied. Compared to the existing fusion plants (e.g. JET, TFTR, ASDEX-Up) the vacuum chamber of a DEMO will be much more complex in design. Because of the numerous components which have to be installed within it (e.g. blanket modules), a large number of small gaps, cavities and crevices will exist and the total material surface area exposed to the vacuum will be much larger (up to 10 000 m^2, depending on the blanket design).

Furthermore, during the plasma burn complete equilibrium must be established with respect to impurity control and fuel content in the plasma chamber to achieve long burn times. Finally, in case of a pulsed device, the dwell time between burn times must be kept short to get high integrated neutron flux doses for test purposes and to demonstrate the potential of the pulsed tokamak for power production.

The evacuation procedures for the first startup will be similiar to those of large conventional vacuum installations and previous fusion devices. Some complications may arise because some of the most effective surface cleaning and degassing procedures will not be applicable to large portions of the internal surfaces and some specific materials with bad degassing properties may have to be used (e.g. electrically insulating contact surfaces between adjacent blanket modules).

The evacuation after repair is similiar to the first task. However, some important factors will be different because of the preceding power operation. Large amounts of the hydrogen-isotopes will be stored in the materials and the surface conditions will be different as a result of the erosion and redeposition of sputtered material. It is expected that these surface conditions enhance gas trapping rates so that more gas must be evacuated to get the required vacuum conditions for startup. An opposite effect can be expected from the strong ionizing radiation of the stored tritium and the highly activated blanket and structural materials /3; 4/. At present, reliable quantitative values are not available in these fields. Further research activities will be necessary.

Helium and impurity flows during the plasma burn are determined primarily by the ion, neutral and impurity transport within the plasma scrape off layer and the divertor/limiter region. For this operational regime, the divertor/limiter plates for conversion of the escaping energetic plasma particles to neutral gas must be provided. This implies that the metal atoms liberated from these dump plates by sputtering processes are also properly removed and do not enter the main plasma. Of great concern during the plasma burn are also the hydrogen recycling effects, i.e. the repeated interchange of hydrogen fuel between the plasma and first wall or other surfaces. With a pulsed operating regime, balancing molecular vacuum flow conditions will prevail in a periodic manner in the great number of gas filled gaps connected to the core of the plasma chamber.

The fourth task (the reconditioning of the plasma chamber after each burn) is relevant only for pulsed devices. The fusion process gases and the different kinds of impurities generated by plasma-wall interactions must be exhausted before the next startup. Here some similarity with task 2 may exist, e.g. the tritium outgassing characteristics of structural materials. The main goal is the realization of short dwell times (< 60 s).

In the following sections, the research and development needs of the main topics in the field of vacuum, exhaust and high heat flux components are listed and briefly described. Table 8-1 covers the basic process phenomena, Table 8-2 the technology issues of the components within the vacuum boundary and Table 8-3 the technology of the external components. The symbols used should be interpreted as follows:

\+ the facility is applicable for research in the field of concern. It is expected, that results can be achieved on which the construction decision for a tokamak DEMO can be based.

(+) the facility has some potential for specific research in the field of concern, so that its use is worthwhile. However, a clear gap exists to the real target needs of a tokamak DEMO.

(-) the facility will provide some useful information mainly in the course of its scheduled operation modes. Definite research programs in the field of concern seem not justified.

\- the facility has no common features with the needs in the field of concern, or, even if there are some common features the gap between the needs and the capability is so wide, that neither in normal operation nor with specific applications valuable results for a tokamak DEMO can be provided.

Tab. 8-1: Issues for Vacuum, Exhaust and High Heat Flux Components: Basic Process Phenomena

R + D Needs	Target Needs and Data for tokamak D E M O	Fusion Facilities NET-P	Fusion Facilities TASKA	Plasma devices	Non fusion Facilities Fission Reactors	Non fusion Facilities Simulation Tests
8.1.1 Plasma-Materials Interaction (bulk material highly irradiated, surface morphology according to sputtering/ redeposition equilibrium)						
.1 Hydrogen recycling effects (Reflection, desorption processes, solid state transport and thermal release)	Wall conditioning for ignition; control of fuel content in plasma chamber during burn; Tritium confinement (safety)	+	(+)	(+) JET, TFTR ASDEX-Up	-	-
.2 Impurity Generation (Impurity desorption, physical sputtering, evaporation, sublimation, melting, chemical erosion, arcing, helium trapping and blistering)	Dependent on edge plasma physics (and recycling of divertor plasma) max. erosion rates: ≈ 5 mm/FPY for divertor/limiter plates	+	+	(+) JET, TFTR ASDEX-Up MFTF-B	-	(+) PISCES
.3 Disruptions	Armor plates made of refractory material required as long as plasma control cannot avoid disruptions	+	-	(+) JET, TFTR	-	-
8.1.2 Sputtering Material						
.1 Properties of redeposited material (bonding stability with base material; thermomechanical, sputtering, hydrogen adsorption, and energy reflection properties)	Input for important design decisions and corresponding calculations	(+)	+	(-) ASDEX-Up	-	(+) PISCES
.2 Inherent deposition characteristic	Time and space dependence over plant (component) lifetime for forecast of realistic operation scenarios	(+)	(+)	(+) JET, TFTR TEXTOR ASDEX-Up	-	-
.3 Controlled precipitation	Probably indispensable for reliability of external vacuum components	(+)	+	-	-	(+)
8.1.3 Standard Vacuum Issues						
.1 Wall conditioning	Complex design of plasma chamber (gaps, cavities, special materials) and equilibrium surface morphology require new techniques	+	+	-	-	(+)
.2 Gettering	Probably potential for realization of short dwell time	not yet defined		(+) TFTR, MFTF-B	-	(+)
.3 Flow conductance (in gaps, channel networks and ducts, without and with penetrating plasma or neutral particle flux)	To discard any excessive safety margin to realize optimal magnetic plasma confinement neutron shielding and short dwell time	(+)	(+)	-	-	(+)
.4 Leakages (Weldments, mechanical seals, electrical penetrations)	To boost the technological state of the art of large vacuum process production units	(+)	(+)	-	-	(+)
.5 Leakage detection and localization for complex plasma chambers	Indispensable to get the required availability values of the plant	(+)	(+)	-	-	(+)

Tab. 8-2: Issues for Vacuum, Exhaust and High Heat Flux Components: Technology of Internal Components

R + D Needs	Target Needs and Data for tokamak D E M O	Fusion Facilities NET-P	Fusion Facilities TASKA	Plasma devices	Non fusion Facilities Fission Reactors	Non fusion Facilities Simulation Tests
8.2.1 Plasma and Particle Flow Stop Plates (Divertor, Limiter, Beam Dumps)						
.1 Neutron irradiation behavior	Input for 8.1.1.1; 8.1.3.4; 8.2.1.3/.5	see Chapter 6		-	see Chapter 6	-
.2 Sputtering and redeposition (see 8.1.1.2 and 8.1.2.1/.2)	Definition of thickness of sacrificial layer and time history and distribution of thermal loading	+	+	-	-	-
.3 Thermomechanical stress-strain behavior (low cycle fatigue, thermal ratchetting, crack propagation)	To allow realistic lifetime forecasts; Target: lifetime > 1 FPY (pulsed operation > 10^5 burn cycles over lifetime)	(+)	(+)	(-) ASDEX-Up	-	(+)
.4 Bonding stability between substrat and protection plate or sacrificial cladding	To allow realistic lifetime forcasts; Target: lifetime > 1 FPY	(+)	(+)	-	-	(+)
.5 Effectivity and stability of Tritium barriers (Tritium solid state transport see 8.1.1.1)	Realistic assessment of primary tritium confinement, input for secondary containment and tritium recovery facilities	+	+	(+) JET, TFTR	-	(+)
8.2.2 Divertor/Limiter and Beam Dump Inserts with Particle Flow Stop Plates						
.1 Shape distortion effects (swelling, creep, thermal ratchetting, sputtering and redeposition)	To include the feed back effects in the final design and the specification of the loading conditions	(-)	(-)	-	-	-
.2 Overall performance	To allow realistic reliability and lifetime forecasts	(-)	(-)	-	-	-
8.2.3 Other components within the vacuum chamber						
.1 First Wall	Definition of thickness of sacrificial layer and other design parameter	see Chapter 6		-	see Chapter 6	-
.2 Plasma Heating devices	To allow realistic lifetime forecasts; Target: lifetime > 1 FPY	see Chapter 4		see Chapter 4	-	-
.3 Intrasector electrical connectors, insolating support pads, etc.	Requirements of vacuum conditioning and impacts of material deposition must be satisfied	+	+	-	-	-
8.2.4 Exhaust Duct (see 8.3.1)						
.1 Flow conductance (for section close to Divertor/Limiter see 8.1.3.3)	To discard any excessive safety margin in order to realize optimal magnetic plasma confinement, neutron shielding and vacuum pump dimensioning	(+)	(+)	-	-	(+)
.2 Special shielding (neutron shielding close to superconducting coils and magnetic shielding close to turbomolecular pumps)	Limitation of max. neutron irradiation dosis and max. magnetic field	(+)	(+)	-	-	(+)

Tab. 8-3: Issues for Vacuum, Exhaust and High Heat Flux Components: Technology of External Components

R + D Needs	Target Needs and Data for tokamak D E M O	Fusion Facilities NET-P	Fusion Facilities TASKA	Non fusion Facilities Plasma devices	Non fusion Facilities Fission Reactors	Non fusion Facilities Simulation Tests
8.3.1 Exhaust Duct (see 8.2.4)	high conductivity in order to realize optimal vacuum pump dimensioning	+	+	-	-	(+)*
8.3.2 Gate Valves	proven design for diamters of 1,5 to 2 m with high reliability and leak tightness	(+)	+	-	-	(+)
8.3.3 Turbomolecular Pumps or Cryocompound Pumps	proven design for oil free gas flow and pumping speeds of > 50 m^3/s	(+)	+	-	-	(+)
8.3.4 Roughing Pumps	proven design for oil free gas flow and pumping speeds of > 0,5 m^3/s at 0,3 mbar	+	+	-	-	(+)

* only for controlled precipitation of sputtered material

8.1. Basic Process Phenomena (Tab.8-1):

8.1.1 Plasma-Materials Interaction (PMI)

All plasma wall interaction phenomena strongly dependend on edge plasma physics, which is not covered under this Chapter. The main technological issues and questions for a tokamak device are the following /5/:

1) Will poloidal divertors be necessary to solve the impurity control and ash removal problems or can pumped limiter schemes be devised to handle these tasks? Will supporting techniques, e.g. gettering, be necessary?

2) The effects of long pulse discharges on the "conditioning" of the first wall, and on the divertor/limiter structures with respect to impurity generation, impurity redeposition and hydrogen recycling effects.

3) For routine operation with long pulse length and short dwell times, the reliability, availability and lifetime of all components.

In mirror machines the situation is somewhat different. There is some prospect, that plasma-materials interactions can be minimized /6/. This is due to the inherent open field lines of the confinement configuration. In particular, plasma-surface interactions can be controlled here at radial surfaces by controlling the axial confinement of the edge plasma.

8.1.1.1 Hydrogen Recycling Effects

The issue of recycling is closely coupled to the conditioning and types of wall materials. During long term operation the specific morphology of the first wall surface generated by sputtering and redeposition will be of decisive impact. In case of graphite, for example, it may well be that preparation techniques can be devised to make the surface less able to soak up tritium. This could have important consequences when tritium inventory is examined.

Complete investigation of these phenomena is only possible in full sized fusion plasma facilities of the NET-P/TASKA-class. However, because of the fundamental impact of the material selection for the first wall components on the hydrogen recycling effects, adequate simulation experiments must be carried out before a real fusion facility can be finally specified. For NET-P this question is of greater importance than for TASKA-class machines because of the larger particle wall load and the pulsed operation mode with its frequent change of the material temperatures.

Some first preliminary information can be expected from the operational experience of the JET/TFTR class plasma devices provided that adequate R and D efforts are devoted to this subject at these plants. Unfortunately, the short pulse lengths and relatively long dwell times, the almost not existing neutron irradiation and the very low sputtering and redeposition effects will limit the usefulness of these results. A similar situation will exist with ASDEX-Up.

