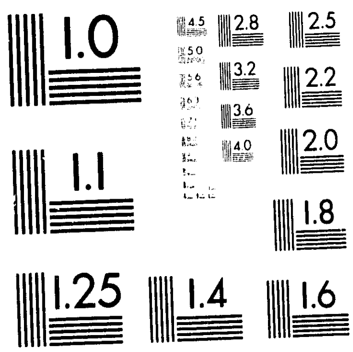
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THE ENGINEERING DESIGN OF TPX

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ABSTRACT

The Tokamak Physics Experiment (TPX) is designed to develop the scientific basis for a compact and continuously operating tokamak fusion reactor. TPX has a long pulse (1000s) capability, can accommodate high divertor heat loads, has a flexible PF system, and auxiliary heating and current drive systems that make it an ideal test bed for development of attractive reactor concepts. The design incorporates superconducting magnets in both the toroidal and poloidal field magnets. Long pulse deuterium operation will produce 6×10^{21} neutrons per year requiring remote maintenance of the in-vessel hardware. This paper provides an overview of the TPX Project and describes the design approach with emphasis on salient features of the tokamak.

INTRODUCTION

The mission of TPX is to develop the scientific basis for a compact and continuously operating tokamak fusion reactor. Supporting objectives are to demonstrate:

- operation at high beta (β_N>3.5%) with enhanced confinement (H>2),
- efficient current drive,
- advanced power/particle handling, and
- reliable (disruption-free) operation in fully steady state plasmas.

A conceptual design has been developed to meet these mission objectives by representatives from National Laboratories, industry, and universities. TPX is conceived as a national project - designed by a national team and operated as a national facility.

PERFORMANCE REQUIREMENTS

Plasma Parameters

Plasma parameters are summarized in Table I. The toroidal field, plasma current, and size were established to provide sufficient performance for advanced tokamak physics studies. The choice of aspect ratio (A=4.5) was motivated by reactor studies (e.g., ARIES-I, SSTR, and ITER-HARD) which found attractive design points at high aspect ratios with high bootstrap fractions. TPX will supplement the limited experimental database at high aspect ratio. Strong shaping (κ_x=2, δ_x=0.8) is provided to facilitate investigation of high beta, enhanced confinement regimes and goes hand in hand with the double null divertor.

The facility is initially configured for pulse lengths up to 1000s which is ample for current profile equilibration. The tokamak is designed for steady state operation. Ancillary systems are capable of being upgraded for steady state operation should longer pulse lengths be necessary.

Deuterium (D) operation is required to provide better confinement, more neutral beam (NB) heating, and more favorable access to advanced regimes than could be provided in hydrogen (H) alone. TPX is capable of extensive operation in deuterium, up to 2×10^5s per year. Total operating time (in H and D) is limited to 5×10^5s per year.

At the end of TPX operations, there may be interest in running short pulse deuterium-tritium (DT) experiments. The facility is capable of being upgraded to permit forming a deuterium plasma, over a period up to 1000s, and then injecting tritium for a 2s pulse.

Table I
Baseline Plasma Parameters

Major radius, R	m	2.25
Minor radius, a	m	0.50
Aspect ratio, A		4.5
Toroidal field, B_t	T	4.0
Plasma current, I_p	MA	2.0
Elongation, κ_x		2.0
Triangularity, δ_x		0.8
Pulse length	s	1000
Fuel species		H, D

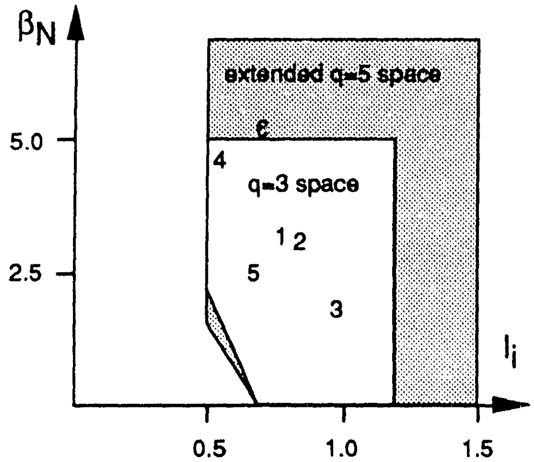
Flexibility

Flexibility is essential to the TPX mission. Advanced tokamak (AT) operating modes have been identified with parameters which span a wide range of operating space in beta, l_i, and q. The location of the AT operating modes in β-l_i-q space is shown in Figure I. TPX is designed for operation from l_i=0.5-1.2 and β_N=0-5 at q=3 and from l_i=0.5-1.5 and β_N=0-7 at q=5. A broader envelope than that defined by the AT operating modes has been specified for TPX to allow for successes not predicted by theory. Operation at low beta and low l_i is not required because it is not a region of scientific interest and vertical stability requirements are difficult to meet.

