OVERVIEW OF THE U.S. FUSION TECHNOLOGY PROGRAM

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ABSTRACT

The U.S. magnetic fusion R&D program is focused on four key technical issues: magnetic confinement systems, properties of burning plasmas, materials for fusion systems, and nuclear technology of fusion systems. The technology program now has in place a number of activities which provide major contributions to resolving the near- and long-term aspects of these key issues. This paper presents an overview of recent technical accomplishments of key U.S. fusion technology programs in the areas of plasma technology, nuclear technology and materials, and system studies.

Plasma technologies (magnets, heating and fueling) are being developed mostly in existing plasma experiments. The construction of the large super-conducting magnet systems for MFTF-B and LCT has been completed.

Nuclear technology elements include the blanket/first wall, radiation shield, tritium processing system, and nuclear elements of plasma-interactive components. Initial testing of the Tritium System Test Assembly, which simulates the fuel cycle, has been successfully completed. Relevant data for the blanket technology program are generated from an MHD fluid flow facility, a number of corrosion loops, and a number of fission reactor irradiation experiments. Measurements and analysis of tritium production and other neutronics parameters in Li2O blanket mockup have been performed in collaboration with JAERI.

Plasma-matertials interaction (PMI) and structural alloy development are the two major materials programs. PMI problems are addressed in an number of existing plasma confinement experiments and in two dedicated test stands. The present emphasis of the structural alloy development program is on reduction of long-term activation, improvement of theoretical modelling of radiation effects, and utilization of fission reactor irradiation.

System studies for near-term experimental devices include preconceptual design and system trade-offs for a compact ignition device and an engineering test reactor. Preconceptual design studies for commercial reactors are being performed for a reversed field pinch, an advanced tokamak and an improved tandem mirror.

International cooperation is a significant aspect of a number of the above programs.
1. INTRODUCTION

The present U.S. strategy for magnetic fusion research and development (R&D) is stated in the Magnetic Fusion Program Plan (MFPP) [1]. The goal of the U.S. magnetic fusion program is to establish the scientific and technological base required for fusion energy. As a focus for the remaining work necessary to reach this goal, four key technical issues have been identified in MFPP. The four issues are:

- **Magnetic Confinement Systems** -- developing a plasma science understanding leading to improved confinement concepts suitable for commercial applications of fusion energy.
- **Properties of Burning Plasmas** -- understanding the effects introduced when the plasma is internally heated by the fusion reaction.
- **Fusion Nuclear Technology** -- developing nuclear technologies unique to fusion for the commercial application of fusion energy.
- **Fusion Materials** -- developing materials which will enhance fusion economic and environmental potential.

An important element in the present U.S. effort is the Technical Planning Activity (TPA) [2]. The TPA is carried out by the U.S. fusion community to assist the Department of Energy's Office of Fusion Energy by developing a methodology for technical planning, and by preparing technical planning documents in support of MFPP.

The U.S. fusion technology R&D effort can be broadly classified into three areas directly related to the key issues: 1) plasma technology, 2) nuclear technology, and 3) materials. The plasma technology area includes those components necessary for creation and maintenance of plasmas such as magnets, heating, fueling, and plasma-interactive components (PIC). The nuclear technology area encompasses those components required to extract and utilize fusion energy, to produce and recover fuel, and to protect components and personnel from radiation. Included in this area are the blanket/first wall, radiation shield, tritium processing, power conversion and nuclear aspects of PIC. The materials area includes structural and non-structural blanket materials, and special materials such as insulator and magnet materials.

The current effort in plasma technology is focused on supporting present confinement experiments, and providing the components necessary for near and medium-term devices such as those for demonstrating plasma ignition and long burn. While nuclear technology and materials R&D is providing significant input to near-term devices, the major focus is on the long-term, long lead-time issues whose resolution is necessary for enhancing the economic and environmental attractiveness of fusion as an energy source.
In addition to the R&D in the three technology areas, there are a number of systems analysis and design studies underway. These studies provide a crucial framework in which the technical issues can be identified and innovative ideas for competitive fusion energy systems can evolve.

This paper provides an overview of the present activities in the U.S. fusion technology program. Section 2 is devoted to plasma technology. Section 3 is a summary of present efforts in selected areas of nuclear technology and materials. Highlights of systems studies are presented in Section 4. Although the paper covers many topics, no attempt has been made to be comprehensive in summarizing all areas of R&D in the U.S. Therefore, the topics covered and the results presented should be viewed only as examples.

2. PLASMA TECHNOLOGY

The pace at which parameters are improved and understanding is developed on plasma physics devices is frequently determined by the availability of reliable, high performance and economical components which heat, fuel and magnetically confine the plasma. Present R&D programs on plasma technologies are focused on supporting the resolution of the MFPP key issues of confinement and burning plasmas. The key elements of plasma technology are: magnets, plasma heating, plasma fueling, and plasma-interactive components. Examples of U.S. activities in these four areas are given in this section.

2.1 Superconducting Magnets

Development of superconducting magnets for fusion applications covers a wide spectrum of activities, from the subtleties of composition and processing variables for superconducting materials, through conductor manufacturing processes, to the construction and operation of large magnets whose scale and performance approach the requirements of future fusion reactors.

In the area of large-scale magnets, there are two distinct categories of activity. One is the design and construction of magnets that are incorporated in plasma confinement devices and which are operated as required to support physics experiments. The other activity consists of component development programs, in which conductors and coils are produced solely to yield information on manufacturing and performance under test conditions. In such programs it is typical to investigate several different designs, to provide sufficient diagnostic instrumentation in the coils to elucidate performance characteristics, and to extend test conditions beyond the nominal design objectives in order to explore limits of operability. Examples of project-related magnet development are Tokamak-15 in the USSR, Tore Supra in France, and MFTF-B in the U.S. Magnet development programs include extensive work in Japan on both toroidal
and poloidal magnets for tokamaks and the international Large Coil Task (LCT) for development of toroidal field coils, in which the U.S. is the Operating Agent.

2.1.1 Large Coil Task

In 1976-77, Oak Ridge National Laboratory (ORNL) prepared specifications for test coils, contracted with three U.S. industrial teams to design and build one coil each, and started on design of the International Fusion Superconducting Magnet Test Facility (IFSMTF). In 1978, Japan, Switzerland, and the European Atomic Energy Community (EURATOM) joined with the U.S. in the LCT, agreeing to provide one coil each [3]. All six coils are being tested together in a compact toroidal array in the IFSMTF.

The LCT has been called "a technological experiment." This is an especially apt description for the U.S. coil program, where the approach was to determine the response of three selected industrial design teams to a common set of requirements. Each team retained the latitude to choose the internal configuration and fabrication processes that it deemed best. When the international collaboration was organized, the same basic specifications were used by all participants. As the experiment turned out, the chosen designs covered the spectrum of concepts favored in 1977-78 to meet requirements of stable operation at 8-T peak toroidal field, with tolerance for externally imposed pulsed fields up to 0.4 T.

Four of the basic design decisions faced by the LCT teams were

- the choice of the superconductor material,
- the method of cooling the superconductor,
- the method of winding the conductor, and
- the structural configuration, whether concentrated in an exterior case or distributed throughout the winding.

Features of each chosen coil design are listed in Table I. Each coil has a D-shaped 2.5 x 3.5 m bore, contains about 7 MA-turns (5 to 7 km) of conductor, and weighs about 40,000 kg. All coils are highly instrumented, with each having over 200 sensors of temperature, voltage, strain, displacement, helium pressure, and acoustic emission. The IFSMTF can accommodate up to six test coils in a compact toroidal array that is contained in an 11-m-diameter vacuum vessel (see Fig. 1). Tests with a lesser number of coils can be performed, however, since the coil and test stand structures are designed for the out-of-plane forces that are involved in partial arrays.