More realistic surface conditions can be realized with the help of steady state Plasma-Material-Interaction (PMI) experiments on the basis of the mirror confinement concept, as presently discussed /7/. Unfortunately, neutron irradiation is missing also in these experiments. A combination with irradiation experiments in fission reactors would be necessary, if neutron radiation is expected to show a larger impact.

8.1.1.2 Impurity Generation

The experimental situation here is similar to the one for hydrogen recycling effects. Only the operation of real fusion plants of the NET-P/TASKA-class will bring the final proof. Devices like JET, TFTR and TEXTOR will produce some useful information about impurity generation, but the before mentioned limitations of these facilities are in general also valid for this topic. There are some indications, that the halo and plasma end dumps of MFTF-B and the divertor plates of ASDEX-Up may be useful for somewhat more realistic sputtering investigations, but the real potential of these lines was not yet assessed in detail.

Almost all available data of sputtering yields for ion bombardment were generated under idealized laboratory conditions with concentrated energetic ion beams on small samples of pure materials. These scientifically oriented basic studies should be complemented in the future by technological tests with more realistic and larger test specimen. The PISCES device of the University of California, Los Angeles, is a first step in this line. The proposed class of PMI experiments on the basis of the mirror confinement concept /7/ are primarily devoted to this kind of research. They can realize an adequate electrostatic sheath potential in front of the test surface by impinging electrons, so that the combined effects of ion sputtering and selfsputtering can be studied.

8.1.1.3 Disruptions

Plasma disruptions are associated with tokamak. Improvements in plasma control must bring down the frequency of disruptions to an adequately low level, so that they become "infrequent operational occurrences" and the resulting interruptions and disturbances of the operational sequence of the plant can be tolerated. Also the severity with respect to local wall material loss and the stress impacts by magnetic forces must be kept within tolerable limits. In the meantime, the consequence of disruptions on the plasma chamber walls and on the overall structure of the device should be carefully analysed.

8.1.2 Sputtering Material

8.1.2.1/.2 Properties of Redeposited Material and Inherent Deposition Characteristic

The chemical, physical and technological properties of the redeposited sputtered material within the plasma chamber and the exhaust ducts of a large fusion energy production unit is one of the most important open questions in the field of fusion technology. The spectrum of opinions extends from the assumption that the redeposited sputtered material is solidly bound to the substrate and has nearly the same properties as the original state, up to the other extreme that the redeposited material flakes off from the substrate as soon as a certain (small) thickness is reached and distributes within the system as macroscopical "dust".

In present plasma devices, no greater quantities of sputtered material are produced so that realistic investigations about the situation in a DEMO cannot be made. However, it was observed, that each specific kind of material liberated at a certain location, distributes across the whole inner surface of the plasma chamber and that areas of greater distance from the core plasma show larger redeposition rates. More information can be expected from the operation of NET-P and especially from the steady state operation of TASKA. Also ASDEX-Up shows some promise because of the larger power density of its divertor.

The existing uncertainties make it very difficult to define important parameters for a "conceptual design" of a large fusion plant at present (e.g. loading characteristic and lifetime of divertors and limiters, hydrogen adsorption and recycling of the first wall, operating conditions for external vacuum components). It is therefore one of the most urgent R and D needs to get more insight into these phenomena.

The new class of PMI experimental facilities mentioned before /7/ are effective means for this purpose also. Their steady state operation regime will allow to build up layers of redeposited material of realistic thickness under controlled environmental conditions (e.g. with respect to small amounts of trace elements in the low density hydrogen atmosphere), which is the indispensable precondition for any true technological research in this field. At the same time, the PMI zone is large enough to also study the influence of the magnetic field on the space distribution of the redeposited material across the sample plate and the adjacent surfaces of the machine.

8.1.2.3 Controlled Precipitation

Within the present R and D activities the controlled removal of sputtered material from the fusion device is not addressed adequately One reason for this is the fact that at present large discrepancies exist in the expectations of how these materials may behave.

Proposals for suitable active control measures of the sputtered material are therefore naturally spare at present. In /5/ it is mentioned in connection with the composition of the exhaust gas that "The metallic impurities can be removed by electrostatic precipitation". In the INTOR-Study, Phase one /8/ a "Debris Separator" is shown interconnected between the divertor and the D, T cryocondensation pumps in the fuel cycle flow diagram (page 537). Finally, in the latest US DEMO study /2/ it is stated on page 5-145: "Any particulate debris (possibly gamma-emitting) generated in the plasma chamber is captured on filters located between the cryopumps and the roughing pumps".

For the corresponding research needs the same arguments hold as described for the proceeding topic. The steady state operation mode of TASKA should be advantageous in this field. For screening tests and parametric studies, simulation facilities will be useful.

8.1.3 Standard Vacuum Issues

The plasma chamber of any real fusion facility, which would be typically several hundred to one thousand cubic metres, and generally of complex design, must be evacuated down to $< 10^{-5}$ mbar at the initiation of an operating period, and, in case of a pulsed device, between burn times. The ducts and cavities in the neutral beam and rf heating systems, with volumes of several tens of cubic metres, must be maintained even at $\simeq 10^{-7}$ mbar. These substantial vacuum pumping requirements are common to all confinement concepts /9/.

During burn, the exhaust gas from the plasma chamber is expected to have roughly the following atomic composition: Deuterium-Tritium, 85 to 95%; hydrogen, 1 to 10%; helium, 5 to 10%; oxygen, nitrogen, and carbon, $\simeq$ 1%; and metallic impurities, $\simeq$ 0.01%. Oxygen, nitrogen, and carbon are present mainly in the form of the chemical compounds water (DTO), ammonia $N(D,T)_3$, and methane (CD_2T_2) /9/.

8.1.3.1 Wall Conditioning

On presently operating, short pulse plasma devices with smooth inner surface of the plasma chamber, the problems associated with gaseous low Z impurity species (H_2O, CH_4, CO_2), which are loosely bound to the first wall surfaces, have largely been solved. Well established techniques of vacuum preparation, baking, and glow and pulsed discharge cleaning can be applied, so that within a reasonable period of time after exposure of the system to air, relatively clean, reproducible plasma discharges can be obtained. In some respects, present conditioning techniques may not extrapolate to long pulse devices /9/ and to a DEMO. With continuous operation or when pulse lengths are in the range of fractions of one hour, it will be the hot, high density discharge itself which will "condition" the walls, and some experimental simulation of the effects of this must be undertaken. Another problem concerns the complex design of the vacuum chamber of any new fusion facility. As the vacuum boundary will be situated outside of numerous components such as the blanket/shield structure, there are large remote volumes and surfaces communicating with the plasma chamber proper, and a large number of small gaps, cavities and crevices. These remote volumes and surfaces may jeopardize obtention of the required vacuum conditions, as furthermore some of the most effective surface cleaning and degassing procedures, respectively, will be not applicable to large portions of these surfaces.

8.1.3.2 Gettering

Getter, or sublimation pumps consist of plates of chemically active metals such as titanium or zirconium-aluminum that chemically react with gases (e.g., hydride formation with hydrogen, deuterium, and tritium) and can operate at elevated temperatures. TFTR makes extensive use of solid zirconium-aluminium getter panels within the plasma chamber. In MFTF-B and its forerunners, Beta II and TMX, the sublimated-titanium type gettering method is used. Heating current is applied through Ti wires and a Ti coating, a minimum of 3 monolayers thick, is sublimated onto the large inner surfaces of the end cells, thus providing a clean condition that will trap energetic deuterons by ion implantation with very little reflux.

It is not yet clear, whether the getter technique will be necessary in future fusion reactors and with which design solutions and process methods it could be applied. Among others the frequent liberation of the absorbed tritium would be necessary.

8.1.3.3 Flow Conductance

In present day experimental fusion devices and in large conventional vacuum chambers satisfactory pump down and operational vacuum behavior is in general realized by overdimensioning the vacuum pumps and the duct size to cope with all uncertainties of the real technical design. For a fusion DEMO this approach is not applicable, because the capacity of the vacuum pumps is on the technical limit and the duct, gap and channel sizes within the device must be optimized in conformity with the magnetic confinement, breeding, shielding and space requirements. For this purpose a complete simulation of the whole vacuum chamber is required, to get the time-dependent gas density distribution within it for all start-up and operational conditions. This includes the necessary experimental verification by appropriately scaled laboratory set ups.

A thoroughly investigation of the operational characteristics of NET-P/TASKA will also provide a data basis in this field for verifying corresponding computer codes.

8.1.3.4 Leakages

A large fusion energy production facility like a DEMO will have to cope with leaks from probably more than 50 complex blanket subassemblies, from provisions for plasma heating and from diagnostic and numerous other kinds of equipment within the vacuum chamber (coolant and heating circuits, process instrumentation etc.).

To keep the leak rate as low as possible, the state of the art in vacuum technolgy must be continuously improved. This concerns also the remotely controlled rewelding of weld seals after repair measures on highly radiated and hydrogen saturated material.

8.1.3.5 Leakage Detection and Localization

An effective and reliable system for leakage detection and especially leakage localization will be of uppermost importance to get acceptable availability figures for fusion plants. At present, no adequate procedure or technical concept is known for this purpose. This underlines the need for a highly leakproof system just from the beginning, as discussed before. Also the simulation and later the surveillance of the time dependent gas density distribution within the vacuum boundary as discussed under 8.1.3.3 must be seen in this context, because this could provide a basis for the development of an appropriate leak detection and localization system.

8.2 Technology of Internal Components (Tab.8-2)

All components of the vacuum and exhaust system are highly contaminated by tritium and other radioactive matter. In addition, the parts installed in close vicinity to the burning plasma, e.g. the high heat flux components, become activated by the strong neutron flux in this areas. Therefore all surveillance, service and repair measures must be carried out completely under remote control, and the technical design of the components has to take into account these conditions.

This statement holds also for the external components according Sect. 8.3. However, the most severe requirements in this field exist for the internal components, because they must be designed for the given space limitations.

8.2.1 Plasma and Particle Flow Stop Plates (Divertor, Limiter, Beam Dumps)

The particle flow stop plates are subject of the uppermost loading conditions in a fusion reactor. The very high heat loads (up to 10 MW/m^2) result in severe thermomechanical stress-strain conditions and the strong particle impingement (> 10^{18} particles/cm^2 *s) leads to remarkable surface erosion. In addition, the plates are irradiated by the strong flux of the 14 MeV fusion neutrons. The combined effects of these loadings constitute extreme operational conditions.

Any research and development program must first concentrate on "Separate Effects Tests", as described under 8.1.1 and 8.1.2 to better understand the basic phenomena. Next, potential solutions for promising technical concepts can be defined, and finally, a meaningful integrated test program for the particle flow stop plates of a fusion DEMO can be started.

At present, the technical data base in important areas is by far not adequate, in some areas it does practically not exist (e.g. redeposition of sputtered material under representative working conditions).

8.2.1.1 Neutron Irradiation Behaviour

The main neutron irradiation effects on material properties as e.g. the changes of the stress-strain and ductility values and the values of irradiation induced swelling and creep are known from fission reactors for its neutron energy and flux levels. However, instead of less than 100 dpa in the core components of fast breeder reactors more then 200 dpa in the first wall of a fusion DEMO must be expected. At present, there is no test facility available, in which samples of technical size can be irradiated up to this level. Therefore, a pragmatic, stepwise approach must be applied (see Chapter 6).

In addition to the classical neutron irradiation effects on materials the strong neutron flux of the fusion process will also affect some of the basic process phenomena and design aspects as discussed above.

8.2.1.2 Sputtering and Redeposition

Because of lack of basic information in this field any forecast for the condition of a DEMO is highly uncertain at present. It is expected, that the test results of NET-P and TASKA will make it possible to define and work out a promising concept for the particle stop plates of a DEMO.

8.2.1.3/.4 Thermomechanical Stress-Strain Behavior, Bonding Stability between Substrate and Protection Plate, Stability of Tritium Barriers

Besides the loading parameters, the physical size and shape of the test plate will be of importance for the results. Neither NET-P nor TASKA will allow a complete simulation of the DEMO stop plates, but it is expected that the tests which can be carried out are adequate to justify the construction of a DEMO. Additional information can be expected from the operation of ASDEX-Up and appropriate simulation tests.

