2. National Compact Stellarator Experiment

2.A. Introduction

Compact stellarators hold the promise that the advantages of the stellarator– high-beta, disruption-free operation with low recirculating power– can be realized at low aspect ratio. Promising compact-stellarator plasma configurations, which use both coil currents and bootstrap currents to generate the rotational transform, have been developed theoretically to the point where they will soon be ready for test. Experiments are needed to assess the beta limits and the ability to achieve enhanced confinement in compact stellarators, as required for an attractive fusion energy concept. A proof-of-principle-scale facility capable of high beta with collisionless bootstrap current effects is required, similar in scope to those of tokamak proof-of-principle facilities such as Doublet III and PBX-M.

In describing the proof-of-principle (PoP) stage of concept development, the 1996 FESAC-SciCom Alternative Concepts Review Panel noted:

> This is the lowest cost program aimed at developing an integrated and broad understanding of basic scientific aspects of the concept which can be scaled with great confidence to provide a basis for evaluating the potential of this concept for fusion energy applications. Experimental activity in this step requires at least one device with a plasma of sufficient size and performance ($5 to $30M/year) that a range of physics issues can be examined. For example, for a toroidal confinement system, the plasma should be hot enough and large enough to generate reliable plasma confinement data, explore MHD stability, examine methods for plasma sustainment, and explore means of particle and power exhaust. The diagnostic set must be comprehensive enough to measure the relevant profiles and quantities needed to confront the physics. Proof-of-Principle experimental results are probably far from the fusion-relevant regime in absolute parameters but provides initial data for scaling relationships useful in establishing a predictive capability for the concept.

In practical terms, a PoP experiment provides a larger plasma volume, higher magnetic field, more plasma heating power, and a more extensive set of diagnostics than concept-exploration experiments. The greater capabilities afforded by a PoP are needed to test beta limits, study physics issues and scaling over a wide range of conditions, and operate at reactor-like collisionalities. A PoP facility satisfying these criteria and meeting the needs of the compact-stellarator program will be proposed by Princeton Plasma Physics Laboratory and Oak Ridge National Laboratory in partnership, with many other institutions collaborating. This facility, the National Compact Stellarator Experiment (NCSX), will be reconfigurable to be capable of expeditiously testing improved concepts that will be developed by the program.

Proof-of-principle scale toroidal confinement facilities are typically valued at $100M or more, including the confinement device itself, ancillary systems, and infrastructure, typically built up over many years of investment. However, the cost of NCSX will be reduced by making use of the existing PBX-M tokamak facility at PPPL, which provides substantial site credits, including plasma heating, diagnostic, and power systems, as well as the device itself. Most of the PBX-M tokamak itself will be re-used, including the toroidal and poloidal field coils, vacuum vessel, and support structure. By re-using the major torus systems, the NCSX can take advantage of their established technical capabilities and avoid not only the cost of re-creating the equivalent

capabilities but also some of the uncertainties attendant with commissioning and debugging of new systems.

The challenges of stellarator design have stimulated an intensive effort over a short period to adapt design tools, explore the properties of compact-stellarator configurations, and rapidly develop the technical basis for NCSX. The project is ready to proceed to the next stages of design development. The physics design process will continue with developing an attractive plasma configuration compatible with PBX-M and further defining the detailed physics requirements. The physics basis will be proposed for peer-review in a Physics Validation Review in about November, 1998. The engineering effort will develop an optimum low-risk facility design, detailed cost and schedule estimates, and detailed plans for carrying out the project. A Conceptual Design Review will be proposed for about May, 1999.

2.B. Experimental Goals, Research Plan, and Requirements

2.B.1. Goals

The goals of the NCSX are to test and to develop understanding of the physics of compact stellarator configurations with high beta and bootstrap currents, establishing the basis for their continuing development as attractive fusion power plants. An optimized quasi-axisymmetric (QA) plasma configuration will be tested initially. The facility will be modified to test improved configurations as they are developed by the program.

The specific scientific goals of the NCSX are to:

1) Demonstrate the ability of compact stellarators to operate at $\beta \approx 5\%$ without disruptions, with the rotational transform generated by coil currents and the bootstrap current. Determine the configuration requirements to avoid disruptions at high beta and high density.

2) Determine the beta limit and limiting mechanisms and their scaling with plasma parameters.

3) Determine the adequacy of the neoclassical-transport optimization to ensure good confinement at a reactor scale, and to ensure confinement of energetic particles (e.g. alphas).

4) Determine the ability to control turbulent transport and enhance confinement using flow-shear, the magnetic configuration, and control of particle fueling and radiation. Determine the dimensional and non-dimensional confinement scaling. Compare the observed transport with theoretical predictions and empirical scalings.

5) Test stabilization of equilibrium islands and neoclassical tearing modes at high beta by proper choice of magnetic shear for the bootstrap-current direction. These modes are observed to limit the achievable beta in tokamaks and are calculated to be important for stellarators.

6) Explore the compatibility of compact stellarators with methods to control the power and particle exhaust.

The initial scientific focus of the NCSX will be to evaluate plasma confinement and to address whether external rotational transform (from the 3D coils) can be used to suppress beta-limit disruptions in short-pulse operation (0.3-0.5 s of NBI heating), a step toward Goals 1 and 3. The initial hardware configuration will be tailored to this aim. This will allow us to test whether the high beta values can be realized and whether the key advantage of disruption avoidance is retained

in high-beta compact-stellarators, before implementing all of the research tools needed to accomplish all goals.

Ultimately, all goals must be met in plasmas with self-consistent bootstrap current, possibly including a small driven 'seed' current in the core, so that the results can be extrapolated to a reactor requiring low recirculating power. Due to the short pulse length of the available heating systems from PBX-M, the inductive equilibration to the bootstrap current will be studied in two ways: initially, tailoring the time-evolution of the discharge to minimize non-equilibrium currents, and later, extending the heating pulse length (to ~3 seconds) to ensure full inductive relaxation.

2.B.2. Research Plan

The research plan for NCSX will consist of two major phases, separated by a modification of the coil-system. The coil configuration for the first phase will be designed to produce an optimized quasi-axisymmetric stellarator configuration. The coil configuration for the second phase will be designed based upon the research results from the entire program, and could be, for example, either an improved quasi-symmetric or a non-symmetric quasi-omnigenous configuration.

The research plan for the first phase will consist of a sequence of four experimental campaigns:

1) Initial plasma operation and field-line mapping (six months). This campaign will test the accuracy of the stellarator magnetic field generation, the ability to initiate and control the plasma, and the operation of the initial diagnostics.

2) Plasma heating and transport (one year). This campaign will explore the flexibility, plasma confinement, and stability of the stellarator experiment at the initial heating power (6 MW). The adequacy of the neoclassical transport optimization will be assessed (Goal 3). The density limit will be documented, and the configuration requirements (if any) to avoid density-limit disruptions at low beta will be investigated, as a start on Goal 1. Studies of the plasma boundary will establish the readiness for high-power radiofrequency heating and the database for a possible upgrade of the plasma exhaust handling later in the program. In addition, this campaign will commission new diagnostics systems and will test RF coupling to the plasma at low power, to prepare for Campaign (3).

3) Confinement optimization and increasing beta (one year). This campaign will attempt to develop enhanced confinement regimes, Goal 4, using the techniques developed on tokamak experiments, including sheared rotation from NBI, reduced recycling by wall coating (B, Li) and conditioning, by edge radiation (RI-mode), and possibly by pellet fueling. The dimensional and non-dimensional scaling of confinement will be determined and compared to other configurations. These plasmas will then be used to test directly the predicted beta-limit and the predicted beta-limiting mechanisms (Goals 2 and 5). The configuration requirements to avoid disruptions and the disruption-free operating area will be documented (Goal 1). Current profiles approximating the bootstrap profile will be obtained by controlling the evolution of the plasma during the short heating pulse.

4) Long-pulse upgrade (one year). This campaign will be preceded by an upgrade to the heating systems to allow pulse lengths of ~3 sec, and a possible upgrade of the plasma-facing components for improved power and particle exhaust handling for long pulse (Goal 6). These upgrades will allow unambiguous equilibration of the current profile to the bootstrap current,

and will be used to document the high-beta disruption-free operating area in long-pulse operation (compared to the current-profile relaxation time) to complete Goal 1.