Shakedown operation of facility systems and preliminary tests with three coils in the test stand were carried out in 1984. The last three coils were installed in 1985.
Tests of the six-coil array that began in January 1986 demonstrated first the capability of all coils to be cooled to operating temperature without excessive thermal stress or opening of helium leaks. Then each coil was energized to its rated current while internal heaters were used to simulate heating such as might be encountered in a fusion reactor, thereby demonstrating achievement of the design goal of stability. Various tests are exploring the reliability and effectiveness of practical quench detection and protection provisions in multicoil arrays. Meanwhile, valuable information on equipment performance and system availability is being obtained by the operation of the helium refrigeration, liquid nitrogen and vacuum systems that limit the heat ingress to the 400-tonne test array to about 1 kW.

The test program, which was developed jointly by the four participants in the LCT, is continuing. Goals include further demonstration of reliable operation at full design current and toroidal field, with superimposed pulsed vertical fields. Eventually, the limits of operability at higher fields, current densities, and temperatures will be explored by experiments at extended conditions. It is expected that the planned test program will be completed in 1987.

2.1.2 MFTF-B Magnets

The MFTF-B facility at LLNL represents a tremendous accomplishment in superconducting magnet technology for fusion. It is the largest superconducting magnet system yet operated. The magnet system is depicted schematically in Figure 2. Total length of the magnet system is about 55 m. The diameter of the central solenoids (S1 through S6, East and West) is 5 m, and the maximum field is 3.1 T. The bore diameter of the axicell inset coils (A21, East and West) is 0.36 m, but maximum field is 12.7 T. These coils used a multifilamentary-Nb$_3$Sn composite superconductor, whereas all others used NbTi. The "yin-yang" pairs (M1 and M2, East and West) are very large, weighing over 360 metric tons each. In their present configuration they produce 5.8 T maximum field, but the first pair for MFTF-A was tested to 7.7 T in 1982 [4]. In all, the magnet system comprises a 4.5 K cold mass of 1050 metric tons. The entire system, including the 11 kW refrigeration system, was successfully tested in February 1986. All magnets performed flawlessly to full design field with no quenches.

2.2 Plasma Heating

The plasma heating program in the U.S. consists of five elements: Electron Cyclotron Heating (ECH), Ion Cyclotron Heating (ICH), Lower Hybrid Heating (LHH), Neutral Beam Injection (NBI), and Ohmic Heating (OH). The plasma heating program also includes technical issues related to current drive. The physics and technology of the various ICH heating modes are being developed to an extent sufficient for selection of a specific
ICH heating mode for the short-pulse ignition device. In parallel, the key issues of ECH, LHH and neutral beam heating are being explored in order to provide the basis for selecting the optimum mix of heating techniques for the long-burn demonstration facility.

Significant progress has been made in Ion Cyclotron Radio Frequency (ICRF) heating development. One of the largest activities is the ICRF heating program at ORNL. Components produced in this program are already in use on experiments throughout the world, including Doublet III-D (DIII-D), TMX-U, Alcator-C, and TEXTOR. Designs have been developed for implementation on Tore Supra and ASDEX. Examples of recent developments in the ICRF components of vacuum feedthroughs, compact loop antennas, high-current capacitors, and folded waveguides are presented below [5].

2.2.1 Vacuum Feedthroughs

The first product of the ORNL radio frequency (rf) technology program was the 50-Ω vacuum feedthrough with a cylindrical ceramic. This feedthrough has now received field testing under a variety of operating conditions as a result of its immediate application at several laboratories. Attractive features of this basic design include high voltage and current capability, a demountable ceramic assembly, cooling passages for steady-state operation, low insertion voltage standing-wave ratio, and wide frequency bandwidth.

The feedthrough built for TEXTOR is used at two locations in the coaxial transmission line: near the antenna, with vacuum on both sides of the feedthrough, and at a second position farther from the antenna, with nitrogen pressurization on one side. Two versions of this feedthrough were constructed: one with niobium end caps for the ceramic and one with copper end caps. The niobium proved to be superior in voltage rating, reaching peak voltages greater than 150 kV for 100 ms pulses, but the braze joint could not be made reliably and was somewhat fragile. For the case of the copper end caps, nickel and gold platings were applied and tested for effects on voltage breakdown, with some improvement. Table II summarizes the voltage breakdown limits for the various cases.

A prototype 25-Ω feedthrough has been fabricated for ASDEX within the external dimensions of existing ASDEX feedthroughs. This feedthrough incorporates a buffer vacuum between the machine vacuum and the pressurized transmission line. The ceramic is brazed to the hourglass outer conductor. This design should possess greater mechanical strength than the 50-Ω design. A potential disadvantage is that the ceramic cannot be as easily replaced as in the 50-Ω feedthrough in case of a failure in the braze joint.
2.2.2 Compact Loop Antennas

A low-power prototype cavity antenna has been installed on DIII-D [5]. The primary purpose of this antenna is to measure plasma loading for a wide range of plasma conditions and frequencies and to compare the results with calculations by GA Technologies and ORNL. The antenna can be moved radially over a small range. The antenna has a Faraday shield consisting of two layers of copper-plated Inconel rods with 1.5 mm thick graphite plates brazed to the rods on the plasma side. Extensive thermal and stress analysis calculations were performed for this antenna to verify the survivability of the design under the worst-case tokamak conditions, such as major disruptions. The results of this analysis are applicable to other large tokamaks, providing a useful base for other compact loop designs, including Tore Supra. Because the prototype antenna will only operate at low power during the first year of DIII-D experiments, inertial cooling was determined to be adequate, and active cooling will not be used.

A development version of a high-power DIII-D cavity antenna is under construction for use in the Radio-Frequency Test Facility at ORNL. It will be operated during the second half of 1986 when the 1.5 MW transmitter becomes operational. This antenna was designed for easy modification so that new antenna components can be easily tested at high power in a plasma environment.

A preliminary engineering design has been done for a resonant double loop (RDL) antenna [5] system for Tore Supra. This design consists of two RDL antennas side by side behind a common Faraday shield. The entire structure is mounted in a large port and can be moved radially to couple to plasmas with different minor radii. The antenna incorporates water cooling, allowing operation for 30 s and 210 s plasma pulse lengths. With improved vacuum variable capacitor design, it appears possible to couple 2 MW per loop over the frequency range 35 to 80 MHz.

2.2.3 High-Current Capacitors

A key to the success of compact loop antennas in coupling several megawatts of power to a fusion plasma is the development of a vacuum variable capacitor with higher current capacity (~ 1 kA) than commercially available units. In addition, the internal inductance of the capacitor must be minimized to allow operation at frequencies up to 80 MHz and possibly higher. An improved design was devised based on postmortem analysis of an existing design. The capacitor industry was approached with this new application, with the results that both Comet (Bern, Switzerland) and Jennings (San Jose, California, USA) have agreed to build prototype units which will be tested at full parameters on a test stand at ORNL during 1986.
2.2.4 Folded Waveguides

A new waveguide configuration, known as the folded waveguide [5], is a promising candidate for ICRF heating on large, high-field machines where second harmonic heating occurs at frequencies on the order of 100 MHz. The folded waveguide can easily be configured to conform to existing port sizes, which are usually larger in the poloidal dimension than in the toroidal dimension for most tokamaks. An advantage of the folded waveguide over ridged waveguide designs is that the rf electric fields at the mouth of the guide are near a minimum.

The folded waveguide can be envisioned as a rectangular waveguide several times wider than its height, which is folded several times like an accordion. The cutoff frequency can be made arbitrarily low for a given envelope by increasing the number of folds. The magnetic field at the mouth of the guide reverses at each fold, so the guide is fitted with an aperture plate to permit only the desired polarization to couple to the plasma.