8.2.2 Divertor/Limiter and Beam Dump Inserts with Particle Flow Stop Plates

The wear and shape distortion effects originated by neutron induced swelling and creep, thermal ratchetting, sputtering and redeposition and the overall performance of the whole insert can be finally verified only in the DEMO plant itself. However, these inserts are designed as replacable parts. Their design details can be subject to final improvements later, according to the actual operational experience gained. Therefore the envisaged development scheme is acceptable, provided that the result of the preceeding tests can justify the assumption that the overall financial risk can be kept in acceptable limits. It should be mentioned, that shape distortions in most cases will have a strong feedback on the loading conditions, because the angle between the incident particle flux and the stop plate surfaces is generally small, so that even small distortions in the solid structure will alter the loading conditions remarkably. Ingenious technical designs will be required for these components.

8.2.3 Other components within the vacuum chamber

A fusion DEMO will have a number of further components in direct contact with the vacuum and plasma space, some of which may even not be known yet. First of all, there is the large first wall area surrounding the fusion plasma with its protecting plates or sacrificial layers. For these components, roughly the same kinds of loading conditions are valid as for the particle flow stop plates described before, but fortunately, the thermal and particle flux load values are remarkably lower (< 2 MW/m^2; $< 10^{17}$ particles/cm^2 * s). Furthermore, there will be different kinds of plasma heating devices as described in Chapter 4. Also special magnet coils may be necessary in a tokamak for active stabilization of the plasma as well as remotely operated, passive electrical connectors between adjacent blanket modules at defined locations to control the eddy currents induced in the structural parts. At other locations electrically insulating support pads will be required for the same purpose. Finally, a large number of different kinds of sensors for plasma diagnostic and process control will be located within the vacuum space (see Chapter 10).

The test conditions provided by NET-P and TASKA will satisfy the needs of these components to different degrees, as indicated in Tab. 8-2. The main bottle neck are the high neutron fluence requirements of a DEMO for the first wall and other structural parts located in a similiar radiation environment.

8.2.4 Exhaust Duct

The first section of the exhaust duct from the divertor/limiter to the outer boundary of a tokamak device must be designed under severe space limitations, because of the shielding requirements of the adjacent coils. A careful optimization must be carried out to get the maximum conductivity values achievable.

This task is aggravated by the fact that the Maxwell distribution of the exhaust particles will be heavily disturbed in the vicinity of the divertor/limiter plates. The interaction of highly energetic plasma and neutral particles with the wall surfaces in the first part of this duct section, the resulting erosion processes and generally the overall process control of the volatile sputtered material are further problem areas which will require experimental support. It is expected, that both NET-P and TASKA can satisfy these requirements in combination with appropriate test set ups.

8.3 Technology of External Components

8.3.1 Exhaust Ducts

The conductivity of the exhaust duct has a direct impact on the effective pumping speed at the divertor/limiter or beam dump plates for a given pumping speed of the vacuum pumps. In addition to the existing space restrictions within the machine as mentioned under 8.2.4.1/.2, measures for controlled precipitation of the volatile sputtering material (see 8.1.2.3) could further impair the duct conductivity.

The aim is therefore a high overall conductivity value, to keep the pump requirements in acceptable limits.

Because of the existing uncertainties in these fields, the duct conductivity values of a DEMO are highly uncertain at present, and consequently, also the required pumping speed of the vacuum pumps. The practical experience in the operation of NET-P and TASKA will give a first reliable proof in this field.

8.3.2/.4 Vacuum Valves and Pumps

The main R and D tasks are connected with the extrapolation in size. An appropriate non nuclear test facility for simulation tests will be necessary.

References

/1/ W. Köppendörfer et.al, Status and Progress of ASDEX-Upgrade, Proc. 13. SOFT, Varese Sept. 1984, paper 0.08

/2/ A Demonstration TOKAMAK Power Plant Study (DEMO) Report ANL/FPP-82-1, September 1982

/3/ O. Gröbner, et.al.: "Studies of photon induced gas desorption using synchrotron radiation", Vacuum 33 (1983) 7, p. 397-406

/4/ R.S. Vaughan Watkins et.al.: "An appraisal of glow discharge treatment of copper surfaces by the techniques of electron stimulated desorption", Vacuum 28 (1978) 10/11, p. 459-465

/5/ Magnetic Fusion Energy Plasma Interactive and High Heat Flux Components, Volume I: Technical Assessment of the Critical Issues and Problem Areas in the Plasma Materials Interaction Field, Report PPG-765, January 1984

/6/ S.L. Allen, et.al.: "Plasma-Surface Interactions in Large Tandem Mirror Devices--MFTF-B", International Conf. on Plasma Surface Interactions in Fusion Devices, Nagoya, Japan, May 7, 1984

/7/ J.N. Brooks, et.al.: "An Impurity Control Test Facility (ICTF) for the Study of Fusion Reactor Plasma/Edge Materials Interactions", ANL/FPP/TM-188, May 1984

/8/ INTOR, International Tokamak Reactor, Phase One, International Atomic Energy Agency, Vienna, 1982

/9/ W.M. Stacey, Jr. et.al.: "Future Technology Requirements for Magnetic Fusion - an Evaluation Based on Conceptual Design Studies", Nuclear Technology/Fusion, Vol. 5, May 1984 p. 266-290

9. Magnets and Transient Electromagnetics

The magnet system of a fusion reactor represents one of the largest and most expensive components in all kinds of magnetic plasma confinement systems. Its reliable and safe performance is one of the most crucial issues in fusion technology.

Most present-day plasma experiments, even the big tokamaks, utilize normalconducting magnets only. This can be justified by the short operation periods. On the other hand, the use of superconducting magnet systems will be necessary for the DEMO and later fusion reactors due to economic reasons. But also for the next generation of large scale experimental devices, in most designs superconducting magnet systems are already foreseen to provide high availability and long burn times at reasonable operating costs. Even two medium size tokamak experiments under construction at the present time, Tore Supra (CEA, France) /1/ and T15 (Kurchatov, USSR) /2/, will have superconducting toroidal field coils.

In the area of tandem mirrors, a complete implementation of superconducting magnets is taking place by construction of the large MFTF-B experiment /3/ at LLNL with a full set of superconducting magnets of reactor relevant size. However, the field level in solenoidal coils must be further increased in future devices.

These facts have been recognized rather early in fusion technology program plans and superconducting magnet development for fusion has been proceeding with remarkable progress worldwide for several years. The progress achieved thus far leads to increasing confidence that this technology can be ready in time at an appropriately high level to build additional large fusion experiments with superconducting magnet systems and to extrapolate from these experiences to the DEMO case. The most extensive new developments are associated with the need for pulsed magnets in tokamaks.

In addition to large superconducting magnet systems, there remains the need for normalconducting windings ("loops") for active and passive plasma stabilization, even in a NET or DEMO tokamak. For such windings, whether near the plasma or in the outer blanket region, specific development questions have to be addressed. In tandem mirrors, supplementary normalconducting windings are required for reaching very high field levels such as those needed in choke coils /4, 5, 6/.

Table 9.1 contains in a very compact form the items which are needed for a tokamak DEMO reactor and gives a supplement of additional needs for a conceivable mirror DEMO reactor. An x in the "TASKA-class" or "NET-P-class" column indicates that a device of the TASKA or NET-P class meets the specific need for a tokamak DEMO reactor. Both classes are assumed to have superconducting magnet systems with supplementary normalconducting windings. The last column stands for non-fusion devices (test stands, simulation facilities). An x or a name of a device in parentheses (x) indicates a limited use of the device.

Comments and explanations to the different items of Table 9.1 are given in the following subchapters.

Table 9.1: Development requirements for magnets needed in a DEMO reactor

Item	Needs for a tokamak DEMO reactor	Facilities to meet needs		
		NET-P class	TASKA class	Test stands, simulation devices
9.1	Large Size High Field Magnet Design and Construction			
*	1. Toroidal field coils (~12 T, ~10 m Ø, transient pulsed load)	(x)	(x)	(LCT)/7/, (Tore Supra), (T15)
*	2. Poloidal field coils /8/ (~8 T, ~ 20 m Ø, ~1 T/s)	x		(Tore Supra), (ASDEX-U) /9/
	3. On site and/or in situ fabrication	(x)	(x)	(x)
	4. Appropriate cooling modes and general design criteria	x	x	x
9.2	Overall Engineering Constraints in Reactor Environment	x	x	(Superconducting plasma experiments)
9.3	Operation of Integrated Magnet System			
	1. Automatic cryogenic system	x	x	Large accelerators /10/ LCT, superconducting plasma experiments
	2. Radiation influence on cryogenic liquids (H, C)	(x)	x	Fission reactors /11/
	3. Structural materials			
	1) Fabricability (welding ..)	x	x	x
	2) Fatigue (>10^4 cycles)	x		x
	4. Coil protection (~10^{10} J)	x	x	x

* Special need for tokamak only.

Table 9.1: - cont'd -

Item	Needs for a tokamak DEMO reactor	Facilities to meet needs: NET-P class	TASKA class	Test stands, simulation devices
9.4	Irradiation in Superconducting Magnets		x	(Fission reactors), simulation
	1. Data base for conductor components			
	Superconductor		x	)
	Stabilizer		x	) (Fission reactor
	Insulator (organic, ceramics)		x	) tests)
	Structure		x	)
	Solder		x	)
	2. Entire magnet (annealing, degassing, degradation, warm-up and irradiation cycles)		x	(Large fission reactor tests)
	3. Nuclear heating limits (cooling, economics)	(x)	(x)	(Large fission reactor tests)
9.5	Normalconducting Coils Close to the Plasma			
	1. High heat load cooling	(x)	x	)
	2. Transmutations in conductor		x	) (Fission reactor tests)
	3. Lifetime of ceramic insulation		x	)
	4. Radiolysis, erosion and corrosion of conductor	(x)	x	(Fission reactor tests)
9.6	Transient Electromagnetic Effects			
	1. Start-up effects	x	(x)	x
*	2. Vertical plasma stabilization effects	x		Tokamak experiments
*	3. Disruption effects	x		Tokamak experiments
*	4. Influence of time-varying magnetic fields on the magnet itself and other reactor components	x		x

* Special need for tokamak only

Table 9.1: - cont' -

Item	Additional Needs for a Mirror DEMO Reactor	Facilities to meet needs		
		NET-P class	TASKA class	Test stands, simulation devices
9.7	1. Choke coils (SC part) (~15T, ~3m inner Ø, steady state)		x	MFTF-B Upgrade
	2. Normalconducting choke coil inserts (~ 6 - 10 T, ~ 0.5 m inner bore)		x	MFTF-B, Large fission reactor tests
	3. End coils (8-9 T, 8m x 4m characteristic dimensions of C-shaped coils)		x	MFTF-B, MFTF-B Upgrade

Comments to Table 9.1

9.1 Large Size High Field Magnet Design and Construction

9.1.1 Toroidal Field Coils

Steady state superconducting magnets operating in a maximum field up to about 12 T are required for tokamak reactors. The bore of the D-shaped toroidal field coil is about 10 m x 7 m and the stored energy per coil is about 3-4 GJ; the total stored energy in the TF system is in the range of 50 GJ. Discharge voltages of up to 10 kV are considered to be necessary in the case of a fault. Transient a.c. field components in the order of 1-2 T and $\dot{B}$ of 0.5-1 T/s are expected and require specific conductor designs, efficient cooling and nonmetallic interfaces in the torus structure. Stresses up to about 500 MPa static in plane and about 200 MPa cyclic out of plane require a careful mechanical design. Nuclear gamma and neutron radiation of about 10^7-10^8 Gy (1 Gy = 10^2 rad) require the selection of suitable materials and sufficient cooling for heat removal (peak loads of 1 kW/m³). Coils of the NET-P-class devices have similar field and transient pulsed load levels as DEMO coils, but they are smaller in size and experience less radiation load; therefore the NET-P class is considered to be of limited use. Coils in TASKA class fusion devices would be of similar size as DEMO TF-coils. Due to their different geometry and the lack of field pulsing, their use for specific tasks in tokamak magnet design would be limited.