Flexibility in plasma shaping is important for access to all corners of β-l_i-q space, for forming single null (SN) plasmas, and for forming lower elongation plasmas. The baseline design is configured to operate with double null (DN) divertors. The DN divertor facilitates formation of high triangularity plasmas, minimizes the heat loads (and attendant space requirements) to the inboard divertor targets, and accommodates flexibility in plasma shaping while preserving coupling to the RF launchers. The PF system is designed to produce SN plasmas to allow DN-SN comparisons under

Figure I
β-l_i-q Operating Space

	AT Operating Mode	q
1	Standard Tokamak	3.80
2	Aries-I	6.90
3	Supershot	3.96
4	Non-monotonic q	3.55
5	VH	4.05
6	Aries-II	12.0



advanced plasma and divertor operating conditions. The PF system can also produce lower elongation plasma (κ_X=1.5-2) within the constraints of the baseline divertor design.

Flexibility in plasma heating and current drive (H/CD) is provided by the mix of heating/current drive technologies and is necessary for profile control. Plasma heating and current drive is provided by three systems: neutral beam (NB), lower hybrid (LH), and ion cyclotron (IC). The potential for additional flexibility is provided by substantially upgrading the power in each of the systems and by adding 10MW of electron cyclotron (EC) heating.

The NB system provides ion heating and bulk current drive, maintains plasma rotation, and supports ion temperature and current profile diagnostics. It also provides core fueling. 8MW of NB is provided by an existing 120keV TFTR beamline, modified for long pulse operation. The NB system can be upgraded to provide 24MW by adding two additional TFTR beamlines.

The ICH/FWCD system provides electron heating and centrally peaked current drive. 8MW of ICH/FWCD is provided through two launchers located in dedicated horizontal ports. The system is designed to operate in the range of 40-80MHz. The ICH/FWCD system can be upgraded to provide 18MW through three launchers.

The LHCD system provides off-axis current profile control, efficient current drive at low temperature, and electron heating. 1.5MW of LHCD at 3.7GHz is provided through a single launcher. The launcher is being designed for 3MW to accommodate potential upgrades in LHCD power.

Operating and Disruption Scenarios

Although the tokamak is designed for steady state operation, transients are still important for normal operations and disruptions. Reference operating scenarios have been established to ensure adequate volt-second and dynamic PF capability. The reference operating scenarios feature:

- a 20V startup,
- an inductive current ramp to full current,
- a short, inductive flattop to permit transition to current drive,
- a controlled current rampdown.

The reference operating scenarios establish a conservative basis from which to start machine operations.

Disruption scenarios are important because TPX will be pushing the boundaries of plasma stability. For the design of TPX, it is conservatively assumed that 50% of all shots will terminate in disruptions. Disruptions are characterized by current quench rates of 0.5MA/ms and halo currents up to 0.5MA.

TOKAMAK DESIGN

Magnet Systems

TPX will be the first tokamak with superconducting TF and PF coils. An elevation view of the tokamak is provided in Figure II. The toroidal field is provided by a toroidal array of sixteen TF coils. The requirements for low ripple and tangential NB access were leading factors in determining the size and number of TF coils. TF coils wedge together in the nose region to react centering forces on the coils.

Poloidal fields are provided by seven pairs of coils arranged symmetrically about the horizontal midplane. The four innermost coil pairs (PF1-4) form the central solenoid (CS) assembly. The CS is designed to be removable with the overhead crane. The CS and the ring coils (PF5-7) are all mounted off the TF assembly.

The TF and PF conductors are cable-in-conduit superconductors. All coils feature continuous windings with no joints inside the winding pack. The coils are cooled with forced flow, supercritical helium. Nb_3Sn superconductor is used for the TF, CS, and inner ring coils (PF5). NbTi is used for the outer ring coils (PF6-7) because of the lower fields in those regions.