2.3 Plasma Fueling

The plasma fueling R&D activity led by ORNL is directed at developing one or more advanced plasma fueling systems for experimental magnetic confinement devices and future fusion power reactors. The primary approach taken is that of producing and accelerating frozen hydrogenic pellets in the speed range of 2000 m/s and higher.

Two specific concepts are under development: 1) high speed pneumatic acceleration and 2) mechanical (centrifugal) acceleration. Both approaches are being pursued to meet the projected pellet size and delivery rates for major near-term plasma confinement devices, such as TFTR, Tore Supra, JET, and DIII-D, as well as future applications. In the past two years, the state-of-the-art of pneumatic pellet injectors has been advanced through the development at ORNL of the Repeating Pneumatic Injector (RPI). This versatile machine gun-like device attains maximum pellet speeds of 1900 m/s with 4 mm hydrogen pellets and operates at a repetition rate of 6 Hz. In 1985, the RPI was installed on the Tokamak Fusion Test Reactor (TFTR) and has produced the highest values for plasma density \( n_e(0) = 4 \times 10^{14} \text{ cm}^{-3} \) and Lawson product \( > 1 \times 10^{14} \text{ cm}^{-3} \text{s} \) recorded on TFTR. In April 1986, the RPI was replaced by a new eight-shot pneumatic device developed at ORNL that features variable pellet size capability at a design speed of 1500 m/s. The so-called Deuterium Pellet Injector (DPI), shown in Figure 3, will be used in a collaboration between ORNL and the Princeton Plasma Physics Laboratory to create and sustain high density target plasmas for the high power (22 MW) neutral beam heating phase on TFTR in the 1986-87 time frame.

As a result of the successful outcome of the RPI experi-
ments on TFTR in 1985, ORNL was directed by DOE to develop a more versatile injector based on the RPI design, to be used on the Joint European Torus (JET) tokamak in Culham, England, in 1987 as a part of a major collaboration between the U.S. and the European community on plasma fueling and transport physics in the high plasma density regime. These experiments will coincide with, and are expected to contribute to, the JET program to achieve energy breakeven conditions at reactor relevant plasma parameters.

In the area of centrifugal pellet injector development, ORNL is proceeding with an upgrade of the prototype device that was used in the first steady-state pellet fueling experiment performed on the GA Technologies, Inc., D-III tokamak in 1984. The upgrade is designed to achieve pellet speeds in the 1200 m/s range, pellet size in excess of 2 mm, and long pulse to steady-state capability at 10-20 Hz by employing a novel pellet fabrication technique. The new device is a prototype of the pellet injector system that is to be featured in a collaboration with the Commissariat a L'Energie Atomique (CEA) on the Tore Supra tokamak under construction at Cadarache, France. The objective of the ORNL-CEA collaboration in this area will be to study long pulse reactor relevant tokamak discharges with simultaneous plasma fueling and exhaust capabilities.

In addition to these confinement physics related activities, ORNL is pursuing advanced technologies to achieve pellet velocities significantly in excess of the 2 km/s range attained on the RPI, and has embarked on a development program designed to explore the feasibility of fabricating and accelerating tritium pellets in the velocity range of 1-2 km/s. These new initiatives will be directed toward providing a central fueling option on the compact ignition tokamak (CIT).

2.4 Plasma-Interactive Components (PIC)

The high fluxes of plasma (hydrogen isotope and impurity) particles and of heat to components such as limiters, divertor plates, RF antennas and launchers, as well as to the first wall, lead to effects which place considerable demands on materials. In turn, conditions at the plasma edge can strongly influence the properties of the plasma core. For this reason, plasma-interactive materials are developed with two viewpoints in mind -- engineering design and component performance on the one hand, and plasma performance on the other.

The key issues in this area can be grouped into the categories of particle and impurity control, heat removal, and off-normal events. Activities in each of these depend strongly on materials characterization and development, and are also tied closely to experimentation in actual confinement devices.

Basic plasma-materials interaction/high heat flux (PMI/-HHF) problems can be addressed and materials and components can
be developed before a fusion neutron test facility is available. Materials research and the development and testing of components are done with non-neutron facilities, including both laboratory and confinement devices. Eventually, however, the separate and partially integrated test facilities for PMI/HHF should lead to work done in a steady-state or high duty-cycle hydrogen confinement experiment to assure a reliable technology in this area.

The U.S. program in PMI/HHF, based on the above rationale, is carried out in a variety of laboratory facilities. The efforts are divided into two categories depending on whether or not they involve or require a confinement device such as a tokamak. There was, in the past, one tokamak, the Impurity Studies Experiment (ISX-A and B) at ORNL, which was intended to emphasize PMI issues. At present, PMI/HHF experiments are done on all existing U.S. confinement machines, even though they are not the principal activity of those machines. Several international collaborations, of which the U.S. is a partner, are carried out on confinement machines outside the U.S., which have strong PMI/HHF programs. These include TEXTOR (KEK Juelich), Tore Supra (CEA, Cadarache) ASDEX Upgrade (MPI Garching) and JET (CEC, Culham).

There are three primary laboratory facilities in the U.S. which concentrate on PMI/HHF issues. These are the Plasma Materials Test Facility (PMTF) at Sandia National Laboratories, Albuquerque (SNLA); the Plasma-Surface Interactions Experimental Facility (PISCES) at the University of California, Los Angeles, (UCLA); and the Tritium Plasma Experiment (TPX) at Sandia National Laboratories, Livermore (SNLL). These facilities are described in Sections 2.4.1 to 2.4.3.

In addition, the U.S. program has a significant effort in determining basic PMI property data using accelerator facilities at Argonne National Laboratory (ANL), ORNL, SNLL, and SNLA. These facilities are also used to analyze surfaces after exposure to plasmas by a variety of nuclear reaction and surface analysis techniques. The program also includes the development and use of solid state diagnostic probes in all the principal confinement devices. Other facilities which have been used for significant research on PMI are the Surface-Plasma Research Facility (SPRF) and the Traveling Surface Analysis Station (TSAS) at ORNL. The emphasis here has been on fundamental recycling phenomena and on surface cleaning and conditioning. The FELIX facility at ANL is designed to carry out electromagnetic tests on components. It can be used to simulate the electromagnetic characteristics of current disruptions in plasma-interactive components such as limiters. Examples of facilities and research activities are given below for PMTF, PISCES and TPX.
2.4.1 Plasma Materials Test Facility (PMTF)

The PMTF at SNLA is dedicated to the development of high heat flux components. The PMTF consists of two facilities: a) an electron beam test system (EBTS), and b) a multiple beam test system (MBTS). The EBTS is utilized for materials testing, thermal disruption simulation and steady-state heat removal development. The MBTS utilized ion beams and is a dedicated materials and high heat flux system. Selected parameters of PMTF are shown in Table III.

PMTF has been utilized to perform key tests in support of a number of worldwide devices and programs. Examples of recent activities include: 1) thermal fatigue tests of a prototype beryllium limiter for JET, 2) a series of high heat flux tests on a mockup module of the wall protection scheme proposed for the Tore Supra tokamak, and 3) high heat flux tests on divertor targets for the ASDEX Upgrade.

2.4.2 Plasma Surface Interaction Laboratory (PISCES)

PISCES [6] is a continuously operating plasma device at UCLA, which achieves hydrogen plasma densities of $10^{11} - 10^{13}$ cm$^{-3}$ and electron temperatures of 5-24 eV. Particle fluxes of $10^{17}$ cm$^{-2}$ s$^{-1}$ and fluences of up to $10^{23}$ cm$^{-2}$ have been used to bombard controlled, negatively biased samples inserted into the plasma. The thermal ion temperature is a few eV. Target fluxes of over 1 A/cm$^2$ are produced over an area of 50-80 cm$^2$. The parameters of PISCES are summarized in Table IV and a schematic of the machine is shown in Figure 4.