The successful completion of these four campaigns will complete all of the goals for assessing the first configuration. At this point, the coil system will be modified to a second configuration, as appropriate. This reconfiguration will also involve a new iteration of the edge power and particle exhaust handling design. The new configuration would be experimentally tested and optimized in an additional three campaigns, to accomplish the six goals.

5) Initial plasma operation and field-line mapping (six months). Similar to Campaign (1), but in the new configuration.

6) Plasma heating and transport (one year). Similar to Campaign (2), but in the new configuration.

7) Confinement optimization and increasing beta (one year). Similar to Campaigns (3) and (4), but in the new configuration and with the long-pulse capability.

Upon successful completion of the second phase, two compact optimized stellarator configurations will have been tested for their ability to operate at high-beta with enhanced confinement and without disruptions. This will provide a substantial experimental database for our understanding of confinement and stability in compact optimized stellarators, and will be used to assess the suitability of a Proof of Performance scale experiment.

2.B.3. Facility Requirements

In order to carry out this research plan and accomplish the experiment goals, the NCSX experiment will be flexible, well diagnosed, and heated at the multi-megawatt level. The design goal is to produce plasma configurations with substantial externally-generated rotational transform (ranging up to 40-50% of total), in order to provide a disruption-free operating regime with high confidence, in configurations with theoretically predicted beta limits of at least 3 - 5%. The beta limits may be lower than would be calculated for lower aspect ratio configurations, but are well above existing stellarators, are accessible with the available heating power, and allow the basic physics issues to be studied. The design may allow higher theoretical beta limits (e.g. above 5%, the ARIES-RS level) with larger bootstrap currents and thus possibly lower external rotational transform fraction

In order to be able to study transport scaling at a variety of beta levels, the transport optimization properties must be achievable over a range of beta, down to roughly half the predicted limit. The use of separate 3D saddle coils and axisymmetric coils to produce the external magnetic fields, discussed in Section 2.D, should allow experiments to vary the parameters thought crucial for controlling stability and transport: shear, magnetic transform, and plasma shape. The envisioned coil set includes a separate set of windings located on the outboard side of the plasma to control the edge magnetic shear, as originally discussed by Furth and Hartman [1]. The shear in the plasma core region will be controlled via a small near-axis seed current and the bootstrap current, as discussed in Section 2.C.3.

The NCSX should have an average minor radius of ~0.4 m, similar to the PBX-M average minor radius of 0.44 m. The average minor radius must be large enough to inhibit neutral penetration, which could produce substantial charge-exchange losses affecting beam heating, and must allow good NB-ion orbit confinement. PBX-M had a calculated NB-charge-exchange loss rate of ~3% at

a density of 5×10^{19} m^{-3} [5], while ATC had ~50% charge-exchange loss-rate due to a minor radius 2.5 times smaller and lower-density operation [6]. Neutral-beam heating efficiency is reduced on small machines due to orbit-losses [7]. Fast-ion (50 keV) orbits were well confined in PBX, with orbit losses ~10% of the NB power. Similarly, the non-thermal distortion of the plasma distribution will increase if the plasma volume is reduced, as the inverse square of the minor radius, due to the high-power heating required to reach the beta limit. Finally, the experimental results of this experiment should be comparable to similar-sized tokamaks, to assess the merits of the two configurations. There have been a number of well diagnosed tokamaks in this size region, including PDX/PBX, DIII, and ASDEX.

The magnetic field will be able to range from 1 to 2 T to allow the scaling of the beta-limit and confinement with field strength to be determined. This is necessary for both dimensional and non-dimensional scaling scans. Operation at 1 T will ensure that there is adequate heating power to reach the beta limit (even if the global confinement scales as in tokamaks), while 2 T operation will ensure adequate beam-ion orbit confinement even in a low rotational transform ($q \sim 6$) vacuum configuration. The PBX-M magnetic field range is 1 to 2.4 T. Power supply upgrades to allow the full range of operational flexibility at 2 T can be implemented before Campaign 3.

The beta-limit studies should be conducted at a reactor-like collisionality to ensure that the limiting instabilities are operating in the same kinetic/resistive regime. Representative operating points, accessible with the proposed heating power for different campaigns, are tabulated in Table 2-1. The confinement time is scaled to the ISS95 stellarator global confinement expression [2], assuming a fixed iota = 0.35 for all cases. The first column assumes a confinement time of 2.3 times ISS95, or 1.0 times ITER-89P for the equivalent tokamak. This is similar to the twice-ISS95 confinement routinely observed on W7-AS, and shows that beta ~3.5% will be accessible with the initial heating systems. The second column assumes the achievement of enhanced confinement to a level of 4 times ISS95, as part of Goal 4, and show that NCSX will be energetically able to access beta > 7%. In all cases, the density profile is assumed to be ~flat, as often observed in stellarators, and the temperature profile is assumed to be parabolic. The density has been chosen in each case to obtain a reactor-like ratio of collision frequency to bounce-frequency. In all cases, the density is below the Sudo density-limit scaling [3]. For comparison, PBX-M obtained 6.8% beta at 1.1 T using 5.5 MW of neutral beam injection (NBI) heating, obtaining a confinement time of 53 ms [4], which is 1.7 times the ITER-89P L-mode scaling law or 3.9 times the ISS95 prediction. Beta limits have also typically been studied at $B \approx 1$ T in tokamaks, including DIII-D, ISX-B, and ASDEX.

The PBX-M facility has 6 MW of NBI power at an energy of 40-50 keV, which will be sufficient for Campaign (2). The beams can be arranged to all inject tangentially to limit orbit loss and shinethrough. The heating power will be increased to 12 MW during Campaign 3 with the addition of 6 MW of ICRF power from an available RF system, if needed to reach the beta limit. This will ensure access to the beta limit even if enhanced confinement is not obtained, as indicated in the third column of Table 2-1, or at B = 2 T with enhanced confinement, as indicated in the fourth column.

Table 2-1: NCSX Accessible Operating Points

	L-Mode	Improved Confinement	Enhanced	Enhanced
Major radius, R (m)	1.5	1.5	1.5	1.5
Average minor radius, a (m)	0.45	0.45	0.45	0.45
Aspect ratio, R/ a	3.3	3.3	3.3	3.3
Toroidal field on axis, B (T)	1.0	1.0	1.0	**2.0**
Plasma heating power, P (MW)	6	6	**12**	**12**
Volume-average density n (10^{19} m^{-3})	5	8	6.5	20
τ_E multiplier * ISS95 Scaling	2.3	**4**	2.3	**4**
Energy confinement time, τ_E (s)	0.021	0.047	0.016	0.089
Volume-averaged beta β (%)	3.5%	7.8%	5.4%	7.3%
Central temperature, T_0 (keV)	1.9	2.4	2.0	3.6
Current relaxation time (s)	1.1	1.6	1.2	3.0

For Campaigns (4) and (7), heating pulses of ~3 seconds, substantially longer than the resistive equilibration time at B = 1 T, are required. By then the first wall must be capable of handling the full heating power for this pulse length. Graphite plasma-facing components will be bakeable to 350 C to remove trapped hydrogenic gasses. The system should be capable of boronization and lithium coatings, to control impurities and recycling, and it should be possible to glow-discharge-clean between experimental plasma discharges. It is expected that the edge rotational transform will vary with beta due to the bootstrap current, preventing use of an island divertor during the configuration exploration campaigns. Initial operation may use a limiter or a limited edge ergodic region instead of a divertor, with later upgrades to the edge system to ensure adequate particle exhaust handling. The NCSX should be capable of both gas-puff and inside-launch pellet fueling, to allow plasma density control and some degree of density profile control.

The NCSX must be well diagnosed to ensure accurate comparisons with theoretical predictions and to provide an accurate characterization of compact stellarators for projection to future experiments. The diagnostics will allow the time-dependent control and reconstruction of the 3-D magnetic configuration, the pressure and current profiles, the total plasma energy, and any MHD instabilities present. In addition, the evolution of the profiles of density and temperature will be measured for thermal transport analysis. Ideally, the radial electric field will be measured, or else the components of an ion rotation velocity, for comparison to theories of shear-flow turbulence stabilization. Initially, a basic set of diagnostics required for first-plasma operation will be implemented. Additional diagnostics will be brought on line as required to support the experimental program.