The largest superconducting TF coils to date are the six coils built for the LCTF /7/. The inner bore is 3.5 m x 2.5 m, the maximum field is 8 T and the stored energy per coil is about 100 MJ. Two of the coils have already been tested thus far (November 1984) and performed well in single-coil tests /12, 13/. Experimental data of the mechanical behavior in LCT coil tests will be compared with the result of finite element (FE) calculations. This will lead to an improved utilization of FE-codes for superconducting magnet design.

Coils using NbTi superconductors and 4 K cooling are limited to about 8 T. Cooling at 1.8 K extends the maximum field to about 11 T, but Nb_3Sn, cooled at 4 K, is capable of at least 12 T and offers a wider stability margin.

Nb_3Sn conductors are under development for that high field region. Manufacturing methods and automatic quality control methods have to be studied. The conductor must withstand the nuclear radiation and must tolerate some pulsed load. No complete data base for structural materials exists at low temperatures and cycle numbers greater than 10^4. For all these tasks, experiences with a TASKA class magnet system would be extremely valuable, but not complete.

9.1.2 Poloidal Field Coils

Maximum magnetic fields in PF coils are in the range of 8 T which allows the use of NbTi at 4 K. The development of high current, low a.c. loss conductors is required. The conductor must operate under cyclic variation of the current and the magnetic forces. This requires the knowledge of the fatigue behaviour of the structural materials at low temperatures. There is no complete data base yet. High current, in the range of 50 kA, is required in order to limit the coil voltage to a reasonable value while charging and discharging the stored energy of several GJ. Power leads for this current level and a voltage level of several ten kV have to be developed.

On the one hand, the pulsed mechanical loads favor the incorporation of steel or other support materials into a rigid conductor design; on the other, high voltage insulation must be provided between the windings and layers to withstand the inductive voltage during a pulse. Further cooling channels in close proximity to the conductor surface must be provided to cope with the heating from the a.c. losses, and the metal parts of the conductor must be subdivided and oriented to reduce the eddy current losses to a tolerable level. Low loss helium cryostats also have to be developed, utilizing either organic materials or insulating strips in the cryostat wall.

No pulsed coil of the required size ($\sim$ 20 m diameter) has been built thus far. Small coils (typically 1 m - 3 m diameter, stored energy 1 MJ - 30 MJ, $\sim$ 5 T/s) have been built and tested. Development programs are underway for intermediate-scale PF-coils (8-10 m diameter).

If high field ($\geq$ 8 T) superconducting coils are required, e.g. in the tokamak OH-system, the development of Nb_3Sn superconductors for pulsed coils is necessary.

As mirrors, TASKA-class devices are d.c. machines and will not contribute to the development of poloidal field coil systems, even though it is possible in principle to design and operate central cell coils in a pulsed mode. More appropriate test beds, providing realistic conditions, are tokamaks under construction such as TORE-SUPRA or ASDEX-Upgrade. With radiation damage being a minor concern for PF-coils, NET-P will demonstrate their availability for DEMO.

9.1.3 On Site and/or in Situ Fabrication

TF-coils of 7 m x 10 m characteristic dimensions and solenoids of about 10 m in diameter for the central cell of tandem mirror reactors are not amenable to transportation by conventional means. Thus on site fabrication should be investigated. PF-coils of 20 m diameter may not even be transportable from a fabrication location on site to the reactor building due to the dimensions and weight. For this reason, an in situ fabrication process should be developed.

Due to the smaller size of NET-P coils, the NET-P class is considered to be of limited use. A number of coils in TASKA-class devices will most likely be fabricated on site and will thus provide some relevant information for tokamaks as well.

9.1.4 Appropriate Cooling Modes

Appropriate cooling modes for superconducting pulsed or steady state coils have to be investigated. Two cooling modes (bath cooling and force flow cooling) will be tested in LCTF for large TF-coils. Both cooling modes have already been tested in single-coil tests.

The cool-down and warm-up must be performed in a reasonable time. This requires large and reliable cryogenic systems with sufficient adjustable refrigeration power over the whole temperature range to handle the huge masses to be cooled down or warmed up. The design of cooling paths in the cold structures must be optimized to limit thermal stresses.

These problems are of equal importance for tokamaks and mirrors.

9.2 Overall Engineering Constraints in Reactor Environment

The magnet geometry and the support structure have to be compatible with the reactor geometry. The number and size of the magnets should be minimized to provide space for heating and exhaust and for other reactor components which will be changed occasionally such as first wall components and blankets. The choice of the support structure for the magnetic forces will strongly influence the maintenance scheme and general reactor engineering concepts.

9.3 Operation of Integrated Magnet System

9.3.1 Automatic Cryogenic Systems

Automatic cryogenic systems of 20 kW - 30 kW are available. Higher power levels are easily attainable without fundamental development work, but their long time reliable operation must be demonstrated. Operation of a TASKA-class device will give the essential information for other confinement systems as well.

9.3.2 Radiation Influence on Cryogenic Liquids

If the neutron and radiation flux and energy spectrum is known, e.g. from operation of a TASKA-class device with appropriate blanket and shielding, it is easy to calculate nuclear reaction products in the cryogenic liquids helium and nitrogen. Experiments in fission reactors can support these calculations. Simulation loops can demonstrate the reliable operation of a purification system.

9.3.3 Structural Materials

A reliable continuous operation for the whole lifetime of a fusion device implies to take into account endurance and fatigue limits and the accumulated radiation damage effects for all materials. For tokamaks the fatigue properties for more than 10^4 cycles at 4 K have to be known, mainly for the structural materials used. The data base is available for the 300 SS series, but does not exist for many other materials.

Mechanical data, especially fracture data, of metallic and nonmetallic materials at 4 K have to be known for the structural materials. Such data can easily be accumulated in test stands. The influence of nuclear radiation on these properties must be assessed, too, especially for nonmetallics. Fission reactor test stands are suitable for that purpose, but they will not produce the full information due to the different spectra. A TASKA-class magnet operation can provide final integral proofs.

9.3.4 Coil Protection

The high stored energy of the magnets must be safely discharged in the event of a magnet failure. Reliable fault detectors and energy discharge systems must be designed and demonstrated in large superconducting facilities. Detailed event trees leading to accidents must be evaluated. Possible consequences of the accidents in the overall system must be evaluated and minimized.

It is important to distinguish between "abnormal operating conditions", typically represented by a quench which cause only a temporary shutdown of the magnet system without damage and "accident situations" with the danger of subsequent damage.

Specific to a fusion magnet system is the strong coupling of many coils with significant changes of the internal force distribution in case of single coil failures. This has to be taken into account carefully in the design of the mechanical structure and of the safety discharge system.

Due to the fact that accident situations may be associated with the generation of high power arcs driven by the magnetically stored energy, such phenomena need to be studied in more detail. Most of these issues can be addressed in specific experiments such as LCT.

9.4 Irradiation in Superconducting Magnets

9.4.1 Data Base for Conductor Components

The data base for superconductors NbTi and Nb_3Sn and the stabilizer materials Cu and Al is very broad for low temperature irradiation and fields up to 6 T. For fields up to 12 T only a few data exist. Low temperature irradiation data for organic insulation materials are only satisfactory for a few materials (e.g. G-10CR, G-11CR, and PG-10CR). Data for ceramics, structural and soldering materials are very spare. Sufficient irradiation data can easily be obtained by fission reactor irradiation of all the conductor components, because the energy spectrum in fission reactors is similar but not identical to the fusion reactor spectrum at the magnets, expecially the ratio of gamma to neutron radiation is different.

9.4.2 Entire Magnet

No data of magnet response to irradiation under full operation condition (cooling, current, irradiation, warm-up, annealing, and cyclic warm-up and irradiation) are available. Fission reactors can be used for that purpose.

9.4.3 Nuclear Heating Limits

A trade-off study between nuclear heating limits and economic considerations (costs for thicker shield material and larger magnets) must be performed in order to define an optimum value. This study can be done analytically rather than experimentally.

9.5 Normalconducting Coils Close to the Plasma

9.5.1 High Heat Load Cooling

Near-plasma normalconducting coils have to be cooled very effectively, because in addition to the Ohmic heat (5 - 10 W/cm^3) a high level of radiation heat (~ 5 W/cm^3) has to be carried away by the coolant water. This can be done in large fission tests.

9.5.2 Transmutations in Conductor

Due to neutron irradiation in copper, transmutation products are produced (mainly the neighbour elements nickel and zinc). These products act as impurities and cause an enhancement of the copper resistance leading to more power consumption. Fission reactor irradiations should clarify this behaviour in order to define the time when the insert coil must be exchanged.

9.5.3 Lifetime of Ceramic Insulation

No organic insulation materials can be used in the insert coils due to the high radiation level. Ceramic insulation materials are foreseen, but the irradiation data base for these materials in the temperature range of room temperature to about 150 ^{o}C is very insufficient. So, irradiation tests are required to define the lifetime of the ceramic insulation.

9.5.4 Radiolysis, Erosion and Corrosion of Conductor

The effect of the radiation on the cooling water (radiolysis and subsequent corrosion) has to be investigated. Fission tests are of limited use in this case, because the energy spectrum in a fusion device near the plasma is much harder than in a fission reactor.

9.6 Transient Electromagnetic Effects

9.6.1 Start-up Effects

Plasma start-up in a tokamak is associated with plasma current generation and the pulsing of equilibrium field coils and possibly divertor coils. Parametric studies must be performed on several start-up scenarios (variation of time) and different structural arrangements (variation of resistance). The resulting variation of the penetration time of the toroidal electrical field (or voltage penetration time) and the damping of the vertical plasma equilibrium field have to be calculated and assessed. The effect on the superconducting coils has to be investigated.

In a tandem mirror, transient electromagnetic effects upon plasma start-up are limited to diamagnetic plasma currents, albeit at a relatively high β-level ($\beta \sim 50$ %). The amplitudes of transient fields at the superconducting coils and the structure are markedly smaller than in the tokamak.

9.6.2 Vertical Plasma Stabilization Effects

For vertically elongated plasmas, additional stabilization magnets are necessary. Simple plasma models should be used for quick calculations and preliminary trend investigations, e.g. of pulsed loads due to stabilization currents. Afterwards complex and complete plasma models have to be developed for detailed investigations in which the geometry (distance of plasma to coils and/or structure) and the electromagnetic parameters (resistance and self-inductance of coil and structure) are varied. Also the interaction of passive and active stabilization elements has to be analyzed in order to minimize the power level in these coils and the cooling power in the TF-coil system.

9.6.3 Disruption Effects

Similar to 9.6.2 parametric studies for disruption effects on the surrounding structure and magnets have to be performed using simple plasma models (filamentary current loop) in an early stage of the investigations, and complex plasma models (distributed currents) later on. The vaporization of material, induced voltages and eddy currents, and also forces in the fusion device have to be analyzed. The goal of those studies and analyses in a NET-P class device is the understanding of the nature of disruptions and to develop means in order to prevent disruptions in a DEMO. Therefore the magnets in a NET-P class device have to withstand hard disruptions, while the requirements to DEMO magnets are mitigated.

9.6.4 Influence of Time-varying Magnetic Fields

The influence of time-varying magnetic fields on other reactor components must be calculated to be taken into account during the design phase of a fusion reactor. 3D-calculations are required. A topic of increasing emphasis is that of eddy current calculations. It is specific to tokamaks, where transient electromagnetic effects play an important role and need to be considered in the design in a detailed manner. The computational principles and codes are not well developed until now.

Additional Needs for a Mirror DEMO Reactor

9.7.1 Choke Coils (SC part)

Steady state superconducting solenoids are required for tandem mirror reactors. The maximum field should be as high as possible, at least 15 T. The inner diameter is of the order of 3 m, the stored energy is of the order of 3-5 GJ.

Large bubble chamber magnets have been built (with 800 MJ stored energy and about 4 m inner diameter) and successfully operated for several years, but they used NbTi conductors. For the high field level cited above Nb_3Sn conductors are necessary. These conductors must operate under very high magnetic forces.