The degree of plasma interaction with the sputtered target atoms can be controlled by selecting the electron temperature and density of the plasma. At low electron temperatures high rate sputtering of the surface occurs. Alternatively, a combination of high density and electron temperature ionizes the sputtered target atoms, and the confining magnetic field and potential distribution return them to the target. Erosion, redeposition and mixing of materials, representative of the behavior in confinement devices, can therefore be studied under controlled conditions.

Candidate fusion materials have been studied in PISCES [7]. Studies of the performance of graphite produced some surprising and important results. Chemical sputtering of graphite at low temperature (< 100 °C) by hydrogen ions with less than 100 eV of energy was first observed in PISCES. The sputtering yield depends strongly on the temperature of the graphite with the peak occurring at 600 °C, as shown in Figure 5. Chemical sputtering occurs at high bombarding fluxes, with the increase in the yield agreeing with much lower flux, ion-beam experiments. Redeposition of hydrocarbons reduces the effective sputtering yield of graphite, and produces surfaces with weakly bound soot. Metal impurities in the plasma, which
are deposited on graphite samples, form clusters, similar to the results from analysis of tokamak limiters. Stainless steel surfaces have been modified by plasma bombardment in PISCES. The surface structure of the stainless steel was found to be determined by the temperature of the stainless steel and by the deposition of foreign atoms such as molybdenum on the surface.

Future work calls for a balance of experiments between materials-oriented plasma interaction experiments and edge physics experiments important to the understanding of pump limiter and divertor physics. In addition, new concepts for edge plasma control and component design will be studied. A second PISCES machine with improved surface diagnostics and material handling capabilities is under construction. This second machine will allow further and more systematic investigation of candidate materials while emphasis in the original PISCES device will be to study the physics problems related to in-vessel components such as pump limiters and divertors.

2.4.3 Tritium Plasma Experiment (TPX) Facility

The TPX facility at the Tritium Research Laboratory of SNL is the only operating tritium plasma apparatus devoted to plasma-materials interaction research. A schematic of the facility is shown in Figure 6. Since 1982, when TPX became operational, tritium plasma has been used exclusively for the study of plasma-materials interactions, and plays a lead role in the area of tritium retention and permeation studies in first wall and limiter materials. Magnetic confinement is used to contain the TPX plasma, and heating is provided by a 200 watt rf generator operating in the electron-cyclotron regime at 400 MHz. Ion densities on the order of \(10^{11}\) ions/cm\(^2\) are attainable with average ion energies of approximately 7 eV. By applying a negative bias to the experimental samples, particle energies between 15 and 300 eV are attainable with tritium fluxes on the order of \(10^{17}\) ions/cm\(^2\)-s.

Extensive diagnostic equipment is available. Two quadrupole mass spectrometers allow measurements of the gasses released from the front and back of plasma-exposed samples. An in-situ Auger electron spectrometer permits the in-situ examination of the elements remaining on the sample surface. Plasma diagnostics such as Langmuir probes and EXB analyzers complete this unique diagnostic package.

Because TPX does operate with tritium, many experiments can be performed that would be impossible with hydrogen or deuterium. Tritium, being a radioactive isotope, is easily detectable at very low concentrations. TPX has been used to measure the plasma-driven recycling and permeation of hydrogen isotopes through fusion reactor materials, to measure plasma erosion in materials exposed to high fluences, and to load tritium into samples for tritium injection studies in PLT. Presently, it is being used to measure the retention of tritium
in graphite at temperatures that simulate TFTR operation. These results are modeled with extensive computer calculations for tritium retention behavior (see Figure 7). Information from this study will lead to a decision whether the graphite limiters in TFTR are acceptable or must be coated with a low-Z ceramic to limit the tritium inventory during D-T operation.

3. NUCLEAR TECHNOLOGY AND MATERIALS

Nuclear technology and materials are two of the four MFPP key technical issues. Both issues deal with key aspects of components and technical disciplines concerned with fuel generation and processing, energy extraction and conversion, and radiation protection of personnel and components.

Nuclear technology includes four elements: 1) blanket/first wall, 2) radiation shield, 3) tritium processing, and 4) nuclear elements of plasma-interactive components. The fusion environment experienced by the nuclear components involves the simultaneous presence of energetic plasma particles, 14 MeV neutrons, magnetic fields, surface and bulk heating, tritium, and vacuum. In addition, many fusion nuclear components are required to perform multiple functions (e.g., simultaneous tritium production/extraction and energy conversion/extraction in the blanket). Therefore, fusion nuclear technology presents unique and challenging requirements. The U.S. program on fusion nuclear technology includes a number of R&D activities. The FINESSE study, led by UCLA, is identifying the role, timing and characteristics of major experiments and facilities. The blanket technology program, led by ANL, is addressing a number of key issues for liquid and solid breeder blankets. The tritium processing program, led by Los Alamos National Laboratory (LANL), is focused on important performance and safety issues of tritium handling subsystems.

The materials program consists of three elements: structural materials, non-structural blanket materials, and special materials for other applications such as insulators and magnet materials. The primary focus of the materials program is the development of materials which provide high performance characteristics and long lifetime in the unique high energy neutron environment of a DT-fueled fusion reactor. Present activities include: a) developing unirradiated baseline properties for candidate materials; b) performing irradiation experiments in fission reactors; c) development of theoretical models to permit extrapolation of irradiation data to the fusion environment; and d) studying the options for obtaining irradiation data in fusion-relevant environments. The U.S. materials R&D activities are carried out at ORNL, Hanford Engineering Development Laboratory (HEDL), ANL and other organizations.

Examples of present activities on nuclear technology and materials are given in this section.
3.1 FINESSE

FINESSE is a study of the issues, phenomena and experimental facilities for fusion nuclear technology. The objectives of the study are to: a) understand and characterize issues; b) develop scientific basis for engineering scaling and experiment planning; and c) identify characteristics, role and timing of major facilities required.

The study is led by UCLA and involves major organizations from the U.S.: ANL, HEDL, SNLA, LANL, Grumman, TRW, EG&G, MDAC, and other organizations. FINESSE has also benefited from the productive participation of a number of scientists and engineers from Canada, Japan and Europe.

The results of the study to date have been documented in References 8 and 9. A summary of the fusion nuclear technology testing requirements defined in FINESSSE is presented in another paper at this conference (see Ref. 10).

3.2 Liquid Breeder Facilities

One of the key areas addressed by the blanket technology program is R&D for liquid breeder blankets. This R&D includes experiments on MHD effects on self-cooled liquid metal blankets, corrosion of structural materials by liquid metals and tritium recovery from liquid metals and molten salts. Examples of activities in two key areas are given below.

3.2.1 Liquid Metal MHD Testing

A key facility for studying MHD effects is the ALEX (Argonne Liquid Metal Experiment) facility, which became operational in 1985. The facility is shown schematically in Figure 8. The sodium-potassium eutectic, NaK, is used as the working fluid since it is a liquid at room temperature and its electrical characteristics are similar to liquid lithium. The NaK is pumped through a 2.0 T magnet (bore dimensions - 1.5 m x 0.8 m x 0.15 m) where the velocity flow profiles are measured. The facility is designed to allow measurements to be made for Hartmann numbers and interaction parameters up to $10^4$. A key goal is to determine velocity profiles through high magnetic fields in increasingly complex geometries. These results can then be used to compare with model predictions.