The TF and PF coils are designed to criteria established in Ref. [1]. These criteria are modeled after the ITER CDA design criteria.

Figure II - TPX Elevation View

Vacuum Vessel

The vacuum vessel is a double-wall, titanium structure. Titanium was selected as the structural material because of its low activation, high electrical resistivity, and good strength at elevated temperatures. Titanium surfaces with line-of-sight to the plasma will be coated to prevent hydrogen embrittlement.

The vacuum vessel is fabricated in 90° sectors as shown in Figure III. Substantial access to the plasma chamber is provided through vertical and horizontal ports for plasma heating and current drive, vacuum pumping, diagnostics and remote maintenance.

TPX is required to bake out at 350°C. Bakeout is accomplished by circulating 350°C steam in the space between the vacuum vessel walls. During normal operation, the space between the walls will be filled with 150°C water.

The vacuum vessel is immediately adjacent to the TF assembly which operates at a temperature of 5K (-268°C). In order to reduce the heat leakage from the vacuum vessel to the TF assembly to manageable levels, superinsulation is wrapped on the outside of the vacuum vessel assembly.

Figure III - Isometric View of 90° VV Sector

Plasma Facing Components (PFC's)

Plasma facing components are designed to accommodate the radiation and particle heat loads associated with 45MW of plasma heating and current drive in steady state operation. Active cooling is required for all components for steady state operation. Plasma facing surfaces are carbon-based to minimize the influx of high Z impurities to the plasma. Plasma facing components include:

- a double null divertor for particle heat loads across the separatrix,
- an inboard first wall to protect the vacuum vessel,
- poloidal limiters to protect equipment in the region of the horizontal ports,
- armor to protect the passive stabilizer,
- ripple armor between TF coils for heat loads from ripple-trapped particles, and
- armor for neutral beam shinethrough.

Figure IV - Plasma Facing Components

High heat flux surfaces in the divertor are designed for $15MW/m^2$. This is viewed as a "technology limit". The R&D program addresses material and manufacturing development for high heat flux surfaces.

The outboard divertor features a closed divertor design. The objective of the closed divertor design is to develop a gas or radiative target to disperse the heat over a larger area thereby reducing the peak heat loads. The dispersive divertor is required for operation with 45MW of plasma heating and current drive and is important because of its reactor relevance. The divertor assembly features gas injection ports to assist in development of the gas/radiative target. A baffle is provided in the private flux region to minimize backflow of neutrals to the plasma.

The inboard divertor features an open design to maximize shaping flexibility (for operation with a single null plasmas, with low elongation plasma, and at the corners of β-l_i-q space.) Peak heat loads on the inboard divertor are projected to be less than $15MW/m^2$, even with 45MW of plasma heating and current drive.

Fueling and Pumping

Based on estimates of the effective particle confinement time and a plasma density of $0.5x10^{20}/m^3$, a typical particle loss rate from the plasma is 36 Torr-l/s. The loss rate of particles from the plasma must be compensated by fueling. Neutral beams fuel the plasma at a rate of 15 Torr-l/s per beamline. With three beamlines operating, neutral beams provide 45 Torr-l/s. Gas puffing can provide additional fueling. Pellet injection can be accommodated as an upgrade.

All of the particles lost from the plasma must be pumped. The re-entrant divertor configuration is favorable for providing a high plasma pressure. Vacuum pumping is provided by sixteen cryopumps connected through the vertical ports. The vacuum pumping system provides a pumping speed of 85,000 l/s which can be controlled with variable apertures in the vacuum pumping ducts.

Plasma Stabilization and Control

The strongly shaped plasmas on TPX are vertically unstable. Passive stabilizing plates are provided inboard and outboard of the plasma to slow the vertical motion of the plasma so active control can be effected.

Control coils inside the vacuum vessel provide vertical position control on fast timescales. The coils were located inside the vacuum vessel to minimize eddy current heating of the cold mass. Plasma position control on slow timescales is provided by the external PF coils.

At very high beta, TPX plasmas can become unstable to kink modes. Adding multiple connections between the upper and lower outboard stabilizer plates, as shown in Figure V, is being studied as an option to enhance kink stability.