Plasma initiation in NCSX will consist of breaking down the pre-fill gas using RF heating and closed stellarator vacuum flux surfaces. Since the contribution of the bootstrap current to the magnetic transform at high-beta will be significant, the startup of the plasma and the equilibration of transform to the eventual bootstrap current must be designed. In addition, there must be sufficient magnetic transform at the start of NBI to confine the beam ion orbits. Three general strategies have been identified and will be investigated: 1) raise and hold the beta over a time longer than the resistive equilibration time, using first RF and then NBI; 2) inductively drive a plasma current to have a bootstrap-like profile to reduce the equilibration time and so accommodate

the short initial heating pulse; or 3) modify the 3D shape of the plasma during the heating pulse, by varying the coil currents, to vary the rotational transform and control the iota evolution.

2.C. NCSX Physics Design

2.C.1. Quasi-Axisymmetric Stellarators

The initial NCSX design will use quasi-axisymmetry [8,9] to obtain well-confined drift trajectories in a compact stellarator configuration. W7-X and quasi-helical configurations have previously solved the problem of poorly confined drift trajectories at relatively high aspect ratio, R/ a = 8-11. The quasi-axisymmetric approach is well suited to lower aspect ratios. Studies have focused on aspect ratios comparable to those of tokamaks, R/a = 3-4.5, corresponding to R/ a = 2-4. (When the tokamak convention is used for measuring the aspect ratio, R/a, a is taken to be the minimum minor radius.)

In going to lower aspect ratio, the requirement of reduced pressure driven currents is removed. The W7-X configuration has been designed to minimize Pfirsch-Schlueter and bootstrap currents, and quasi-helical configurations also have reduced currents. Quasi-axisymmetric configurations, in contrast, have Pfirsch-Schlueter and bootstrap currents comparable in magnitude to those of tokamaks. The bootstrap current, however, can be used to advantage. To the extent that it provides a significant fraction of the rotational transform, it allows the design of coils that are simpler and farther from the plasma. This is important for a reactor, where the distance to the coils divided by the major radius is a critical parameter that determines the minimum size of a device with adequate space for blanket and shielding. In addition, by designing the configuration to have the appropriate sign of shear relative to the direction of the plasma current, the perturbed bootstrap currents also suppress magnetic islands. This is the inverse of the neoclassical tearing instability that has been seen in tokamak experiments. An estimate of this effect finds that, for a configuration in which 50% of the rotational transform is supported by the bootstrap current, an island whose width would otherwise be 10% of the minor radius is suppressed by a factor of ~20, to about 0.5% of the minor radius.[10]

Although the bootstrap current can potentially be used to great benefit, it also brings with it some potential issues. Bootstrap currents may drive instabilities and may reintroduce disruptions into the stellarator. This risk is minimized by using the results of extensive stability calculations to guide the design, but experimental studies will be needed for a definitive determination of the conditions under which disruptions are avoided. An experimental study of the potential benefits and dangers of bootstrap currents will be a key focus of the NCSX experimental program, and the desire to address the physics issues associated with bootstrap currents is a key determinant of the required plasma size, magnetic field, heating power and pulse length. To provide a good probability that a disruption-free regime can be accessed in the NCSX, the coils will be designed to be capable of providing 50% of the rotational transform at the predicted beta limit. Experiments on hybrid tokamak-stellarator configurations on W7A[11] and CLEO[12] found that disruptions were suppressed when the fraction of the transform generated externally exceeded about 20%. These experiments studied the stabilizing effect of external transform on disruptions at low q and high density. The aspect ratio was high, and beta was low. The NCSX will investigate the suppression of disruptions at lower aspect ratio, for values of beta near the predicted beta limit.

One theoretical explanation proposed for the suppression of disruptions in W7A is the stabilization of the external kink mode by the externally generated transform.[13] These and related global

MHD stability calculations have suggested that stellarators are more stable than tokamaks to external kinks.[14,15] The studies have found that externally generated shear is particularly stabilizing for these modes. Our stability calculations confirm the strongly stabilizing effects of externally generated shear for quasi-axisymmetric configurations, allowing them to dispense with the tight fitting conducting wall and feedback that advanced tokamaks require for kink stabilization. These calculations will be discussed below.

In contrast to the situation for kink stability, ballooning stability tends to be more of an issue in stellarators than in tokamaks. Calculated ballooning beta limits for stellarators with R/ a < 10 are typically on the order of 2%, and quasi-symmetric stellarators have not been an exception to this.[16,17] The ballooning beta limit is a critical problem in the design of an attractive quasi-axisymmetric stellarator. This problem can be solved by imposing a strong axisymmetric (n = 0) component of ellipticity and triangularity on the shape of the plasma boundary in quasi-axisymmetric configurations. This approach opens up a previously unexplored regime of low-aspect-ratio, quasi-axisymmetric configurations having good ballooning stability properties. Configurations in this regime have drift trajectories similar to those of tokamaks, aspect ratios comparable to those of tokamaks, and bootstrap current as well as average ellipticity and triangularity comparable to that of advanced tokamaks. They therefore tend to look like hybrids between stellarators and advanced tokamaks. Relative to advanced tokamaks, however, they have the advantages that the externally generated transform reduces or eliminates the need for rf current drive, provides control over MHD stability properties through the ι profile, and should provide disruption suppression. Relative to the large-aspect-ratio stellarator, these configurations should provide much more compact designs, with much higher wall loading. The NCSX will be designed to flexibly explore a range of configurations in this regime.

The physics properties of a range of quasi-axisymmetric configurations in this regime have been examined. Ballooning stability, self-consistent bootstrap current profiles, neoclassical transport, and kink stability have been evaluated. Much of the initial analysis has been done on configurations having R/a 3 (R/ a 2.1). The studies of these configurations have been motivated by an interest in evaluating the potential attractiveness of the quasi-axisymmetric stellarator as a reactor concept. For the NCSX, auxiliary coils and associated support structure will be added inside the PBX-M vacuum vessel, and this will constrain the plasma aspect ratio to a higher value. The physics design studies have therefore been refocused to higher aspect ratio configurations (R/a 4.5, corresponding to R/ a 3.3) that are compatible with PBX-M. Constraining the configuration in this way leads to a reduction in the predicted absolute beta limit, but the beta limit is nevertheless well above the canonical 2% value predicted for earlier quasisymmetric designs, and is adequate to allow the basic physics issues (Section 2.B.1) to be well tested. An array of stellarator codes is being applied to assess the physics properties of configurations at the higher aspect ratio, and the designs are being adjusted as suggested by the results. To produce a reference design, two-, three, and four-period configurations that fit into PBX-M are being investigated. The calculations with these configurations thus far, and the calculations with the lower aspect ratio configurations (R/ a 2.1), lend confidence that attractive configurations having adequate quasi-axisymmetry, self-consistent bootstrap currents, and attractive ballooning and kink stability beta limits will emerge from this design process.

The analysis of designs that fit into PBX-M is still in progress. The status of these studies is described below, and lower aspect ratio configurations are also described to give a more complete picture of the physics properties of QAS configurations.

2.C.2. Plasma Configurations

In this section we introduce four quasi-axisymmetric configurations, two at the lower aspect ratio and two at the higher aspect ratio, whose physics properties have been examined in some detail. Configurations are generated using an optimizer based on the fixed-boundary VMEC equilibrium code. [18] The four are summarized in the following table; their configuration details are described in the remainder of this section, and their physics performance will be discussed in the remainder of Section 2.C.

Quasi-Axisymmetric Configurations Studied

	I	II	III	IV
Aspect Ratio, R/ a	2.1	2.1	3.3	3.4
"Tokamak" Aspect Ratio, R/a	3.0	3.0	4.5	4.5
Number of field periods	2	4	3	3
Rotation transform (i) rising near edge?	No	Yes	No	Yes
Externally-generated transform	0.16 at edge, flat profile	0.2 at edge, 0 in center	0.13 at edge, flat profile	0.15 at edge, 0.12 at center
Volume averaged beta (%)	5.3	6.5	3.7	4.7
Bootstrap current (kA) at B=1 T	300	350	265	200
Internally-generated transform at edge	60% of total	50% of total	65% of total	55% of total

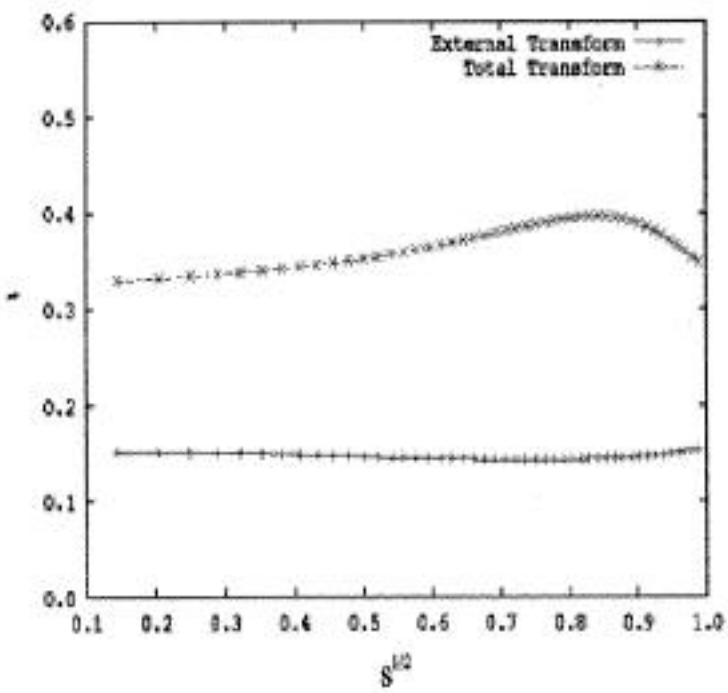

Fig. 2-1. Rotational transform profile for a two period configuration with R/ a 2.1, flat external transform profile. (Configuration I).