9.7.2 Normalconducting Choke Coil Inserts

Normalconducting water-cooled copper magnets are designed for choke coil inserts in a tandem mirror reactor. These inserts have a bore of about half a meter and generate fields in the order of about 8T to raise the magnetic mirror field above 20 T at the ends of the central cell. These near-plasma coils have to withstand high irradiation loads, but due to their position not the full load as the near-plasma components in the central cell or in a tokamak.

9.7.3 End Coils

C-shaped coils for a tandem mirror reactor of nearly the required dimensions as cited in Table 9.1 have been built for the MFTF-B facility and were successfully tested. Therefore, such coils are state of the art and not much development work is needed.

References to chapter 9

/1/ TORE SUPRA, Basic Design Tokamak System, Report EUR-CEA-FC-1068, Oct. 1980.

/2/ B. G. Karasev et al., IAEA Techn. Committee Meeting on Large Tokamaks, 4th., Tokyo, Japan, April 1980.

/3/ K. I. Thomassen, V. N. Karpenko, Tandem Mirror Fusion Test Facility - MFTF-B Conceptual Design Report and Project Proposal, LLNL, Report LLL-Prop-163, Jan. 23, 1980.

/4/ S. Wang et al., The Axicell MFTF-B Superconducting Magnet System. 9th Int. Cryogenic Eng. Conf., Kobe, Japan, 1982.

/5/ TASKA-Tandem Spiegelmaschine Karlsruhe, KfK 3311, 1982.

/6/ B. G. Logan, The Mirror Advanced Reactor Study (MARS), Nucl. Techn./-Fusion, 4 (1983) 2/2, p. 563

/7/ P. N. Haubenreich, Superconducting Magnets for Toroidal Fusion Reactors, IEEE Trans. on Magn., Vol. MAG-17, No. 1, Jan. 1981, p. 31ff.

/8/ INTOR - Phase IIA, IAEA, Vienna, 1983.

/9/ W. Köppendörfer and ASDEX Upgrade Design Team and Tokamak Theory Group, ASDEX Upgrade - a tokamak experiment with a reactor-compatible poloidal divertor. Proc. 12th SOFT, Jülich, Comm. EC, Pergamon Press, Oxford 1982, 1, 187-192.

/10/ J. Theilacker et al., Commissioning of the Tevatron Satellite Refrigeration System. Adv. in Cryogenic Engineering, Vol. 29, 1983, pp. 429-437.

/11/ see e.g.:
K. Böning et al., Journal of Nuclear Materials 108 and 109 (1982), pp. 10-20,
and
L. R. Greenwood, ibid., pp. 21-27.

/12/ S. Shimamoto et al., Domestic Test Result of the Japanese LCT Coil. IEEE Transaction on Magnetics, Vol. MAG 19, No. 3, May 1983, p. 851ff.

/13/ H. Krauth, Test of the Superconducting Toroidal Field Coil for the Large Coil Task (LCT) in TOSKA, KfK Nachrichten, Jahrgang 16, 3/84, p. 123ff.

10. Instrumentation and Control

10.0 Introduction and Summary

This section deals with the question of what kinds of instrumentation and control are required for a tokamak DEMO; in particular, how are its needs different from currently available techniques, and where and how should these development needs be addressed ? Instruments and controls are included together because handling of the data is one of the larger tasks of a control system, and to a good extent, much of the information deriving from instruments is used in control, especially in a reactor-like machine.

Just as the present generation of fusion machines (JET, MFTF-B, TFTR) represents a qualitative change from the previous generation in size and hostility of environment, a DEMO represents a quantum change: whereas up to now almost all machines have been pulsed, DEMO and TASKA-class machines will run continuously (or nearly so). Moreover, the presence of tritium and the continuous burn create a different environment because of the large volume of tritium handling, because of the much higher radiation level and total dose, and the activation thereby induced. It will be presumed here that a NET-P machine will still be pulsed, albeit for long pulses of up to 500 seconds.

The mode of operation of the NET-P and TASKA-class machines is seen as quite different, and therefore strongly affecting the Instrumentation and Control Systems. The NET-P device would be designed to be used in a more flexible manner, to search out the various corners of its parameters' phase space, and to be changed relatively frequently in order to pursue development and characterization of confinement concepts, whereas the TASKA-class device would run as a facility, with long periods at more or less constant conditions, in order to provide high total exposures and long operation times for engineering design purposes. Thus, it is more like a reactor, less flexible and subject to change, than a physics machine.

Consequently, it is seen that both DEMO and a TASKA-class machine would run DC, have less complicated instrumentation, but more robust, more easily (or less frequently) maintained in the ractor-like environment of higher exposures and long periods of constant running between relatively short maintenance breaks.

Table 10-1 listes the five major areas of needs in instrumentation and control, and summarizes how and where these development needs can be met. Each of the individual items is discussed in more detail in the following sections.

The conclusion of our study here is that, for the purpose of instrumentation and control, DEMO and a TASKA-class machine are very similar, that nearly all requirements of DEMO could be designed, developed and tested in a TASKA-class machine. The exceptions are two: First, the identification of the mechanisms underlying disruption in tokamaks, and their control, will have to be done on tokamaks, including any final demonstration on NET-P. However, because of the similarity (often identity) of hardware and certain plasma parameters such as density, temperature, and characteristic rise and decay times of plasma oscillations, the tools of such control can be developed and tested on a TASKA-class machine. Which signals to monitor, and which input parameters to change, in what way, will have to be determined by research on relevant tokamaks, the present generation as well as NET-P.

The second exception is similar, namely the overall control of the reactor, e.g., how to take the power output from one level to another. Again, development and test on a TASKA-class machine will provide the tools, but not the model used to turn measured signals into control element commands.

All other elements are compatible with, and most required by, the operation of a TASKA-class machine. Characteristic times are similar, plasma parameters are similar, sub-systems are similar (most are conceptually alike) and an item-by-item examination of the instrumentation list indicates that all could be tested on a TASKA-class machine, both for physical performance, and environmental survival. The instruments are, for the most part, the same, so that evaluation of component and material performance in the high radiation level environment and in magnetic fields is the same for DEMO instruments as for TASKA-class instruments.

TABLE 10-1: Instrumentation and Control Needs for a DEMO Reactor

Item	Description	Investigated in NET-P	Investigated in TASKA-class	Other Facilities
10.1	System Control:			
	.1 With real-time feedback	x	(x)+	Other tokamaks.
	.2 Open-loop	x		Other tokamaks.
10.2	Subsystem Control:			
	.1 Heating	(x)	x	
	.2 Fueling	x	x	
	.3 Impurities Control	x	(x)+	Doublet III ASDEX-Up
	.4 Vacuum	(x)	x	
	.5 Cryogenics	x	x	
10.3	System Safety:			
	.1 Impact of subsystems on each other	x	x	
	.2 Specific tokamak behavior,e.g. disruptions.	x	-	Other tokamaks.
10.4	Data Handling:			
	.1 Acquisition	(x)	x	) Elmo Bumpy
	.2 Processing	(x)	x	) Torus
	.3 Display	(x)	x	)
	.4 Archiving	(x)	x	
10.5	Instrumentation:			
	.1 Instruments	x	x	(x)
	.2 Sensors	(x)	x	(x)
	.3 Components	(x)	x	(x)
	.4 Materials	(x)	x	(x)

+ testing of control elements and methodology, not of specific control functions

10.1 System Control

The main purpose of a reactor is to run at a specified power output. In order to achieve that power level in a reasonable optimal manner, certain densities, temperatures, and volumes must be obtained by means of setting various subsystem levels, such as fuel feed, neutral beam injection current and energy, etc. This control may operate in an open-loop fashion, i.e., where input levels are set, and later changed if so desired, or via loop-closing feedback control, wherein a measurement of the output is made, a model of reactor operation is used to calculate new desired input parameters, and commands are issued to set the subsystems at this new set of parameters. Since the response of the system to variations of any of the controllable input parameters is characteristic, certainly, of tokamaks, and often characteristic of the specific design, the model used for control cannot be tested on a tandem mirror machine, but must be developed on a machine much more similar to a DEMO, such as a NET-P class machine. However, the process of detecting fluctuations of some amplitude and of some frequency content, and using the resulting signal to initiate a change in some input parameter, such as neutral beams, or RF power, and executing that change in 10-50 milliseconds, can be tested on TASKA-class, for all the subsystems relevant to DEMO.

An example is control of disruptions, a characteristic of tokamaks, wherein, after an increase in fluctuations of density and X-ray spectra, a large amount of plasma is suddenly lost. In a large tokamak both the sudden heat loading on the wall and the forces associated with the sudden reduction in current are potentially damaging. It is presumed here that research on predecessor machines will produce sufficient understanding of disruptions, so that they may be identified early enough to control them by modifying inputs before they get out of hand. Given transport, rotation, and drift times in large toroidal machines, it is generally assumed that a 10 millisecond response time is quick enough. If it is not, then DEMO control is much more difficult, because control of so complex a device in a shorter time, while technically possible, is much more expensive.

An actual demonstration of the effectiveness of the method of control of disruption, and operation in general would be appropriate tasks for NET-P; demonstration of the ability to control all plasma and auxiliary subsystems, within required control times, with a reactor-like control system, in a reactor environment, would be appropriate for a TASKA-class machine.

10.2 Subsystem Control

This is required for startup and maintenance of operating level. Control may be in the form of determining and realizing a controllable input parameter, such as a valve setting or a voltage -- which could be time dependent -- or may involve the measurement of some parameter of the subsystem, e.g., beam current, or reflected microwave power, followed by simple processing, which would produce a new set of values for the controllable input parameters. The time from measurement to execution of the instruction command would be $\geq$ 10 milliseconds. This is both within the easily obtainable state-of-the-art and meets the requirements of tokamak operation.

The same methods of control would be used for all subsystems; plasma systems (heating, fueling, and impurities control) as well as auxiliary (vacuum, cryogenics, etc.). Some subsystems, such as main magnets and cryogenics require only much slower control. As has been seen in earlier sections, the subsystem for a TASKA-class facility are very similar to those of a DEMO, and so the control techniques and hardware could be developed and fully tested for reactor operations conditions on a TASKA-class machine.

10.3 System Safety

For the most part, system safety (interlocks, hazards to personnel and environment, etc.) are usually handled by a separate system, independent from the control and data handling system. However, it is unavoidable that some indication of the status of the system safety will lie within the control system, and therefore must be considered here. All of the subsystems under 10.2 are included. Each of these will have, of course, its own self-protective features in its own design, but the Control System must assume the task of protecting subsystem A from conditions in subsystem B, which while not harmful to B, could place A in jeopardy. Such protective controls can be on a NET-P machine but, because it is a pulsed machine and its integrated running time is much less, it cannot be as thorough a testbed as a TASKA-class machine, whose operation and environment have all the features of a DEMO reactor.

10.4 Data Handling

This includes acquisition, processing, display, and archiving. The requirements are quite different for DEMO from prior practice on previous experimental machines, which have been pulsed and have collected and archived enormous amounts of detail on each shot. Even JET, TFTR, and MFTF-B, while pulsed for relatively long times (5 - 30 seconds), have had their data systems based on taking data on a shot-by-shot basis, displaying some results between shots, and archiving most, if not all, data on a shot-by-shot basis.

Like a future power reactor, a DEMO will run DC, or nearly so, and will have changes occuring in real time (rather than between pulses) and so the model for the data system is the chart recorder, rather than the oscilloscope. As a prototype, a DEMO will probably have many more sensors than a power reactor, for the purpose of gathering engineering data, and more operating data will be archived, so that details of the conditions of exposure will be available after-the-fact, when analyzing the performance and failure mechanisms of various test subsystems. In a reactor one need only record a few data, such as the net output power and time of operation of subsystems; a DEMO will need a large number of sensors providing temperatures, densities, pressures, and spectral information, as a function of time, as information logged continuously over a long period of time.

Experience on both toroidal and tandem mirror machines demonstrates conclusively that the task of off-line analysis should take place using a computer system separate from the one which acquires the data and stores them. Thus the task of a DEMO's control system is to archive the data in a place and format for use by another system (or other systems).