The first series of experiments centered around 3-D MHD effects in a circular, thin conducting wall duct located at the inlet or exit of a strong transverse magnetic field. Early experimental results on pressure and flow distributions show good agreement with transverse pressure gradients and high velocity wall jets predicted by analysis for such fringing fields. An example of the velocity flow profile across the diameter of the duct is shown in Figure 9. There are jets formed near the walls with peak velocities 3 to 4 times the
average velocity. In addition, there is a stagnant region near the center of the duct. Details of these tests are to be given in Reference 11.

Future plans include testing of more complex geometries such as bends and ducts with varying cross sections. Upgrades of the ALEX facility to increase the magnetic field strength and test volume, and to conduct heat transfer tests are under consideration.

Progress has also been made in the area of model development. In addition to predictions for the circular duct, analytical solutions have been obtained for flow in a rectangular duct in a non-uniform magnetic field, a thin-walled elbow in a plane perpendicular to a uniform field, and for a duct whose cross section changes from rectangular to trapezoidal (i.e., flow tailoring).

3.2.2 Corrosion Experiments

The objective of this work is to develop a data base in the critical areas of corrosion, compatibility, and the influence of chemical environments on the mechanical properties of structural alloys. Programs in these areas are conducted primarily at ANL and ORNL. The current effort includes both lithium and 17Li-83Pb alloys with austenitic steels, ferritic steels and vanadium alloys.

The ANL facilities include forced-circulation lithium and LiPb loops in which both corrosion/mass transfer experiments and in-situ mechanical testing are conducted. The current effort is focused primarily on effects of temperature, system temperature difference, and impurity concentration in the liquid metal. Effects of oxygen concentration in recirculating pressurized water autoclaves on the corrosion and stress-corrosion of vanadium alloys are also being investigated.

Corrosion studies at ORNL are conducted in static capsule tests and thermal convection loops with liquid metals (Li and LiPb) and molten salts. Effects of hydrogen pressure on corrosion of vanadium alloys are being investigated in static pressurized water autoclaves.

Results on temperature dependence of the corrosion rates of HT-9 ferritic steel and Type 316 stainless steel in lithium and LiPb have been obtained in the range of 350–550 °C. Average corrosion rates in high purity liquid metals at ~450 °C are given in Table V.

Figure 10 shows the temperature dependence of the corrosion rates of HT-9 and Fe-9Cr-1Mo ferritic steels in lithium containing ~100 ppm nitrogen [12]. Corrosion rates of V-15Cr-5Ti alloy in pressurized water at 288 °C vary from 2–7 mg m⁻²h⁻¹ depending on the oxygen content of the water [14].
Future work on liquid metal corrosion will be focused on definition of impurity effects and the velocity and velocity profile dependences.

3.3 Solid Breeder Materials

A principal objective of the solid breeder materials program is to develop a materials properties and irradiation behavior data base that can be used in selection of a prime candidate breeder material for use in fusion reactors. Currently, research focuses on the following materials: Li₂O, γ-LiAlO₂, Li₂SiO₃, Li₄SiO₄, and Li₂ZrO₃. Laboratory studies to measure the thermochemical, thermophysical, and mechanical properties for each material are in progress. Complementary experiments testing the response of each material to a neutron environment are also required. Such irradiation experiments generally are of two types: closed capsule tests for lifetime evaluation, and open capsule tests for evaluation of in situ tritium recovery.

Laboratory studies have shown that the solubility of H₂O(T₂O) in Li₂O is very low, but increases with increasing temperature. From a thermodynamic perspective, these solubility data exhibit positive deviations from ideality, meaning that the partial pressure of T₂O above Li₂O is much greater than expected (tritium recovery is very easy). The oxygen activity determines the thermochemical characteristics of the ceramic breeder and the chemical form of the release tritium species. Such thermodynamic data have been used to predict the thermochemical performance of candidate breeder materials.

The solid breeder blanket may be in sintered block form or in sphere-pac form. Detailed analysis of the thermal conductivity of the sphere-pac form has shown its attractiveness with regard to thermal and thermomechanical performance.

The in-reactor experiments will serve to give information on the response of the breeder material to the neutron environment and ease of tritium recovery. Year-long irradiation experiments (FUBR-1A) are affirming the attractiveness of the ceramic materials because of their chemical stability, resistance to radiation damage, and low tritium retention. In-reactor swelling of Li₂O may be the most undesirable property of Li₂O in comparison to the other ceramics. Since swelling is thought to be caused by defect and helium production, it is likely that swelling will be proportional to burnup and increase at high burnup levels. In-situ tritium recovery experiments (TRIO) have affirmed low tritium solubility, indicating that blanket tritium inventory is likely to be well below design guidelines.

Recently, greater emphasis has been placed on running fully instrumented in situ tritium recovery tests on large volumes of candidate breeder materials to evaluate their test performance. The mechanism of tritium transport and release is
complex, involving bulk diffusion at lower temperatures and surface desorption at higher temperatures. Thus, complementary laboratory experiments on both irradiated and unirradiated materials are being undertaken to more rigorously define the tritium transport and release mechanism. Such data are needed if one is to be able to set rigorous breeder blanket operating temperatures. Further, the in-reactor experiments are likely to impose thermal gradients which may produce changes in the solid that could severely impede tritium release. In-reactor experiments should more clearly define the relation between the thermal behavior of a solid and ease of tritium recovery.

In response to the detailed technical knowledge required to understand the operation of a ceramic breeder blanket, an international collaboration has been established for the timely development of solid breeder materials. Under the auspices of Annex II to the International Energy Agency Implementing Agreement for a Program of Research and Development on Radiation Damage in Fusion Materials, a program consisting of the exchange of materials and shared irradiation testing has been implemented. This exchange program has been named BEATRIX (Breeder Materials Experimental Matrix) and participants include Canada, the European Community, Japan, and the United States. Contributing laboratories include CEA Saclay, KFK Karlsruhe, Tokai/JAERI, AERE/Springfield, CREE/Casaccia, and Westinghouse Hanford. This experiment will allow comparison of preparation/fabrication methods, irradiation techniques, and tritium extraction methods. Materials prepared by one partner will be irradiated in another partner's reactor. Capsule and purge flow experiments in mixed spectrum reactors (HFR, OSIRIS, SILOE, and NRU) and capsule experiments in a hard spectrum reactor (EBR-II) will be carried out.

The static experiments include:

FUBR-1B/BEATRIX USA
EXOTIC Netherlands
ALICE France
DELICE Germany
CREATE Canada

The purge-flow experiments include:

LILA (γ-LiAlO₂) France
LISA (Li₂SiO₃ - Li₂SiO₄) Germany
VOM-22 (Li₂O - γ-LiAlO₂) Japan
CRITIC (Li₂O) Canada

Probably the most detailed experiment in the sense of coverage of materials is BEATRIX/FUBR-1B because it involves five different materials and three different configurations, supplied by six different partners. This experiment is also the only long-term (2 yrs.), hard-spectrum irradiation experiment of the above grouping. The details of this experiment are given in Table VI.
An additional feature to this collaboration is the assignment of American staff to two laboratories, JAERI/Tokai-Mura and KFK/Karlsruhe, and a strong collaboration between the USA and Canada in developing the CRITIC experiment.

3.4 Fusion Neutronics

Examples of research efforts in Neutronics can be given in two areas that are strongly related. Efforts in the first area concentrate on examining the required conditions to sustain fuel self-sufficiency in fusion reactors operated on a D-T fuel cycle. In-depth and detailed engineering analyses have been performed [15-17] on various blanket and reactor concepts to verify the potential of each blanket concept to exhibit a tritium breeding ratio (TBR) in excess of unity by a margin that compensates for losses, radioactive decay and other inventory requirements. Efforts in the second area [18] concentrate on evaluating the overall uncertainties (both experimental and analytical) associated with the TBR.