Figure 2-1 shows the transform profile and the portion of the transform generated externally for Configuration I, a two-period configuration with aspect ratio R/ a 2.1 (R/a 3). The quantities are plotted versus the square root of the toroidal flux normalized to its value at the plasma boundary. The configuration has β = 5.3%, and an externally generated transform of about 0.16, with the profile of the externally generated transform relatively flat. At β = 5.3%, the bootstrap current generates about 60% of the transform at the plasma edge. The total transform profile is non-monotonic.

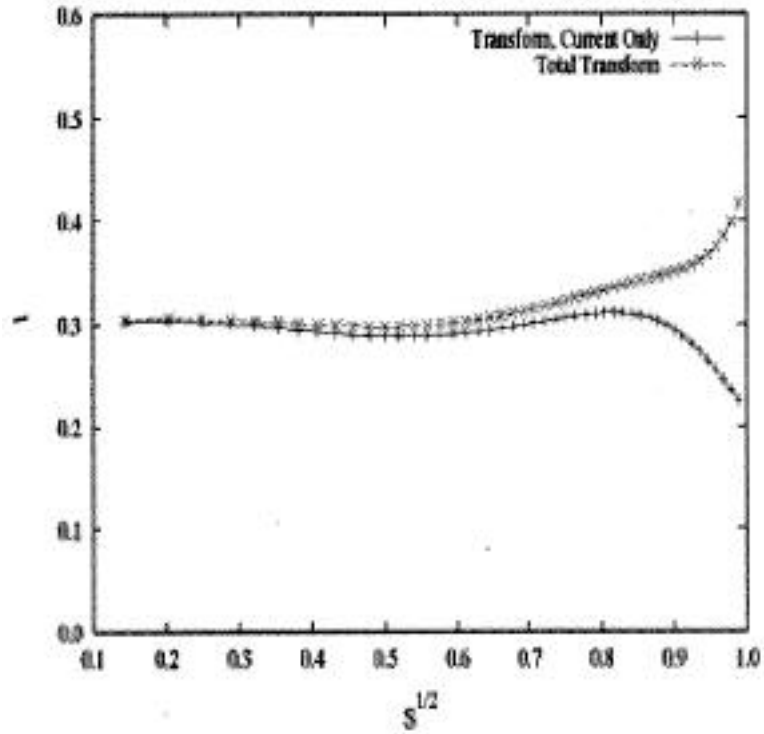

Fig. 2-2. Rotational transform profile for four period configuration with R/ a 2.1, monotonic rotational transform profile. (Configuration II)

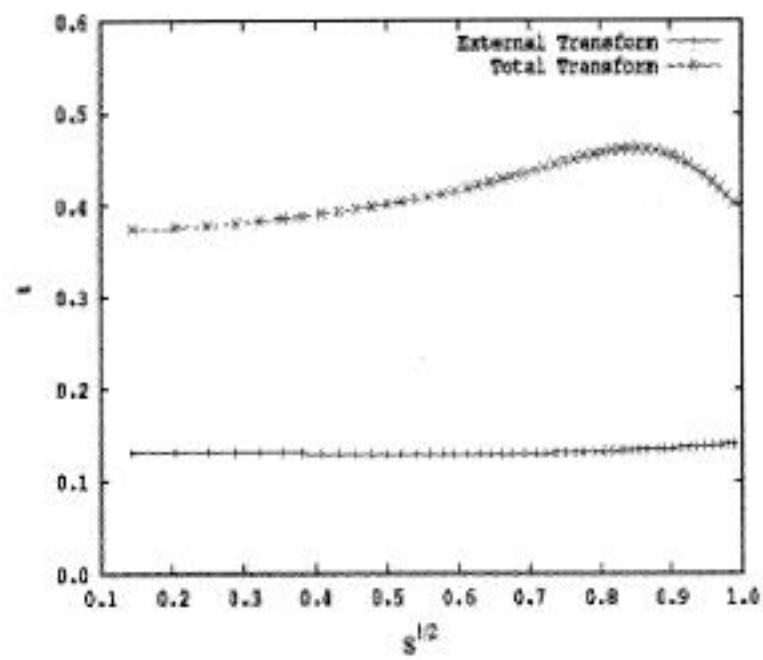

Fig. 2-3. Rotational transform profile for a three-period configuration with R/ a 3.3, flat external transform profile (Configuration III).

Externally generated transform can be used to produce quasi-axisymmetric configurations with monotonic transform profiles. One such profile is shown in Fig. 2-2. This corresponds to Configuration II, a four period configuration with R/ a 2.1 (R/ a 3), β 6.5%.Externally generated transform is produced only near the plasma edge, and it rises to about 0.2 there. The possibility of producing a monotonically increasing transform profile (monotonically decreasing q profile) is a potential advantage of stellarators over tokamaks, where this is not possible. A reversed-shear tokamak must always have a shear reversal layer outside of which q is increasing. The shear reversal layer tends to be associated with stability problems, such as infernal modes. In the region of increasing q, the tokamak is potentially unstable to neoclassical tearing modes.

Figure 2-3 shows the rotational transform profile for a plasma which is constrained to fit into the PBX-M vacuum vessel with adequate room for coils and support structure, Configuration III. It has a higher aspect ratio, R/ a 3.3 (R/a 4.5), three periods, β 3.7%, and about 35% of the transform at the edge generated externally. The profile is nonmonotonic. Figure 2-4 shows the plasma boundary for Configuration III. Variants of Configuration III with increased externally generated shear have recently been developed, and these are still undergoing optimization. One such variant is Configuration IV, with R/ a 3.4, a

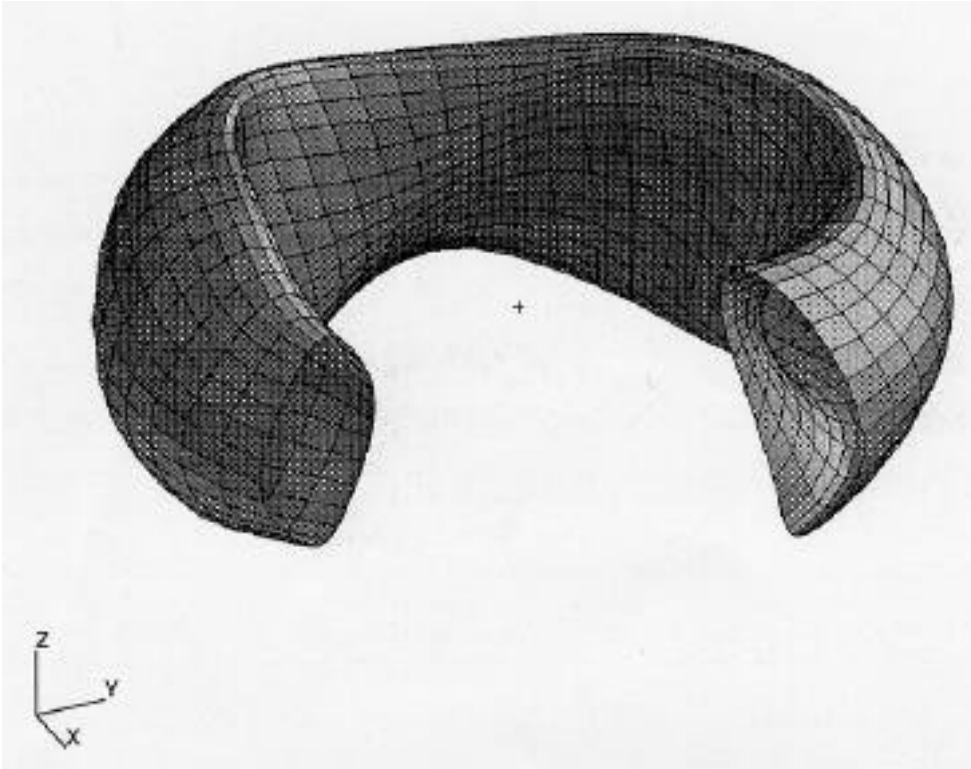

Fig. 2-4. Plasma boundary of quasi-axisymmetric configuration with R/ a 3.3.

rising ι near the edge, and about 45% of the transform at the edge generated externally.