As noted above, some data will be used for online tasks (Operational safety, startup, maintenance, and operation) and must be displayed in a useable, convenient way, appropriate for DC operation. Because a TASKA-class machine is also a DC machine, it will be the appropriate place to develop and test the data system for a DEMO.

Typically, on pulsed machines this is done in a combination of intershot processing and post-run (i.e., nights and weekends) processing, which is not consistent with DC operation, and so a different methodology and a different systems are required, which could be developed and tested in a TASKA-class facility.

Finally, while data acquisition for a pulsed machine consists typically of real-time analog-to-digital conversion, followed by storage in local CAMAC memory, and read-out thereafter for processing and archiving, such a process is poorly suited, if not impossible for DC operation. Thus, the entire data system must be designed and developed from the beginning for DC operation, an appropriate task for a TASKA-class facility. A rough estimate of the number of signals and data rates indicates that the volume of data to be archived is similar to JET and TFTR: 300-800 Mbytes per day. Thus the power of the control and data system must be of the same order of magnitude as present-day fusion machines, but the software architecture will be quite different.

10.5 Instrumentation

Table 10-2 is a compilation of expected doses and dose rates for DEMO and a TASKA-class machine (NET-TP would have dose rates similar to TASKA, but much smaller doses, due to the lower duty cycle). The figures for the DEMO are from the STARFIRE report, which used an average neutron wall loading of 3.6 MW/m^2, comparable to DEMO's 2-4 MW/m^2. Position A is the first wall, and separate numbers are given for doses and dose rates (Rads-Si) for neutrons and gammas. Position B is the corresponding number at ~2.5 m

TABLE 10-2: Radiation Levels at Relevant Positions

	DEMO (=STARFIRE)		TASKA-class	
DOSE RATE (Rads-Si/s)				
Position	Neutrons	Gammas	Neutrons	Gammas
A (first wall)	4.4×10^{5}	2.2×10^{5}	1.7×10^{5}	8.8×10^{4}
B (2.5 m from centre)	1.2×10^{3}	6×10^{2}	4.9×10^{2}	2.5×10^{2}
C (outside shield)	2.9×10^{-5}	1.8×10^{-4}	1.3×10^{2}	1.5×10^{3}
LIFETIME DOSE (Rads-Si)				
A	1.3×10^{14}	6.7×10^{13}	3.7×10^{13}	1.9×10^{13}
B	3.8×10^{11}	1.9×10^{11}	1.1×10^{11}	5.5×10^{10}
C	9.3×10^{3}	5.8×10^{4}	2.9×10^{10}	3.3×10^{11}

from machine center line, for a collimation restricting the view to a 10 cm diameter tube through the plasma. Position C is outside the DEMO shield, and the TASKA-class blanket, respectively. While a TASKA-class machine may not have the full shield assumed in STARFIRE, some additional shielding outside the blanket is certain.

10.5.1/.2 Instruments and Sensors

Because the plasma temperatures and densities are similar for DEMO and a TASKA-class machine, and because access is also similar, a detailed inspection of the exhausitve list of possible INTOR instruments (Tables 10-3, 10-4, 10-5) shows that the physical measurement aspect of every one could be tested and demonstrated on a TASKA-class machine. On the other hand, although the total dose on a NET-P machine is much lower, the dose rate is high enough to show whether there is a noise problem. It must be realized that even if the radiation level is below damage levels, it has been found that background neutrons and gammas can seriously interfere with the signals.[1]

10.5.3/4. Components and Materials

J.F. Baur et al. have compiled an excellent and exhaustive study of radiation effects on diagnostics for fusion reactors.[2] Table 10-6 is a very brief summary of radiation tolerance of electronic components and some relevant materials. Comparison of Table 10-2 and Table 10-6 shows that no known, available electronics component will survive long in the first wall environment. Simple detectors, such as bolometers, Langmuir probes, and diamagnetic loops, which are made of metals and insulators, can be placed there if they are made from selected materials. Depending on specific installation, they may also need active cooling. Components which must see the plasma directly, such as windows, mirrors, photodiodes, etc., will have to see a collimated view, and be placed even farther from the plasma than assumed in the calculations for Table 10-2.

Finally, components which are out of a line-of-sight, such as optical fibres, and semiconductor devices, will be satisfactory in the well-shielded outer regions of DEMO and TASKA-class facilities.

TABLE 10-3: INTOR Diagnostics for Real Time Control.
(Courtesy of K.M. Young, Princeton University)

Information Needed	Diagnostic Instrument
Plasma current	Rogowski coil
Plasma position, shape	Position sensors
Electron temperature	ECE radiometer
Electron density	Interferometer
Neutron flux	Neutron detectors
Radiated power	Bolometers
Hot wall definition	IR camera
Ion densities	Charge exchange
Impurity identification and flux	Survey, UV, visible spectrometer
Helium ion density	Monochromator
Disruption precursor	Soft X-ray array/Neutron det. array
Current density profile	Faraday rotation
Fast pressure gauges	Shielded ion gauges
Residual gas concentrations	Residual gas analyzers
Runaway electron flux	X-ray detectors
Torus chamber inspection	Video cameras

TABLE 10-4: INTOR: Diagnostics for Plasma Evaluation and Optimization. (Courtesy of K.M. Young, Princeton University)

Need	Diagnostics for Baseline Performance
I_p plasma current	Rogowski loops (sum of B_θ loops)
V_p loop voltage	Set of toroidal loops
a, K plasma shape, position	Array of B loops, saddle coils
n_e electron density	1 mm microwave horizontal system
	FIR interferometer array
	1 mm divertor system
T_e electron temperature	$2\omega_{ce}$ fast scanning radiometer
	Vertical multi-point, multi-time Thomson scattering
T_i ion temperature	Neutron flux detectors
	Charge exchange analyzers
	X-ray crystal spectrometer
n_D, n_T ion densities	Charge-exchange analyzers
n_I impurity concentrations	Survey UV spectrometers
	Visible bremsstrahlung array
	X-ray crystal spectrometer
n_{He} helium ash	Charge-exchange spectrometer
n_α fast-alpha density	Technique to be determined
P_r radiated power	Bolometers--horizontal and vertical arrays
Plasma magnetic pressure	Diamagnetic loops
J_p Current density profile	Faraday rotation in FIR interferometer
Runaway electrons	X-ray scintillators in forward cone
Wave activity, sawteeth	Soft X-ray array/ Neutron wave array
Ion wave absorption	Microwave scattering system
Wall temperatures	Infrared cameras with wide-angle view
	Divertor cameras
Neutron production	Neutron flux detectors
Charged fusion product particles	Technique to be determined
Base pressure, background gas	Vacuum gauging/Residual gas analyzer
Torus chamber inspection	Video camera

TABLE 10-5: Instruments for a TASKA-class Machine.

A. OPERATION

Instrument Type	Measurement or Purpose
Bolometers	Output power monitor
Current loop	Instability monitor
Diamagnetic loop	Plasma energy
Electron cyclotron emission	T_e
Faraday cup	End loss current
Fast magnetron gauge	Background gas pressure
Interferometer (microwave, FIR, IR)	Plasma density
Langmuir probe	Halo density
Neutral flux spectrometer	Plug potential monitor
Neutron counter	Fusion rate
Residual gas analyzer	Gas content, esp. T
RF probe	Instability monitor
Secondary emission detector	Sloshing ion monitor

B. SPECIAL USE

Instrument Type	Measurement or Purpose
Ion spectrometer	End loss spectrum
Magnet alignment	Minimize radial transport
Neutron spectrometer	T_i, scattering rate
Thomson scattering	Calibration of n_e, T_e
UV spectrometer	Impurities, transport
Video imaging (IR, visible, UV, X-ray)	Trouble-shooting, heating, impurities, beam aiming, leaks
X-ray spectrometer	Electron heating

TABLE 10-6: Radiation Tolerance of Components for Fusion Diagnostics.

Component or Material	Ionization Dose (Rad (Si)): 10^1 10^3 10^5 10^7 10^9 10^{11} 10^{13} / Neutron Fluence (n/cm2 (1 MeV)): 10^{10} 10^{12} 10^{14} 10^{16} 10^{18} 10^{20} 10^{22}	Dominant Damage Mechanism
Nuclear Diode (PIN,Si,Ge)	----------*	B
LF Transistors	------------	B
Phototransistors & Opto-Couplers	------	B
CMOS IC	------------	I
Photodiodes (Si,HgCdTe,PbSnTe)	------------	B
Optical Fibre	----------------------	I
Gunn Oscillator	---------	B
Linear Circuits	---------	B
Zener Diodes	---------	B
Electrolytic Capacitors	----------------	I
CMOS IC	----------	I
Optics	--------------------	I
HF Transistors	-----	B
Photo-Tubes (PMT, TV)	---------------	I
Power MCS	---	I
Pyroelectric Sensors	-------	B
PTFE (Mechanical Strength)	-----------	I
JFET (Si, GaAs)	------	B
Hardened MOS IC	-----	I
Scintillators	---------	I
Diffused Si Resistors	-----	B
Ferroelectrics	-------------	B
Capacitors	---	I
Plastic Insulators	-----------	I
Photoconductive Photosensor	-----------	B
Resistors	------	I
Hall-Effect Sensors	---------	B
Piezoelectric Crystals	----	B,I
Inorganic Insulators	-----	I
Magnetic Materials	-----	B
Radiation Resistant Alloys	----	B

*Bar begins at dose and fluence at which most sensitive components show significant degradation and ends at dose and fluence at which least sensitive components shows significant degradation. Dominant damage mechanism: B = Bulk Damage, I = Ionization.

Sources

The sources of information for the considerations made here are, of course, manifold and wide-spread. In particular, however, the authors would like to express their gratitude to K.M. Young of the Princeton University, J.E. Osher of the Lawrence Livermore National Laboratory, J.F. Baur and D. Drobnis of GA technologies, and W.R. Wing of Oak Ridge National Laboratory for many fruitful discussions and many of the ideas presented here.

Bibliography for Chapter 10

1. "Tokamak Plasma Diagnostics", Equipe TFR, Nuclear Fusion 18, 5 (1978), pp. 647-731.
2. N.C. Luhmann, Jr., and W.A. Peebles, "Instrumentation for Magnetically Confined Plasma Diagnostics", Rev. Sci. Instrum. 55 (3) March 1984, pp. 279-331.
3. S. Battisti et al., "Radiation Damage to Electronic Components", Nuclear Instruments and Methods 136 (1976), pp. 451-472.
4. T. Iida et al., "Fusion Neutron Irradiation Effects on Electric Characteristics of Seminconductor Electronic Devices", J.Nucl. Sci. Technol. 21 (1984), pp. 634-641.

References for Chapter 10

|1| D.R. Slaughter and K.M. Young, Private communication; Princeton

|2| J.F. Baur et al., "Radiation Hardening of Diagnostics for Fusion Reactors", General Atomic Company Report GA-A16614

11 MAINTENANCE

11.1 Introduction

Maintainability of fusion reactors, because of its impact on availability, is one of the most important requirements in the realization of fusion power as a viable energy source for the future. It is therefore imperative, that any device which is intended to have tritium in it at any time in its life, be designed with remote maintainability as a prime consideration. The next generation of fusion devices, those of the INTOR class such as NET and TFCX all fall into this category.

Before describing the remote maintenance needs of a DEMO reactor and where these needs can be satisfied, it is useful to elaborate on the types of maintenance and classes of components. Components are usually classified into four classes:

Class 1 - Components which have a lifetime of < 0.1 of the machine life. They are repaired / replaced during machine shutdown.

Class 2 - Components which have a lifetime = 0.5 - 1.0 of machine life, but have a low reliability. Such components are replaced during a regular maintenance period.

Class 3 - Semi-permanent components designed for the machine lifetime, with high reliability and / or redundancy. Such components as magnets would fall into this category. Their replacement will require extended machine shutdown.

Class 4 - Auxiliary components or those of long lifetime. Their replacement may not be required or can be easily achieved.

There are basically two types of maintenance, contact and remote. Contact maintenance is performed on non-activated components and is usually limited to areas outside the biological shield. Some contact maintenance of short duration can be performed inside the biological shield 24 hours after shutdown and as long as the reactor shield is intact.