A key element of the U.S. neutronics activities is a collaborative program [19] with the Japanese Atomic Energy Research Institute (JAERI) to perform integral experiments at the Fusion Neutron Source (FNS) facility at JAERI. The U.S. organizations contributing to this effort are UCLA, ANL and ORNL. Comparison between experimental results and analytical predictions for tritium breeding carried out in this collaborative effort will provide, among other information, data needed for evaluating the overall uncertainty in predicting the TBR in primary candidates for fusion blankets and means to resolve the discrepancies found between experimental and analytical results. Other objectives of this collaborative program are to provide experimental data: a) to determine the accuracy, guide the development, and establish the validity of the computational methods and nuclear data base; b) to assist in the selection of materials and the configuration of candidate blanket concepts from the neutronics viewpoint; and c) to improve and advance present measuring techniques and detectors, and reduce experimental uncertainties.

The U.S.-Japan collaborative program consists of two phases. Phase I experiments are focused on simple, but design-oriented engineering benchmark experiments to obtain experimental data on blanket characteristics (e.g., tritium production rate, neutron spectra, etc.) of the Li$_2$O breeder and compare it with the calculational values. Phase II experiments are planned such that the chosen system specifications (geometrical arrangement, materials selection, etc.) give closer simulation of the fusion environment (spectra).

Integral experiments considered in Phase I are: a) a D-T neutron source characterization experiment; b) tritium production rate (TPR) measurements in a reference Li$_2$O assembly; c) first wall experiments with and without coolant simulation; and
d) beryllium neutron multiplier experiments in various configurations. Some of the results from the Phase I program are discussed below.

Characterization of the source neutrons from the rotating target was performed through spectrum measurements using the time of flight method and NE213 spectrometer [19-20]. The U.S. results for flux mapping have shown, for example, that the discrepancy between experimental and analytical values for the $^{27}$Al($n$,α) reaction rate is within 4% in the horizontal direction and within 3% in the vertical direction [21].

The ratio of calculated to experimental values, C/E, for TPR from $^6$Li (T$_6$) along the central axis of the Li$_2$O reference assembly show a large deviation from unity at locations near the front surface. Values found at these locations were as large as C/E ≈ 2, even after correcting for self-shielding effect resulting from flux depression in the detectors used to measure T$_6$ (see Fig. 11 and Ref. 21). Measurements were performed by the Lithium-Glass scintillator method. The TPR from $^7$Li (T$_7$) was also measured along the central axis in all the experiments performed. In the reference system, values calculated by the U.S. are within 1-3% of the experimental results measured by the Li-metal method and the NE213 indirect method.

In the first wall experiments, a 0.5 and 1.5 cm-thick SUS316 first wall was placed in front of the Li$_2$O assembly with and without a 0.5 cm-thick polyethylene (PE) layer (used to simulate H$_2$O coolant), and four experiments were performed for tritium production measurements. It was observed by both the U.S. and JAERI that the agreement between calculations and measurements for T$_6$ is, in general, better in the cases considered than in the reference case, with a discrepancy of 1-15% in T$_6$ and, again, the largest discrepancies occur at locations adjacent to the first wall. A larger discrepancy, however, was observed in the U.S. calculations for T$_7$ in the 1.5 cm SS + 0.5 cm PE experiment where the discrepancies are within ~10% (1-3% in the reference experiment).

In the beryllium experiments, 5 cm Be, 10 cm Be, and 5 cm Li$_2$O + 5 cm Be (beryllium-sandwiched system) layers were placed in front of the Li$_2$O assembly, and T$_6$ and T$_7$ measurements were performed in the three experiments. Spectral indices reaction rate measurements in the beryllium-sandwiched system were also performed. It was observed that better agreement with the experimental T$_6$ and T$_7$ values was obtained when the Be ($n$,2n) cross section from the latest LANL evaluation (rather than from the current ENDF/B-V data) was used to obtain calculated T$_6$ and T$_7$ values.

Phase II experiments are planned to be performed during FY86-87. A closed-geometry is chosen for these experiments where a Li$_2$CO$_3$ container surrounds the source with a Li$_2$O test assembly placed at one end of the geometry. The objective of
Phase II experiments is to investigate the range of the uncertainty in tritium breeding in a test module which simulates a real fusion blanket (first wall/breeder/coolant) and placed in a closely-simulated fusion environment. The study of the impact of the presence of penetrations in the \( \text{Li}_2\text{CO}_3 \) container on the TPR in these experiments is also planned.

### 3.5 Tritium

Two of the key tritium issues are related to tritium processing from the plasma exhaust and tritium permeation. The key experiments in the tritium processing area are discussed below.

#### 3.5.1 Tritium Processing

The major U.S. fusion tritium facility is the Tritium Systems Test Assembly (TSTA) at the LANL Laboratory. The principal goals of TSTA are to--

- demonstrate the fuel cycle for fusion power systems;
- develop and evaluate advanced safety systems for personnel and environmental protection;
- develop, test, and qualify equipment for tritium service in the fusion program;
- develop tritium-compatible components with long-term reliability;
- demonstrate long-term safe handling of tritium without hazardous incidents; and
- investigate and evaluate the response of the fuel handling and environmental systems to normal, off-normal, and emergency situations.

The TSTA process flow loop consists of components typically proposed for fusion reactors. Because fuel will not actually be burned, impurities and helium "ash" are added to the deuterium and tritium to simulate the fusion reaction process. The components of the flow loop are designed full-sized for a 1.5-gigawatt (electric) commercial power plant. The processing steps, all computer controlled, are as follows (see Figure 12).

1. A mixture of deuterium and tritium gas is injected into the vacuum system (VAC).
2. Impurities, such as water, methane, and ammonia, are added.
3. The VAC and the transfer pumping system (TPU) are used to empty the vacuum chamber (torus mockup) and to transfer the gas to the fuel cleanup system (FCU).
4. The FCU is used to remove all impurities other than helium. The impurities are decomposed and hydrogen isotopes are recovered for recycling.
5. The purified fuel stream is sent to the isotope separation system (ISS), where the hydrogen isotopes are fractionally distilled to produce streams of pure deuterium, tritium, a deuterium/tritium mixture, and a waste stream of hydrogen.
6. The fuel components are then recombined for reinjection into the VAC.
Research into tritium contamination is conducted in the experimental contamination studies laboratory (XCS), which is tied to all the support systems, although independent from the TSTA process loop itself.

A series of major, successful operations of the TSTA process flow loop in January 1986 yielded progress in a variety of technical areas. Highlights included:

- The tritium inventory was increased to 31 g.
- Extensive data were obtained on the performance of the isotope separation system (cryogenic distillation columns). Data included axial composition profiles within a column.
- The fuel cleanup system was integrated into loop operation with the distillation system (without added impurities). Dynamic interactions were satisfactory.
- Practical operation of the uranium tritide storage beds was demonstrated.
- Safe, efficient maintenance techniques were demonstrated on tritium-contaminated systems. Demonstrated capabilities included component change-out and rewelding without personnel exposures or environmental releases.

In addition to integrated loop operations, recent progress at TSTA has included evaluations with tritium of a variety of potential alternative system components. Among these are:

- a palladium-alloy diffuser for gas purification;
- a ceramic electrolysis cell for water decomposition; and
- a metallic getter for glovebox atmosphere detritiation.

Each of these components has performed satisfactorily in the tritium environment appropriate to its system function.

3.6 Modeling of Radiation Effects

The U.S. materials program has been placing increased emphasis on theory and modeling in order to provide better extrapolation of experimental data from non-fusion irradiation sources to future fusion reactor conditions. One example of such modeling activities is given below.