2.C.3. Self-Consistent Bootstrap Current Profiles

The bootstrap current is determined by the Fourier spectrum of mod(B) in Boozer coordinates. The n ≠ 0 Fourier coefficients vanish in perfect quasi-axisymmetry, and the bootstrap current therefore tends to look like that in a tokamak. In particular, the bootstrap current is comparable in magnitude to that in a tokamak, providing substantial rotational transform. The design procedure for quasi-axisymmetric stellarators adopts, as a starting point, advanced tokamak pressure and current profiles in which the current profile is well aligned with the bootstrap current drive. The current is about 90% bootstrap driven in configurations I and II, and about 80% bootstrap driven in Configuration III.

For steady state operation in a reactor, it would be desirable to have the internal current driven almost entirely by the bootstrap effect. Configurations with substantial externally generated transform require little or no seed current to maintain an equilibrium. The seed current in Configuration III has been reduced from 38 kA to about 1.2 kA (less than 0.5% of the total current), and the equilibrium has been reconverged with self-consistent bootstrap currents calculated using a bootstrap code developed at NIFS in Japan.[19] Fig 2-5 shows the rotational transform profiles for seed currents of 38 kA, 12 kA and 1.2 kA. The sensitivity to relatively small seed currents gives external control over the transform profile near the magnetic axis, increasing the flexibility of the experiment.

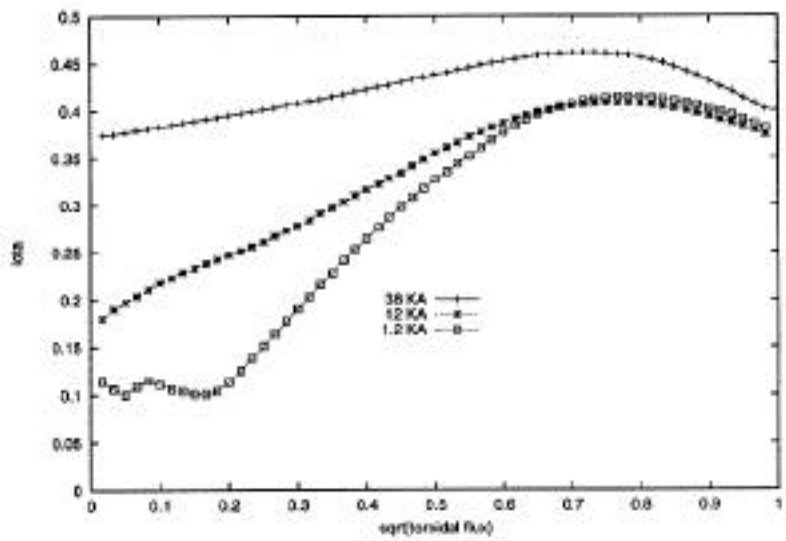

Fig. 2-5. Rotational transform profile for several values of the seed current in Configuration III.

2.C.4. Ballooning Stability

Ballooning stability[20] is assessed using the three-dimensional ballooning code developed by Cooper at CRPP, Lausanne. To validate the results, the predictions have been benchmarked against those of the three-dimensional ballooning code developed by Nuehrenberg's group at IPP-Greifswald.

The four configurations described above are calculated to be ballooning stable at β values of 5.3%, 6.5%, 3.7%, and 4.4% for Configurations I, II, III, and IV, respectively. The difference in the ballooning beta limit between the low and high aspect ratio configurations is generally consistent with an inverse aspect ratio scaling of the beta limit. The difference in the ballooning beta limit between Configurations I and II, and between III and IV, appear to be due to shear stabilization of the ballooning mode near the plasma edge.

Figure 2-6 shows the ballooning eigenvalues calculated for Configuration III as a function of the flux coordinate at $\beta = 3.7\%$. All of the eigenvalues are negative, indicating stability. Note that there is significant headroom in the plasma interior to increase the pressure gradient there and raise beta, while maintaining ballooning stability. The same is true of Configuration IV.

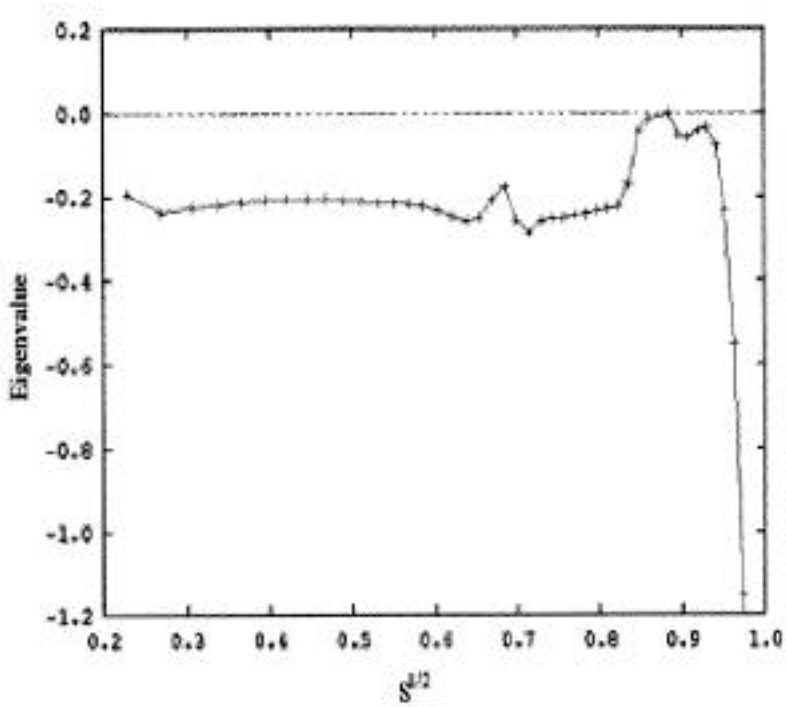

Fig. 2-6. Ballooning eigenvalues for Configuration III.

2.C.5. Neoclassical Transport

The configurations described here were generated using an optimization code to minimize the $n \neq 0$ Fourier components of mod(B) in Boozer coordinates, and they are of necessity only approximately quasi-axisymmetric. Even in principle, it is possible to impose exact quasi-axisymmetry on at most a single flux surface, with the equilibrium equations dictating a deviation from quasi-axisymmetry that is third order in the inverse aspect ratio away from that flux surface.[21]

We evaluate the neoclassical transport in our configurations to verify that they are sufficiently close to quasi-axisymmetry. For that purpose, transport assessments have been made using a combination of numerical and analytic tools.[22] When applicable, analytic theory is advantageous, permitting one to assess scalings and effects such as the self-consistent radial electric field. However, the theory is generally based on simplified models of the stellarator fields, which often do not strictly apply to configurations of interest. In such cases theory can often still be used as a more approximate estimate of the confinement performance one may expect. Complementary to this has been the use of Monte-Carlo guiding-center codes developed at Oak Ridge and Princeton, which use numerical descriptions of the magnetic field produced by MHD equilibrium codes. Such results make no assumptions about the characteristics of the field. The two methods have been benchmarked against each other. Analytic theory permits one to perform the integration over energy, as well as computing the value of the radial electric field needed to insure ambipolar particle fluxes, not included in the Monte Carlo calculations but needed for proper estimate of the confinement times.