Tritium contaminated components can be handled with contact maintenance by people in appropiate clothing (bubble suits). Contact maintenance can consist of in-SITU inspection, repair or replacement. Such maintenance is considered state-of-the-art and requires little or no development.

Maintenance of activated components has to be performed by remote control. Remote maintenance can be divided into three categories, in-SITU repair in a vacuum or protective atmosphere, component replacement and maintenance in a hot cell. Each will be discussed separately.

In-SITU remote maintenance can consist of optical inspection of surfaces, welds or diagnostic equipment, vacuum leak checking, measurement of coating layer thickness and a multitude of functional tests. Equipment needed to perform such tasks by and large exists today, however, development is needed to radiation harden it. When the question comes to in-SITU repair, it is a different matter. Although some general purpose equipment is available today for remote welding, machining, etc. most of the equipment needed will be special purpose. It is, therefore, hardly conceivable to carry out extensive in-SITU repair with presently available equipment. A considerable amount of investigations and development is needed in this area.

The preferred method of maintenance is component replacement. Although it may appear extravagent at first, in the long run it saves costs by increasing availability. The key elements needed are remote viewing, making and breaking connections be it by welding or otherwise, transporting, positioning, controlling and checking. It is very likely that existing remotely controlled tools can be adapted to special purpose equipment to perform many of the tasks. The new features which will be needed for fusion are the large sizes and somewhat cumbersome components as well as large masses, restraints needed to react electromagnetic loads during operation and the ability to operate in high radiation fields. However, a great deal of experience can be obtained on full scale and mass mockup devices.

Finally there is remote maintenance in hot cells. These operations are similar to those performed in hot cells at the present time with the difference that the components may be larger and heavier Fig.11-1. Operations include inspecting, dismounting, repairing and assembling components. Some of these tasks are similar to those performed at the reactor but more time will be available as well as more access and better viewing. Tests and development can be performed on realistic mockups.

11.2 Maintenance Needs of DEMO

Table 11.1 lists twelve components of a toroidal DEMO which will have to be maintained, giving the class of the component, its dimensions and weight.

A contained transfer unit (CTU) is supposed to be needed for transport of highly contaminated components. This complicates the remote operations. Further, development is required for removing and replacing components from vacuum enclosures without breaking the vacuum.

The accuracy of remote positioning in most cases is probably not a severe problem. Components which need precise alignment, such as e.g. ion sources, will have dowels and stops to insure correct positioning.

Problems are likely to be caused by the maintenance time requirements which can not be assessed at this early state. Recent design studies for NET have shown that e.g. a blanket sector may have more than 60 coolant pipe connections and nearly 100 m seamwelds. Thus considerable development will be needed in order to perform positioning, connecting and checking within the prescribed replacement period.

Automated and computer controlled remote maintenance operations are probably needed in addition to appropriate components and equipment design.

CLASS	COMPONENTS	TOROIDAL "DEMO" Dimension [m] Weight [t]	"NET" CLASS Dimension[m] Weight [t]	"TASKA" CLASS Dimension [m] Weight [t]
1	Divertor/Lim.	5.5x5.0x1.0/50	3.9x1.0x0.4/ 5	
			Remote Replace.	
	Neutral Beams	(FINTOR-D)		
1	Ion Source	0.75x0.65x0.65/<1		0.75x0.65x0.65/<1
2	Other Comp.	2x2x3/10		2x2x3/10
2	Cryop./Getters	3x3x0.3/<1		3x3x0.3/<1
				Remote Replace. Contained Transp.
2	RF Heating	1.5x2.6x6/50	1.0x1.7x2.4/ 10	4.45x2.5x0.88/35
			Remote Replace.	Remote Replace. Contained Transp.
2	Blanket Module	8.6x5.8x5.9/290*)		3x3x1.4/50
				Remote Replace. Contained Transp.
2	Vacuum Pump	2.8Dx4.6/ 25	2.6Dx4/ 25	3.2Dx4.3/ 25
			Remote Replace.	Remote Replace.
2	Coolant Connect.	0.10 - 0.50	Remote Replace.	Remote Replace.
2	Diagnostic Equip.	/<1	Remote Replace.	Remote Replace.
3	PF Coil	R = 13 m/350	R = 6.0/50	R = 3.5/150
			Contact Maint. If Shield Intact	Contact Maint. If Shield Intact
3	TF Coil	8.6Dx10D/400	4.0Dx5.9D/30	Barr.Coil/350
			Contact Set-Up Remote Replace.	Contact Set-Up Remote Replace.
3	Cryostat Dome	20Dx6/200	13Dx2.6/ 25	
			Contact Maint.	

Table 11.1 Dimensions, Weights and Maintenance Characteristics of typical Components

*) incl.empty space between inboard and outboard

11.3 Conclusions

Many issues for the maintenance of fusion devices must be solved in simulation tests on full scale and full weight mock ups. These tests must precede the construction of any fusion engineering test facility such as a TASKA class machine or a NET-EP device.

Operation of such a machine will provide maintenance experience in realistic environments (magnetic field, radiation).

Although in principle no unsolvable maintenance problems seem to exist, there are doubts whether time requirements can be met.

A multitude of complicated components must be maintained during economically limited reactor shut down periods. Developments and improvements expected in an engineering test facility will certainly be indispensable for DEMO operation.

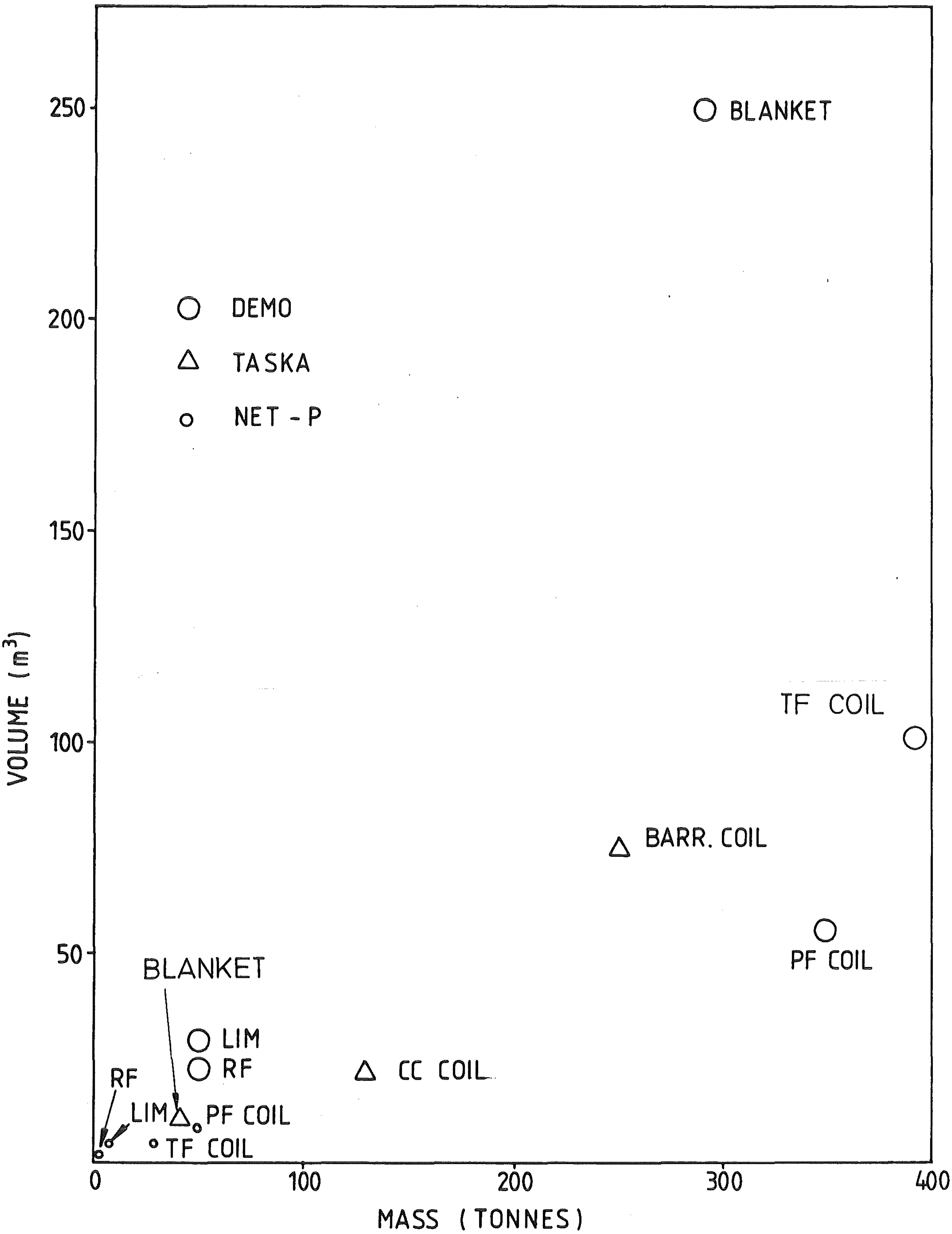


Fig. 11-1: Typical volumes and masses of components in future fusion devices

12. SAFETY

In general, safety considerations are necessary for every kind of fusion devices. Therefore both strategies discussed in Chapter 1 fulfill the safety related requirements. The safety concerns for fusion are still in the process of being identified and quantified and much work, both theoretical and experimental, remains to be performed. While it is expected that the safety problems of fusion systems will be less severe than those of fission reactors, nevertheless this must be demonstrated in a quantitative fashion. In the following some relevant safety related concerns of a toroidal DEMO will be discussed in relation to how other facilities can contribute to responding to them. The various areas are considered in random order and not in order of importance.

A system or event becomes a concern for safety when it has the possibility of leading to a condition which would result in a hazard to the plant operating staff or to the public. The primary hazard to the public would be caused by those events which could lead to the release of radioactivity either from the radioactive fuel (tritium) from the plasma, the blanket and its reprocessing system, the fueling system, or the storage system or from the radioactivity induced in the system by the neutrons produced in the fusion process.

Events which could lead to the release of radioactivity can be caused by direct failures in these systems or can be caused indirectly through the failure of other systems in the plant. Even if an accidental event would not lead to a radioactivity release and thereby lead to a hazard to the public, it might result in a hazard to the operating staff and to the plant itself. Consequently any safety study must address a wide variety of problems and not concentrate simply on those directly involving radioactivity.

To analyze the effects of a particular condition or event it is first necessary to specify the source term for the event. For example the amount, distribution, and nature of the radioactivity must be determined to be able to assess the effects of a release of blanket coolant. Similarly evaluation of the consequences of a magnet failure of some sort require, among other things, a knowledge of the stresses and stored energy in the magnets. Most of the safety related work in fusion performed to date has been in quantifying these source terms. In addition to analytical work or to supplement it, experimental studies must be performed. These can be in the form of special experiments designed to answer specific questions or can be in the nature of integral operational tests to verify the performance of safety related systems. An example of the first type might be an experiment to measure the reaction between water and a blanket coolant. An example of the second type might be the operation of a large superconducting magnet to test the reliability and effectiveness of the magnet protection in the context of its regular operating routine.

The phenomenon of the plasma disruptions must be of no safety concern for the DEMO. If it is not possible to control a plasma disruption, the construction of a DEMO is considered to be impossible. This is true for safety and operational reasons. Hard plasma disruptions can deposit enough energy on the first wall or in the divertor to cause significant damage, such as melting of structure. It is anticipated that through the operation of a NET-P device sufficient understanding can be developed to operate the DEMO without disruptions. With respect to disruption initiation the requirement is to understand the factors causing disruptions well enough to avoid them in the design. For the investigation of the disruption consequences, especially the questions of the energy deposition time, the spatial distribution of the energy deposition, and the current decay time during a disruption, plasma current disruptions have to be understood in more detail.

In the following discussion an attempt is made to state the safety related requirements for a toroidal DEMO and answer the question as to whether the various other fusion devices can provide sufficient information for the DEMO or whether some special experimental facility needs to be built.