Figure V - Outboard Passive Stabilizer Configured for Kink Mode Stabilization

Shielding and Maintenance

Shielding is provided to limit nuclear heating in the TF winding due to DD fusion reactions and to limit dose rates outside the shield envelope to permit hands-on maintenance. The water between the two walls of the vacuum vessel provides effective moderation of fast neutrons. The baseline design feature lead glass tiles doped with boron carbide on the outside of the vacuum vessel. The lead provides gamma attentuation while the boron is a strong absorber of thermal neutrons. The option of eliminating the lead glass tiles, increasing the VV envelope, and borating the water inside the vacuum vessel is presently being studied.

Around the vacuum pumping ducts, polyethylene doped with lead and boron carbide is provided. The large, horizontal ports are streaming paths for neutrons. The IC and LH launchers have shielding incorporated in the launcher designs. Diagnostic ports have shield plugs with minimum size holes provided for diagnostic access.

Provisions for recovery are required for all credible failure modes on TPX. Hands-on maintenance can be accomplished for ex-vessel components. TF/PF leads and coolant feeds have been located for easy access. Low activation materials need to be used where streaming paths through the shield envelope exist.

Remote maintenance will be required for in-vessel components after the two years of operation. (See Figure VI.) The remote maintenance concept for TPX is to use an in-vessel vehicle deployed through a horizontal port. The vehicle would mount on a toroidal rail which is part of the outboard passive stabilizer assembly. The vehicle would be equipped with manipulators and specialized end effectors as required for the assigned task.

Figure VI - Preliminary TPX Dose Map (in mrem/h)

Hands-on disassembly of the tokamak is possible following a one year cooldown period during which activated in-vessel components are removed remotely. This may be necessary to recover from a TF coil failure or to implement a major machine reconfiguration.

ASSEMBLY

The tokamak will be assembled in the TFTR test cell. The test cell will be prepared for assembly after decontamination and decommissioning of TFTR is completed.

TF coils will be shipped as two coil modules and re-assembled into quadrants on site. The 90° vacuum vessel sectors will be installed in TF quadrants.

Inside the test cell, the cryostat base will be installed. The remainder of the cryostat will then be installed and vacuum tested. The cryostat will be disassembled after vacuum testing, leaving only the base installed. The tokamak support structure is installed next. The lower ring coils are then lowered onto the cryostat base beneath where they will eventually be mounted on the TF assembly.

TF quadrants are installed individually. An alignment fixture will be situated in the bore where the OH will eventually be installed. Once aligned the TF quadrants will be welded together followed by the vacuum vessel. The tokamak is then ready for installation of the PF solenoid and ring coils.

Once PF coils are in place, the cryostat can be re-assembled and vacuum vessel ports installed. The in-vessel components are then installed. Assembly operations are expected to be completed in twenty-four months following successful completion of integrated systems testing and first plasma.

SCHEDULE

Preliminary Design is scheduled to begin in October, 1993. Industrial contracts will be awarded beginning in February, 1994.

The magnet systems are on the critical path for the project. Final Design of the TF and PF systems is scheduled to start in February, 1995 with manufacture of the TF/PF conductor beginning in June, 1995.

TPX will receive beneficial occupancy of the TFTR test cell in February, 1998, following decontamination and decommissioning of TFTR. Tokamak assembly will begin two months later. Integrated systems testing will begin in December, 1999. The construction project will be completed in March, 2000 at the time of first plasma.

COST

The total project cost (TPC) for TPX is \$540M in FY93\$. The TPC includes all systems required for TPX to be fully operational at the time of first plasma. Tokamak systems account for 50% of the TPC with the TF system being the single largest cost element. TFTR site credits, which include the test cell complex, neutral beam system, ICH/FWCD equipment, PF power supplies, and MG sets. are valued at \$150M.

Figure VII - TPX Cost Breakdown

Industry will play a major role in TPX, both intellectually and monetarily. Tokamak systems will be designed and fabricated by industry. Construction management and systems integration will be performed by industry. Essentially all new hardware will be industrially fabricated. It is estimated that 76% of the total funding will go to industry.

SUMMARY

TPX is a national program with laboratories, industry, and universities all playing major roles. It plays an essential role, complementary to ITER, in developing an attractive, compact DEMO reactor. The engineering design for TPX is sound and can meet the performance, cost, and schedule objectives which have been established.

ACKNOWLEDGMENTS

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[1] P. Heitzenroeder, Editor; "TPX Structural and Cryogenic Design Criteria; Doc. # 94-921012-PPPL/PHeitzenroeder-01; September, 1992.

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