It is desired that the neoclassical energy confinement times be long compared to the energy confinement times expected for turbulent transport. We compare the calculated neoclassical energy confinement times with an estimate of the level of turbulent transport, the ISS95 empirical International Stellarator Scaling. Our evaluations assume the PBX major radius R=1.5 m. The calculations for the R/ a 2.1 configurations used B = 1.4 Tesla, T = 3.5 keV, and $n_{e0} = 0.3 \times 10^{20}$ m^{-3}. For Configuration III, the parameters were shifted to B = 1.0 Tesla, T = 2.2 keV, and $n_{e0} = 0.7 \times 10^{20}$ m^{-3}. The following results were obtained:

Configuration	$\tau_{Ei}^{neo} / \tau_E^{ISS}$	$\tau_{Ee}^{neo} / \tau_E^{ISS}$
I	2.3	1.3
II	4.5	4.9
III	3.7	10.3

The neoclassical confinement of Configuration IV has not been fully evaluated, but the level of residual ripple indicates that further optimization is necessary. For another recently developed variant of Configuration III with monotonic rotational transform profile, the non-quasisymmetric ripple has been reduced to the point where the neoclassical transport in the absence of any electric field, is indistinguishable from tokamak neoclassical, but the shear is not adequate to provide good kink stability.

2.C.6. External Kink Modes

We find that it is possible to stabilize external kink modes even in the absence of a close fitting conducting wall by imposing a sufficiently strong externally generated shear near the plasma boundary. This possibility of MHD stabilization via externally generated transform is one of the unique advantages of stellarators. The potential use of externally generated shear to stabilize kink modes was suggested in several early papers.[13,14] We have evaluated stability to external kink modes using the Terpsichore[24] 3D MHD stability code developed in Lausanne, Switzerland.

Figure 2-7 is a plot of the growth rate as a function of the shear at the plasma edge, calculated at $\beta = 6.5\%$, for two series of low-aspect-ratio (R/ a 2.1), four-period equilibria having the values of ι at the edge indicated. There is a conducting wall at twice the minor radius, where it is believed to have little effect on the stability of the external kink mode. For each series of equilibria the pressure profile and the current profile are kept fixed. The shear is controlled by varying the

shape of the plasma boundary. The instability is stabilized in each case for sufficiently large shear. The stable equilibrium with i_{edge} = 0.35 corresponds to a modification of Configuration II in which the current is reduced to 70% of the magnitude retained earlier. A reduction in the bootstrap current of this magnitude can arise from realistic collisionality effects. The equilibrium is also stable to ballooning.

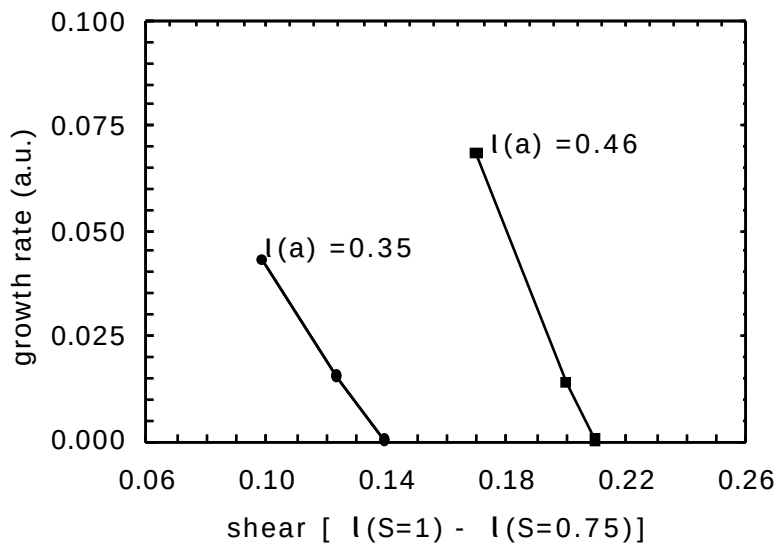

Fig. 2-7. Growth rate of the external kink mode as a function of shear at the plasma edge.

An equivalent tokamak equilibrium has a kink stability beta limit of about 2.5% when the wall is placed at twice the minor radius. Advanced tokamaks rely on a conducting shell for stability to external kink modes. To maintain stability on the L/R time of the shell, they must either rotate the plasma, raising issues of recirculating power, or they must provide multi-mode feedback stabilization on that time scale. The possibility of eliminating these requirements is a very significant advantage of the stellarator.

Configurations at the higher aspect ratio (R/ a 3.3) with various levels of externally generated shear are presently being studied. Configuration IV is kink stable at a beta of 5.2% with the wall at twice the minor radius. It has a ballooning beta limit of about 4.4%, but at present has an unacceptably high level of non-quasisymmetric ripple. Optimization of this and related configurations is in progress. Experience in optimizing the lower aspect ratio configurations lends confidence that an R/ a 3.3 configuration combining attractive MHD stability and good neoclassical confinement will emerge from the systematic design process.

The calculations described in this section have demonstrated the significance of shear near the plasma edge. In the experiment it will be desirable to have the flexibility to control the edge shear. This will be done using a set of high toroidal mode number, outboard coils similar to those proposed by Furth and Hartmann[25]. These coils produce a helical corrugation of the magnetic surfaces localized to the region near the midplane and to the outer region of the plasma, and they produce an associated rotational transform. Limiting the corrugation to near the midplane should minimize ripple-induced transport. (Helically trapped particles on the midplane drift vertically, parallel to the flux surface.) Figure 2-8 shows the modification produced in the rotational transform profile for one quasi-axisymmetric configuration by a helical corrugation of this sort. The costing of the NCSX has made provision for a set of such outboard coils to control the shear.

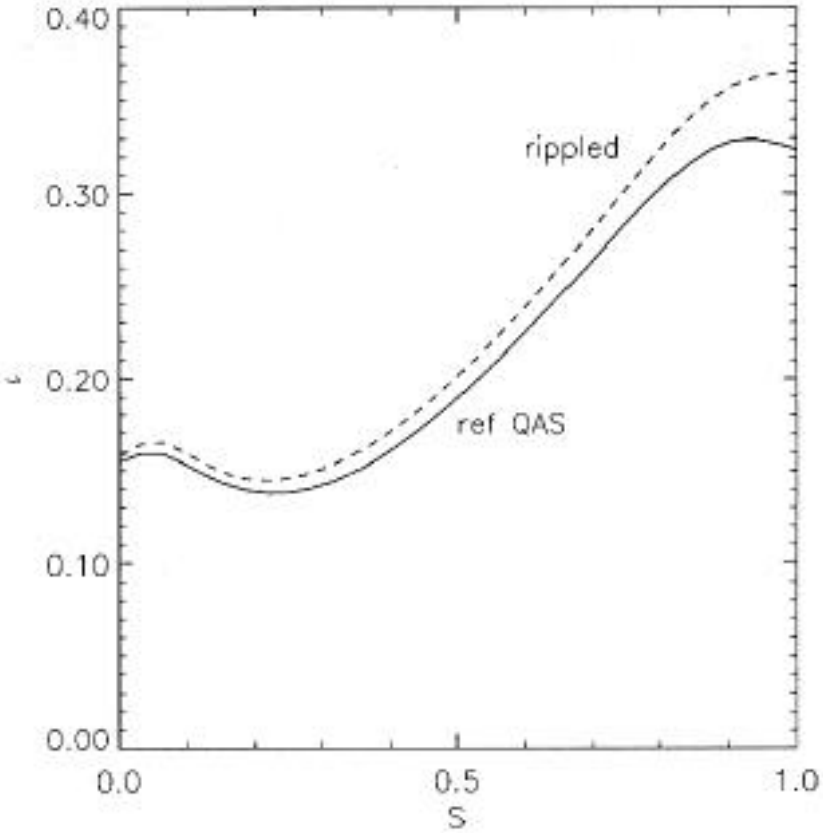

Fig. 2-8. Modification of the rotational transform profile of a quasi-axisymmetric configuration produced by a helical corrugation of the plasma boundary localized to the region near the outer midplane.

2.D. Engineering Design

Engineering design studies have been carried out to investigate feasibility and cost issues associated with the construction of the NCSX based on a modification of the PBX-M facility at the Princeton Plasma Physics Laboratory. The objective is to meet NCSX experimental requirements with a cost-effective and low-risk design. The activity is in the preconceptual stage; further design work is in progress to refine the reference plasma configuration, performance requirements, and machine design.

2.D.1. The PBX-M Facility

The PBX-M facility site credits include: a large, open vacuum vessel; a flexible poloidal field (PF) system; a toroidal field (TF) system; power and energy supplies; auxiliary heating systems; extensive diagnostics; and a test cell and site utilities. The original cost of the PDX tokamak structures alone was $18M in circa FY-77 dollars, and the neutral beams about $12M (about $46M and $30M, respectively, when escalated to FY-98 dollars). The facility has remained intact and in a state of readiness for re-start within a few months.