12.1 TRITIUM

As discussed in the tritium section of this report (Chapter 7) tritium will be present almost throughout the system and large quantities must be handled in relatively complicated systems. Because of its radioactivity, volatility and its tendency to escape from the plant, the presence of tritium is considered to have the most potential radiological hazard in a fusion device. Minimizing the tritium losses (occurring by permeation through the different structures, desorption and leakage) in all components of a plant is a necessity and a tremendous challenge for the designers. The problem is to understand the phenomena of adsorption, implantation, recombination, permeation and desorption, so that the machine may be designed to accomodate them. The effect of irradiation on tritium movement and inventory is of importance, too. The usefulness of a facility like the NET-P will be limited since long term effects cannot be evaluated. The situation in the TSTA is even worse. TSTA will have no radiation effects taken into account in either the short or long term. However, a TASKA-class facility could provide information on these properties at suitable neutron flux and fluence.

Another safety related problem with tritium arises in the analysis of the consequences of a tritium release. It is postulated that most of the tritium would be released as a gas, e.g. HT. Because of the much higher radiological hazard potential of HTO or T_2O compared to the gaseous form HT or T_2 data about the degree and the conditions of the conversion of tritium gas to tritiated water have to be collected. These data are important not only for the specific plant components but especially also for the different environmental conditions, in more detail for example also the conversion in flora and in bacteria. Corresponding parameters are best measured at some facility like TSTA and in specially designed experiments.

12.2 MAGNETS

The overall problem with respect to the magnets is to ensure that their integrity is retained. This is important not only from a safety point of view but also from a cost and plant availability point of view. These items are likely to be quite expensive and replacement will be a difficult and time consuming process. The major considerations for the magnets are listed in Chapter 9. The large amounts of stored energy need a controlled conversion in the case of an unbalance. Development of reliable fault detection and protective energy discharge systems to cope with quenches and short circuiting of s/c coils is necessary. Of major importance for the fault detection system is the ability to discriminate between the different failure signals. Discharges due to false signals should be minimized. It should be mentioned here that quenching is considered to be an abnormal operational and not an accidental event. A prolonged arcing of the superconductor could lead to magnet failure with serious consequences. Similarly a loss of vacuum insulation or a loss of coolant could lead to magnet failure. In both of these cases the requirement for the DEMO would be that the system be designed and operated in such a fashion as to avoid their happening. If they were to happen it must be possible to detect them and to mitigate the consequences either through an active or passive system. Mitigating in this context means preferably limiting the consequences so that damage would be limited to that particular magnet in which the effect took place. Any facility which operates with large superconducting magnets has similar problems and thus can provide a background of experience. This would include such facilities as the NET-P (if superconducting) or a TASKA-class fusion system, the LCT and TESPE experiments and the TORE SUPRA plasma device. An additional problem which could arise from one of the previous conditions or could arise due to a mechanical failure is that of missile generation. In the DEMO this hypothetical event, which is considered to be extremely unlikely could be a rather major accident. The subsequent requirements for the design are that such missile generation be analyzed and steps be taken to mitigate the consequences. This can be done through design as well as through the location of sensitive components and the use of shielding for relevant structures. The design and operation of the various facilities mentioned above should provide sufficient information to meet this requirement.

12.3 CHEMICAL REACTIONS

In any consideration of safety requirements the effects of chemical reactions must be considered. The investigations of breeder material/coolant interactions are strongly dependent on the choice of the breeder material and the coolant, and on the design. In case of an accident (for example caused by a tube break) not only the kind of reaction between the lithium compound and the coolant are of importance but in the further sequence of events also reactions of lithium with air or concrete are possible. The behaviour of the different reactions is strongly influenced by the respective boundary conditions. It is also important to note that the transfer of results gained by small scale tests to large scale systems may be difficult. The break of a high pressure water coolant tube inside a liquid breeder module leading to rapid pressure oscillations may cause also dynamic deformations possibly resulting in module failure. Liquid breeder material and water could then enter the plasma chamber possibly resulting in chemical reactions, leading to fires or explosions. The requirement here again is that these interactions be understood so that their effects can be analyzed and steps taken to mitigate the consequences of the interaction. This work must be performed in separate simulation experiments without reference to any of the proposed fusion devices.

In the operation of the DEMO it is quite likely that either in the divertor region or from the first wall itself fine metallic dusts will be generated. The requirement for the DEMO is that the production rate, character, deposition, and final disposition of these materials must be known so that their effects can be evaluated and handled. In addition to the problems related to plasma physics and the design (like the sputtering behaviour, and the kind and places of depositions) the general, safety related question exists: what is the state of the dusts in the case of an accidental event?

The potential risks are combustion and radiological risks. The dusts have a very small size. A very fast chemical reaction may occur, if oxygen, air or steam enter to the plasma chamber in case of an accident. The dust aerosols may react with tritium or be contaminated by tritium. Corresponding to their origin the dust particles also contain activation products. During maintenance operations the small and light dust particles may be inhaled by the workers.

Here the NET-P machine can provide a significant amount of information however long term effects cannot be evaluated. The characteristics of TASKA is such that at best only a limited amount of information can be obtained from them. Thus the information from NET-P must be supplemented by simulation experiments.
The internal structure of the DEMO will be operating at high temperatures and a vacuum or coolant leak has the potential of suddenly exposing these surfaces to air, water, or perhaps some other material. This in turn could result in chemical reactions with potentially severe consequences.

The possible needs for experiments related to oxidation phenomena in case of an accidental air or water entrance into the torus is dependent on the material selection. Of special interest is the interaction between divertor/limiter components (tungsten) and water. Oxidation of structural material is a question of structure mechanics and of safety. The requirement for the DEMO is that these reactions be understood for design, analysis, and mitigation. This work must be done in a simulation facility.

Concerning hydrogen combustion in case of an accidental release of the isotopes D or T, the design requirement is that the maximum possible release of hydrogen does not lead to a concentration in the whole containment atmosphere which is sufficient for a hydrogen deflagration or explosion. However, it is necessary that local areas of higher concentrations be avoided by design.

In the case of an accidental toxic material release special experiments seem not to be necessary. The purification and filter systems for activation products have to be designed in such a way that they are also able to retain the aerosols from toxic materials.

12.4 THERMAL-HYDRAULICS

The primary conditions of concern in thermal hydraulics are the loss of coolant or loss of flow conditions in a blanket module. In addition to both of the above conditions a further requirement is that the condition can be detected, the subsequent events analyzed and measures be taken to mitigate any consequences. First of all, simulation experiments must be relied upon just as they form the basis for similar analyses in fission reactors. Since NET-P will have a limited capability for blankets, no or less information can come from this source. TASKA however, will have blankets and some limited information could be obtained.

Another condition that could arise results from an air or coolant leak into the vacuum. The requirement again is to be able to detect such a leak and to be able to analyze the resultant effects to the design in order to mitigate their consequences. Thus the ability to predict the transport of air or coolant into the vacuum must be available. Since this is clearly an undesirable situation in an operating plant, the use of simulation experiments is necessary.

12.5 OTHER EVENTS

Safety concerns arise from the consideration of other sources. Among these are earthquakes, internal events such as fires or missiles and external events such as winds, airplane crashes, etc. The basic requirements for all these cases are that the event be defined, the condition avoided if possible, e.g. fires, and the system be designed to cope with these events. The experience in designing either NET-P or TASKA will certainly supply much useful information for the DEMO even considering that NET-P has no blanket and the TASKA geometry is not comparable. Further the work that has been done for the fission plants should be very useful.

12.6 SAFETY INSTRUMENTATION

Instrumentation will be required in the DEMO to assure all systems with their protective devices are working correctly and that in the event of a failure of a component appropriate measures are taken. The key question of each safety and protection system is: How much time is available for starting counter-measures if an accident will occur?

With respect to sensors there is a specially strong need to develop, to prove, and to improve suitable sensors for different areas and under fusion environment conditions (high energy neutrons and other radiation effects, high and low temperatures, high magnetic fields, radio-frequency electric fields). The basic requirement is that these devices operate with a failure rate less than some design value which is factored into the overall system design. NET-P can provide much confirmatory information on the operation of these instruments with the exception that it cannot simulate the effects of neutron fluence in those systems where it is important. TASKA however, can provide this type of information. Neither device can evaluate those instruments under abnormal conditions. This must be done in simulation devices. However, it will be very difficult to simulate the total environment in these tests and to get experience about long time behaviour.

12.7 ENVIRONMENTAL AND POWER CYCLE CONCERNS

Environmental concerns have a basis in safety and are thus included in this section. The primary environmental concerns have to do with radioactivity. The overall requirement for radioactivity from an external point of view is that the dose received by an individual outside the plant from routine releases be less than 10 mrem per year (US reference value). Since other releases are expected, the dose due to tritium is set at 5 mrem per year. For accident situations, the requirement is that no credible accident, i.e. probability $> 10^{-6}$ per year, should result in a dose greater than 25 rem. Experience with NET-P, TASKA and facilities such as TSTA should provide sufficient information to assure that this goal can be met in the DEMO.

The DEMO will generate a certain amount of radioactive wastes. An estimate of the amount of high level waste to the DEMO is 88 t per year. Since NET-P is a low fluence device it will not provide much experience with either high or low level waste. TASKA on the other hand would generate 5 - 12 t per year of high level wastes, more typical of DEMO quantities, plus low level wastes similar to that of DEMO. This information is supplemented by the experience gained in handling with high and low level wastes in fission reactors and fission fuel reprocessing plants.

The power cycle also has concerns which are safety related. Since in some blanket concepts the heat is removed by a liquid metal coolant, at some point in the cycle there must be a liquid metal driven steam generator. One of the major concerns that results is the leakage of tritium from the breeder coolant through the steam generator into the secondary circuit. Since NET-P has no power cycle no information would be available from it. While the basic design information must come from TSTA and other simulation experiments TASKA can provide design and confirmatory operating experience.

A second concern with the steam generators is the possibility of a tube leak or rupture followed by a liquid metal water reaction. Again here the basic information regarding the consequences of this interaction must come from analysis and experiments in special facilities. At the same time the presence of a power cycle on TASKA provides design and operating experience.

In addition to these direct safety concerns the ability of the plant to handle both normal and anticipated off-normal transients safely is of concern. Here again NET-P can provide no information but TASKA can provide some limited confirming operating experience.

12.8 Conclusions

Much safety related information can only come from simulation experiments. This is especially true for those situations in which a direct test would put the plant at a significant risk of damage. The combination of the NET-P and TASKA-class devices provides design and operational experience plus confirmation of operation in a DEMO environment, i.e. in the presence of radiation fields, magnetic fields, and appropriate heat fluxes and temperatures.

BIBLIOGRAPHY

Since the safety related statements are valid in a more general sense, intentionally no special references were cited. However, for further information a few basic references are listed in the following.

INTOR PHASE TWO A, REPORT OF THE INTERNATIONAL TOKAMAK REACTOR WORKSHOP,
IAEA, Vienna, 1983

BACKGROUND INFORMATION AND TECHNICAL BASIS FOR ASSESSMENT OF ENVIRONMENTAL IMPLICATIONS OF MAGNETIC FUSION ENERGY
J.B. Cannon, Ed, DOE/ER-0170 (August 1983)

SAFETY ASPECTS OF FUSION,
M.S. Kazimi, Nuclear Fusion, Vol. 24, No. 11 (1984)

POTENTIAL AIR CLEANING PROBLEMS IN FUSION REACTORS,
J.G. Crocker, 17th DOE Nuclear Air-Cleaning Conference,
Aug. 2-5, 1982, Denver, CO

FUSION SAFETY,
IAEA-TECDOC-277

POTENTIAL FAILURES AND HAZARDS IN SUPERCONDUCTING MAGNET SYSTEMS FOR FUSION REACTORS,
F. Arendt, P. Komarek, Nuclear Technology/Fusion, Vol. 1,
Oct. 1981

A DEMONSTRATION TOKAMAK POWER PLANT STUDY (DEMO)
ANL/FDP/82-1

MARS-MIRROR ADVANCED REACTOR STUDY, Final Report, Vol. 1-B,
UCRL-53480

SAFETY ASPECTS OF FIRST WALL AND BLANKET,
H.Th. Klippel, EUR XII 324/17, NET Report Nr. 17

IAEA STATUS REPORT ON FUSION SAFETY,
in preparation