During the past two decades, numerous phenomenological and deterministic treatments of radiation effects have been performed. Most notable of these theories is the theory of void growth and irradiation creep. The success of this approach has fallen short of a priori predictions of material behavior. This casts some doubts on the ability to extrapolate radiation effects from current radiation sources to anticipated fusion conditions. Recently, however, work has begun on the development of a rich theoretical approach that is based upon nonequilibrium statistical mechanics [22]. This approach starts from stochastic differential equations, or equivalent distribution equations, for the evolving microstructure and yields a more detailed physical description of atomic clustering pro-
cesses. With the advent of supercomputer technology, other tools of theoretical analysis are gaining importance. Microscopic treatments, starting from the basic equations of motion of atomic species, will soon be feasible. Molecular dynamics, Langevin dynamics, and Monte Carlo methods are examples of work in this direction. Figure 13 shows the stages for treatment of a stochastic system that has many degrees of freedom.

4. SYSTEM STUDIES

The present U.S. system studies effort can be broadly classified into four areas:

A. Ignition Studies. These studies are aimed at selecting a reference concept for a tokamak device that achieves ignition.

B. Engineering Test Reactor. Preliminary scoping studies have begun to identify the key technical features of a technology testing and long burn facility. An initial conceptual design, called TIBER [23], has been developed by LLNL.

C. INTOR. The INTOR studies have continued, with emphasis shifting to identifying and evolving innovative ideas for concept improvement.

D. Reactor Studies. The objective of these studies has been the development of conceptual designs for commercial fusion reactors. A conceptual design of a small-size tandem mirror reactor with improved safety features, called MINI-MARS, has been completed. Studies to improve the tokamak power system are now in progress. A new, comprehensive study to develop a compact Reversed Field Pinch (RFP) conceptual reactor design has begun.

A number of papers presented in this conference are reporting on progress in the U.S. effort on system studies. Therefore, the remainder of this section is devoted to a summary of the present design for an ignition tokamak.

4.1 Ignition Experiment

Design studies for the next major physics experiment that achieves ignition have now focused on a compact ignition torus (CIT). The primary objective of the CIT design activity is to satisfy the physics requirements for ignition, pulse length and plasma energy removal (divertor/limiter), with the minimum major radius consistent with prudent engineering choices. Table VII shows selected parameters of CIT. Achievement of small radius is facilitated by the requirements for pulse length and number of full-power pulses, which have been relaxed relative to previously studied fusion engineering test facilities. The relatively short TF flat-top (12 times estimated energy confinement time) can be accommodated by a magnet design in which temperatures rise adiabatically during the pulse. The design can accommodate doubled pulse lengths at 70% of full
field. At that reduced field, the number of allowed pulses is limited in a practical sense by the operating span of the machine, which is less than ten years. In order to achieve the minimum major radius, it has been necessary to utilize liquid nitrogen pre-cooling of the magnet in a manner similar to the Alcator-C and the Frascati FT high-field compact tokamaks now in operation. The need to re-cool between pulses, coupled with the short TF flat-top pulse, results in two operational changes relative to the majority of present day tokamaks; namely, a restriction to one full parameter pulse per hour and the requirement to ramp up the plasma current concurrently with the toroidal field.

The CIT mechanical configuration, as illustrated in Figure 14, is based on the following structural approaches, chosen from a number of alternatives studied during the first phase of the conceptual design: 1) use of an external frame and hydraulic press to apply sufficient pre-load to the TF coil to support the vertical leg of the TF coil; 2) use of the wedging in the inner TF legs to support the inboard load on the TF coils; 3) use of a gap between the TF coil inner legs and the central ohmic heating (OH) coil to assure structural independence of the two systems; 4) use of copper-steel laminated conductors in the TF and OH coils in order to optimize the conductivity/strength ratio of the conductors; 5) use of an average stress criterion where the average TF stress is allowed to approach 0.85 times the yield strength of the composite material, in accordance with the guidelines for allowable stress developed for externally supported structures having to perform through a limited number of cycles; and 6) use of partial coil cases with significant horizontal access between every other coil to provide for overall support against out-of-plane loads.

The design accommodates operation with either a limiter or a divertor by using an external poloidal field (PF) system and supplementary internal control coils. The vacuum chamber allows operation with a limiter at full current (10 MA) and with a divertor at about 90% current. Internal hardware required for each type of operation will be in the vessel at all times. Tiles covering the exposed portions of the chamber will be designed for maximum survivability to disruptions, but the lifetime is assumed to be limited. Therefore, replacing the tiles by remote maintenance must be possible in a reasonable time. The design philosophy for remote maintenance external to the vacuum vessel (ex-vessel) is to provide maximum close-in shielding with the goal of maximizing the hands-on access to the external diagnostic and rf heating systems. Ex-vessel remote maintenance will be provided for areas within the shield penetrations, and as a backup for areas of the test cell to which hands-on maintenance proves impractical. Repair of the major core components of the machine external to the vessel and internal to the shield (TF and PF coils, structure, cryostat, etc.) will be with hands-on during the hydrogen phase, but will not be included in the planned remote maintenance strategy and
cost. This philosophy is based on a review of failure modes of existing tokamaks and is consistent with minimizing the cost of the project.

REFERENCES


OVERVIEW OF THE U.S. FUSION TECHNOLOGY PROGRAM

FIGURE CAPTIONS

Figure 1. IFSMTF experimental set-up
Figure 2. Primary magnet system for MFTF-B (trim coil deleted for clarity)
Figure 3. Deuterium Pellet Injector Facility for TFTR
Figure 4. Schematic of FISGES plasma device
Figure 5. Chemical sputtering yield of graphite as a function of temperature
Figure 6. Schematic diagram of Tritium Plasma Experiment. AES = Auger electron spectrometer, QMA = quadrupole mass analyzer, TMP = turbomolecular pump, MFE = molecular flow element, CM = capacitance manometer
Figure 7. Temperature dependence of tritium retention in graphite
Figure 8. Schema of ALEX facility
Figure 9. Velocity profiles taken at exit of magnet
Figure 10. Arrhenius plot of dissolution rate data for ferritic steels exposed to flowing lithium
Figure 11. Calculated to experimental values, C/E, for T6 along the central axis in the Li2O reference experiment (results from JAERI-U.S. collaboration program)
Figure 12. TSTA main processing steps
Figure 13. Stages for treatment of a stochastic system with many degrees of freedom
Figure 14. Elevation view of CIT
Figure 1. IFSMTF experimental set-up
Coils
12 solenoid coils (5m)
4 transition coils
4 axicell coils (NbTi)
2 axicell insert coils (Nb<sub>3</sub>Sn)
2 Yin-Yang coil pairs

Designation
S1 through S6 (east & west)
T1 and T2 (east & west)
A1 & A2 (east & west)
A2I Inner (east & west)
M1 & M2 (east & west)

Figure 2. Primary magnet system for MFTF-B (trim coil deleted for clarity)
Fig. 3
Deuterium Pellet Injector Facility for TFTR
Figure 4. Schematic of PISCES plasma device
Figure 5. Chemical sputtering yield of graphite as a function of temperature.
Figure 6. Schematic diagram of Tritium Plasma Experiment. AES = Auger electron spectrometer, QMA = quadrupole mass analyzer, TMP = turbomolecular pump, MFE = molecular flow element, CM = capacitance manometer.
Figure 7. Temperature dependence of tritium retention in graphite
FACILITY PARAMETERS

- FLOW RATE: 5-300 GPM (.3-19 l/s)
- MAGNETIC FIELD: 0-2.5 T
- PRESSURE: 0-150 psi (0-1 MPa)
- TEMPERATURE: 20-40°C
- WORKING FLUID: NaK
- MAGNET GAP: 6 INCHES (15 cm)
- TRAVERSING MAGNET: ±50 INCHES (±125 cm)