The PBX-M TF coils and power supplies can provide 1 T at a major radius of 1.5 m with a flattop of 22 s, and 2 T for 1.5 s. They are driven from the C-Site MG sets, which provide ample power and energy to meet NCSX needs. The TF coils are designed with demountable joints, which provides the option of disassembling them, if necessary, to facilitate installation of new components.

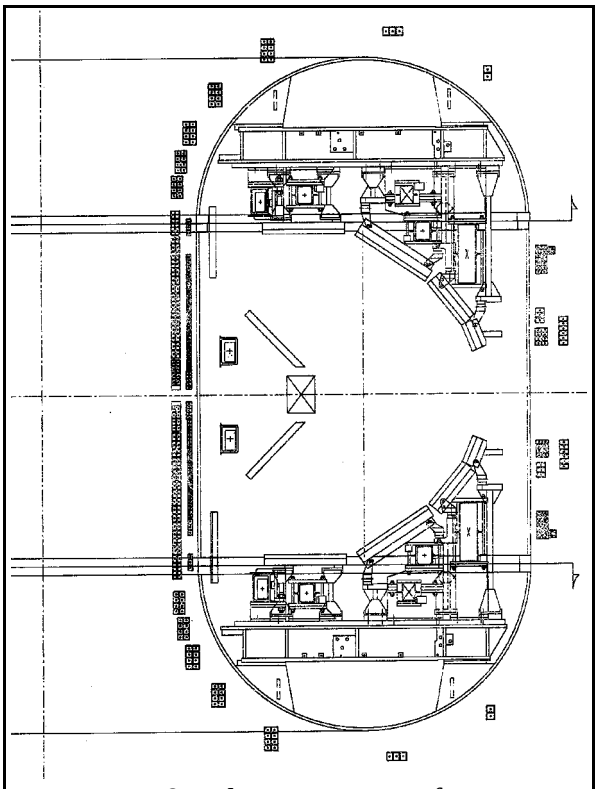

Figure 2-9 - Elevation View of PBX-M Tokamak

The bore of the TF coils is occupied by the PF coils and a large racetrack-cross-section vacuum vessel with a removable upper dome. (Fig. 2-9) The existing PF system provided the poloidal flux change and equilibrium fields necessary to drive and sustain plasma currents up to 600 kA for 1.5 s. Lower currents could be sustained for longer times. The NCSX is anticipated to require plasma currents, perhaps up to 400 kA, with beta values within the range of those achieved on PBX-M (up to 6.8% at 1.1 T toroidal field). The coils which provide the poloidal flux change (the OH coils), vertical field (primarily the outer EF coils), and radial field (for vertical position control) are located outside the vacuum vessel and can be re-used, along with their power supplies, for all of the plasma configurations being considered.

The large, open vacuum vessel of PBX-M provides outstanding flexibility. The inside of the vacuum vessel has already been configured three different ways-- PDX, PBX, and PBX-M. In-vessel PF coils have been relocated or removed, new in-vessel PF coils have been added, and plasma facing components have been reconfigured. The reconfigurations being considered for NCSX are conceptually similar, except that the new in-vessel coils would provide a helical (non-axisymmetric) field, instead of a purely poloidal (axisymmetric) field.

2.D.2. Torus Modifications

For initial investigations of engineering design solutions, a two-field-period, quasi-axisymmetric plasma with 40% externally-generated rotational transform was used as a prototype plasma configuration. A beta value of 4.5% was assumed with a plasma current of 220 kA. This configuration was not optimized for physics performance, but scaled from an optimized low-aspect-ratio configuration to fit within the PBX-M vacuum vessel. The plasma was constrained to fit within two fiducial cylinders defined with radii of 1.05 m and 1.85 m to ensure that it would fit within the PBX-M vacuum vessel with reasonable space left over inboard and outboard of the plasma for a scrape-off layer (0.05 m), first wall and liner (0.05 m), gap (0.025 m), and helical-field coils and structure (0.15 m). The average major radius is 1.53 m with a vacuum toroidal field of 1.05 T. While a fully-optimized reference plasma configuration for NCSX has not yet been developed, this prototype is representative of the configurations of interest for engineering and cost evaluation purposes.

Coil geometries are determined using the NESCOIL suite of codes developed by P. Merkel and M. Drevlak at IPP in Germany. The coils are designed to minimize the calculated error in the normal component of magnetic field on the plasma boundary, taking into account fields from plasma currents and from TF and PF coil currents.

To facilitate the selection and development of the helical-field coil design, a survey was made of possible coil topologies using NESCOIL. The saddle coil configuration appeared attractive for a number of reasons. Saddles can be wound in place without requiring a spool of conductor or a winding form to be rotated, or can be constructed in segments brought in through the horizontal ports. They allow good access on the outboard midplane for plasma heating and diagnostic viewing. From a control standpoint, they have the advantage of no inductive coupling between the saddle-coil and the TF and PF coil circuits. The basic pattern for the inboard coils is a set of six nested saddle coils. These coils are mirrored once per period and rotated 180° for the second period. The total number of inboard saddle coils totals twenty-four. The complete set of inboard saddle coils is shown in Fig. 2-10. The conductors are nominally centered on a surface that conforms to the plasma surface, offset by 0.20 m. Currents in individual saddles range from 40 kA-turns to 95 kA-turns.

The saddle coils are mounted on vertical, radially oriented ribs, as shown in Fig. 2-10. The ribs are supported on the top and bottom by horizontal plates which react the radial bending loads transmitted from the vertical ribs. Preliminary calculations indicate that the stresses in the ribs and horizontal plates are modest (under 100 MPa) for the nominal thicknesses (1.25 cm). The in-vessel structure formed by the ribs and horizontal plates would be pre-assembled with high precision pins and then disassembled, fed through the horizontal ports, and re-assembled within the vacuum vessel. Installation of the saddle coils would follow. The present design concept calls for the use of a flexible conductor which can be laid into precisely located tracks and potted in place. A vacuum seal is provided by welding covers over the coils. Similar techniques were sucessfully accomplished on PBX-M.

Besides the inboard saddle coils, additional saddle coils on the outboard side are assumed to be required to provide detailed plasma shape control and flexibility. A periodic coil structure, also poloidally localized on the outboard side to provide independent edge shear control (e.g., “Furth-Hartmann” coils) is included in the cost estimate as well.

Between the saddle coils and the plasma is a vacuum liner which provides a low conductance barrier between the plasma region and the region of the coils and vacuum vessel. It is constructed of panels which can be removed to provide access to the saddle coils and other components for maintenance and precision alignment of the coils. The liner will be bakeable to 350°C and designed to be armored completely with carbon tiles.

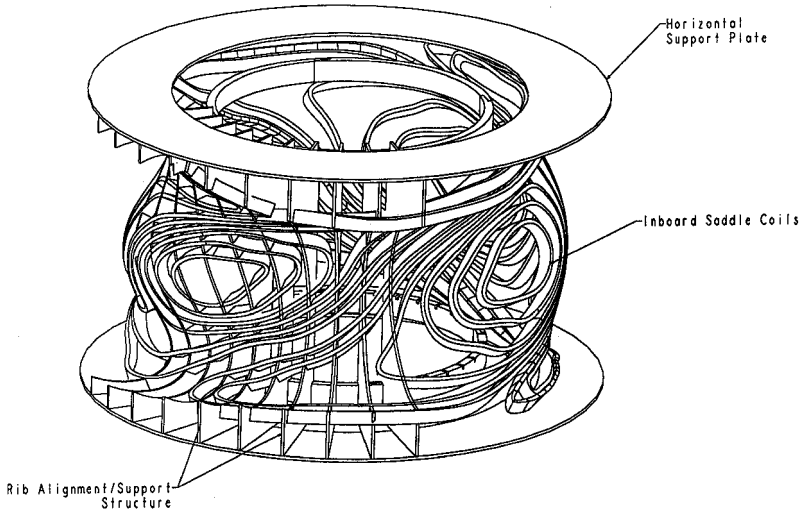

Figure 2-10 - Isometric View of Inboard Saddle Coils and In-Vessel Support Structure

An alternative to the through-the-port assembly method is to install the new in-vessel components from overhead. This would entail clearing away much of the equipment surrounding the device, removing much of the machine support structure, and disassembling the TF coils. The TF coils are jointed at the top and bottom, requiring removal of all TF coils turn by turn. The upper OH coils and upper vacuum vessel dome would be removed, providing access to the vessel interior. In this approach the in-vessel structures can be installed in larger subassemblies than is possible with port access, and assembly activity in the confines of the vacuum vessel is significantly reduced. Assembly options will be further evaluated and the optimum one selected as part of the conceptual design process.