Figure 8. Schema of ALEX facility
Figure 9. Velocity profiles taken at exit of magnet
Figure 10. Arrhenius plot of dissolution rate data for ferritic steels exposed to flowing lithium.
Figure 11. Calculated to experimental values, C/E, for T$_6$ along the central axis in the Li$_2$O reference experiment (results from JAERI-U.S. collaboration program)
Figure 12. TSTA main processing steps
Figure 13. Stages for treatment of a stochastic system with many degrees of freedom
<table>
<thead>
<tr>
<th>Conductors Material</th>
<th>GD/Convair</th>
<th>General Electric</th>
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<th>Japan</th>
<th>Switzerland</th>
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<td>Conductor Configuration</td>
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<td>Subelements around copper strip</td>
<td>486-strand cable in square conduit</td>
<td>Subelements around CuNi strip in rectangular conduit</td>
<td>Flattened cable in roughened copper bar</td>
<td>Solder-filled square cable with tube at</td>
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<tr>
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# TABLE III. KEY PARAMETERS OF PMTF

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<td><strong>A. Electron Beam Test System</strong></td>
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<tr>
<td>Pulse Duration</td>
<td>1-100 cm$^2$</td>
</tr>
<tr>
<td>Cooling</td>
<td>0.05 sec - continuous</td>
</tr>
<tr>
<td>Max. Heat Flux</td>
<td>Closed Loop Water</td>
</tr>
<tr>
<td></td>
<td>Up to 30 kW/cm$^2$</td>
</tr>
<tr>
<td><strong>B. Multiple Beam Test System</strong></td>
<td></td>
</tr>
<tr>
<td>Steady-State Heat Source</td>
<td>40 keV, 1.6 MW H-Beam</td>
</tr>
<tr>
<td>Target Area</td>
<td>800 cm$^2$</td>
</tr>
<tr>
<td>Pulse Duration</td>
<td>Continuous</td>
</tr>
<tr>
<td>Cooling</td>
<td>Closed Loop Water</td>
</tr>
<tr>
<td>Max. Heat Flux</td>
<td>2 kW/cm$^2$</td>
</tr>
<tr>
<td>Parameter</td>
<td>Value</td>
</tr>
<tr>
<td>------------------------------------</td>
<td>---------------------</td>
</tr>
<tr>
<td>Gas Type</td>
<td>H, D, He, N, Ar</td>
</tr>
<tr>
<td>Operation Time</td>
<td>Continuous</td>
</tr>
<tr>
<td>Flux Range (ions/cm²s)</td>
<td>$10^{17}$ - $2 \times 10^{19}$</td>
</tr>
<tr>
<td>Plasma Density (cm⁻³)</td>
<td>$10^{11}$ - $10^{13}$</td>
</tr>
<tr>
<td>Electron Temperature (eV)</td>
<td>5 - 24</td>
</tr>
<tr>
<td>Plasma Area (cm²)</td>
<td>50 - 80</td>
</tr>
<tr>
<td>Sample Bias (Volts)</td>
<td>25 - 500</td>
</tr>
<tr>
<td>Base Pressure (torr)</td>
<td>$1 \times 10^{-7}$</td>
</tr>
<tr>
<td>Sample Temperature Range (°C)</td>
<td>20 - 900</td>
</tr>
<tr>
<td></td>
<td>Li</td>
</tr>
<tr>
<td>----------</td>
<td>------</td>
</tr>
<tr>
<td>HT-9</td>
<td>0.1</td>
</tr>
<tr>
<td>316 SS (cw)</td>
<td>2</td>
</tr>
<tr>
<td>Capsule I.D.</td>
<td>Material</td>
</tr>
<tr>
<td>-------------</td>
<td>----------------</td>
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<tr>
<td>S4T</td>
<td>Li$_4$AlO$_4$</td>
</tr>
<tr>
<td>S4B</td>
<td>Li$_4$SiO$_4$</td>
</tr>
<tr>
<td>S5T</td>
<td>Li$_2$ZrO$_3$</td>
</tr>
<tr>
<td>S5B</td>
<td>Li$_2$O</td>
</tr>
<tr>
<td>B8T</td>
<td>Li$_2$O</td>
</tr>
<tr>
<td>B8B</td>
<td>Li$_2$O</td>
</tr>
<tr>
<td>B8C</td>
<td>Li$_2$O</td>
</tr>
<tr>
<td>B9T</td>
<td>Li$_2$O (SC)&lt;sup&gt;e&lt;/sup&gt;</td>
</tr>
<tr>
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<td>Li$_2$O</td>
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<tr>
<td>B9C</td>
<td>LiAlO$_2$</td>
</tr>
<tr>
<td>B10T</td>
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<tr>
<td>B10B</td>
<td>LiAlO$_2$</td>
</tr>
<tr>
<td>B10C</td>
<td>LiAlO$_2$</td>
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<td>B11C</td>
<td>LiAlO$_2$</td>
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<tr>
<td>B12T</td>
<td>Li$_4$SiO$_4$</td>
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<tr>
<td>B12B</td>
<td>Li$_2$ZrO$_3$</td>
</tr>
<tr>
<td>B12C</td>
<td>LiAlO$_2$</td>
</tr>
</tbody>
</table>

<sup>a</sup>Pellet enrichment to ± 2% $^{6}$Li - except single crystal Li$_4$O

<sup>b</sup>Pellet diameter to ± 0.002 cm - except single crystal Li$_2$O

<sup>c</sup>Pellet column length to ± 0.02 cm

<sup>d</sup>Pellet density

<sup>e</sup>SC = single crystal; half of the pellets enriched on $^{6}$Li to 7.5%, the other half to 0.07%

<sup>f</sup>Approximately 700 micron size high density spheres
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major Radius</td>
<td>1.22 m</td>
</tr>
<tr>
<td>Minor Plasma Radius</td>
<td>0.45 m</td>
</tr>
<tr>
<td>Plasma Elongation</td>
<td>1.8</td>
</tr>
<tr>
<td>Plasma Current, Limiter Case</td>
<td>10 MA</td>
</tr>
<tr>
<td>Plasma Current, Divertor Case</td>
<td>9 MA</td>
</tr>
<tr>
<td>Toroidal Field</td>
<td>10.4 T</td>
</tr>
<tr>
<td>Toroidal Field Flat-top</td>
<td>3.7 s</td>
</tr>
<tr>
<td>Plasma Burn Time</td>
<td>3.1 s</td>
</tr>
<tr>
<td>Plasma Current Ramp-up Time</td>
<td>3.3 s</td>
</tr>
<tr>
<td>Neutron Wall Loading at 300 MW Fusion Power</td>
<td>6.8 MW/m²</td>
</tr>
<tr>
<td>Peak Divertor Plate Heat Flux</td>
<td>13 MW/m²</td>
</tr>
<tr>
<td>Toroidal Field Energy Requirement</td>
<td>1.73 GJ</td>
</tr>
<tr>
<td>Poloidal Field Energy Requirement (Divertor Case)</td>
<td>2.2 GJ</td>
</tr>
<tr>
<td>Combined Peak Power for TF and PF Coils</td>
<td>900 MVA</td>
</tr>
<tr>
<td>ICRH Initial Complement</td>
<td>10 MW</td>
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<tr>
<td>ICRH Full Complement</td>
<td>20 MW</td>
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<tr>
<td>Number of Full Field Pulses</td>
<td>3000</td>
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<tr>
<td>Number of 70% Field Pulses</td>
<td>50000</td>
</tr>
<tr>
<td>Re-Cool Time</td>
<td>60 m</td>
</tr>
<tr>
<td>LN₂ Consumption per Full Field Pulse</td>
<td>18000 l</td>
</tr>
</tbody>
</table>