In summary, a design concept for an experiment to test the physics of compact stellarators has been developed based on an in-vessel modification of the PBX-M facility. More work is required to establish the feasibility and optimize this design concept, but preliminary design and analysis results are encouraging.

2.D.3. Facility Modifications

Plasma heating systems will be provided in the initial facility configuration and augmented as required by the program. Initially, an electron cyclotron heating system (~100 kW, ~100 ms) will be provided for plasma initiation, and a reconfiguration of the existing PBX-M neutral beam injection (NBI) system (6 MW, 0.3 s) for heating to high beta.

The NBI system includes four beamlines, of which currently two are tangential and two are nearly perpendicular to the plasma. The two perpendicular beamlines may be re-oriented to tangential for initial NCSX operation. The NBI system would later be upgraded for long pulses, providing 7 MW for 3-5 s. The approach will depend on technology developments, such as an effort planned by the MAST project at Culham Laboratory to extend the pulse length of similar ORNL-developed neutral beams to 5 s.

Ion cyclotron (ICRF) heating will be added later in the program if needed to increase the total heating power to 12 MW for 3 s pulses. The system would use 30-MHz sources already at the

PBX-M site, which could be shared with the NSTX project. New launchers would be required for high power operation and for special plasma control needs (e.g., flow-shear generation).

The extensive complement of diagnostics available at PBX-M will be re-used to the extent possible, with modifications as needed. A basic set of diagnostics (magnetics, visible and infrared cameras, microwave interferometer, wide-angle bolometer, and SPRED survey instrument) will be implemented for first-plasma operation. Additional diagnostics (for example, Thomson scattering, charge-exchange recombination spectroscopy, motional Stark effect, visible bremsstrahlung array, bolometer array, electron cyclotron emission radiometer, Mirnov loop array, x-ray imaging, edge probes, edge reflectometer, fast-ion loss probe, and beam-emission spectroscopy) will be brought on line as required to support the experimental program.

The TF and PF coils will be powered with their existing power supplies and the site motor-generator system. Required maintenance and testing will be performed prior to operation. New power supplies will be procured for the saddle coils. Later in the program, power system upgrades will be implemented, if necessary, to operate at 2 T.

The central instrumentation and control system and the data acquisition system will be modernized. The approach will be patterned after the NSTX designs to minimize operating costs and facilitate national collaboration in NCSX research.

2.E. Cost and Schedule

Cost and schedule estimates have been developed based on the pre-conceptual design work done to date on the in-vessel structures, experience from previous projects, and detailed knowledge of the PBX-M equipment. Contingency at 25% is applied on an across-the-board basis. A more detailed bottoms-up estimate must await the completion of the conceptual design. Estimated project costs to configure the facility for first plasma are tabulated in Table 2-2 and total $34,860K in FY-99$.

The key proposed near-term milestones for the project are a Physics Validation Review (November, 1998) and the Conceptual Design Review (May, 1999). A complete set of project milestones will be developed during conceptual design. A preliminary budget profile for the project, assuming first plasma at the beginning of FY-03, is provided in Table 2-3. Conceptual Design (estimated at 10% of the TPC) will occur in FY-99 and construction is assumed to be completed by the end of FY-02. These costs are tabulated as “Construction TPC.” During construction, a parallel research preparation activity will be conducted to plan the experimental program, develop experimental data analysis tools, and prepare advanced diagnostics and other facility improvements to be implemented after first plasma. These research preparation costs and the operating costs for the first few years (assuming $20M per year in constant FY-99$ for facility operations, physics research, and experiments) are tabulated as “NCSX Operations.” The expected facility enhancements to achieve long-pulse capabilities with full advanced diagnostics as soon as practical, consistent with the physics research program are tabulated as “NCSX Enhancements.” The facility would likely operate for about ten years (through FY-2012), including an internal coil reconfiguration at an interim point, as appropriate.

Table 2-2. NCSX Construction Cost Estimate (FY-99 M$)

Description		
Torus System Modifications		13.9
Plasma Facing Components	2.7	
Vacuum Vessel & In-Vessel Structures	3.0	
Axisymmetric Coil Systems	0.1	
Non-Axisymmetric Coil Systems	6.7	
In-Vessel Measurement Systems	1.4	
Auxiliary Heating		1.4
Neutral Beam Injection	1.0	
Electron Cyclotron Heating	0.4	
Fueling and Vacuum Systems		0.1
Power Systems		3.1
Utility Systems		0.3
Central I&C and Data Acquisition		2.0
Diagnostics		1.3
Site Preparation and Facility Startup		1.4
Project Managment & Support		4.4
Project Management & Control	1.5	
Project Physics	1.5	
Systems Engineering	1.4	
Subtotal		27.9
Contingency @25%		7.0
Total Project Cost (TPC)		34.9

Table 2-3. NCSX Budget Profile Through FY-2005 (FY-99 M$)

	Fiscal Year						
	1999	2000	2001	2002	2003	2004	2005
Construction TPC	3.5	9.0	12.0	10.4			
NCSX Operations	0.4	0.7	1.3	4.0	15.0	15.0	15.0
NCSX Enhancements	0.0	0.3	0.7	1.5	5.0	5.0	5.0
NCSX Total	3.9	10.0	14.0	15.9	20.0	20.0	20.0

References

[1] H.P. Furth and C.W. Hartman, Phys. Fluids **11**, 408 (1968).

[2] U. Stroth et al, Nucl. Fusion **36**, 1063 (1996).

[3] S. Sudo et al, Nucl. Fusion **30**, 11 (1990).

[4] N. Sauthoff et al, IAEA 1990 - Washington, Vol. 1, 709, (1991, Vienna).

[5] B. LeBlanc, private communication.

[6] R. Goldston, private communication.

[7] S. Murakami et al, Nucl. Fusion **36**, 359 (1996).

[8] J. Nuehrenberg, W. Lotz, and S. Gori, in *Theory of Fusion Plasmas*, E. Sindoni, F. Troyon and J. Vaclavik eds., SIF, Bologna, 1994.

[9] P.R. Garabedian, *Phys. Plasmas* **3**, 2483 (1996).

[10] R. Goldston, Innovative Confinement Concepts Workshop 1997, March 3-6, 1997, Santa Monica, Calif.

[11] W VII-A Team, *Nucl. Fusion* **20**, 1093 (1980).

[12] D. C. Robinson and T. N. Todd, *Phys. Rev. Letters* **48**, 1359 (1982).

[13] V. V. Drozdov,. M. Yu. Isaev, M. I. Mikhailov, V. D. Pustovitov, and V. D. Shafranov, in Plasma Physics and Controlled Nuclear Fusion Research 1988 (Proc. 12th Int. Conf. Nice, 1988), Vol. 2, IAEA, Vienna (1989) 611.

[14] J. L. Johnson et al., Phys. Fluids **1**, 281 (1958); R. M. Sinclair et al., Phys. Fluids **8**, 118 (1965); K. Matsuoka et al., Nuclear Fusion **17**, 1123 (1977).

[15] M. I. Mikhajlov and V. D. Shafranov, Nucl. Fusion **30**, 413 (1990).

[16] J.N.Talmadge and W.A. Cooper, Phys. Plasmas **3**, 3713 (1996).

[17] A. Reiman, L. P. Ku, D. Monticello, C. Nuehrenberg, and W. Cooper, *Plasma Physics Reports* **23**, 472 (1997).

[18] S. P. Hirshman and J. C. Whitson, Phys. Fluids 26, 3553 (1983).

[19] Watanabe, K., et al., Nuclear Fusion **35**, 335 (1995).

[20] J. Nuehrenberg and R. Zille, in **Theory of Fusion Plasmas**, A. Bondeson, E. Sindoni, and F. Troyon eds. (Soc. Ital. Fisica -- Editrice Compositori, Bologna, 1988) pp. 3--23.

[21] D. A. Garren and A. H. Boozer, *Phys. Fluids* B **3**, 2805 (1991).

[22] H.E. Mynick, *Plasma Physics Reports* **23**, 547 (1997).

[23] U. Stroth, Murakami, R.A. Dory, H. Yamada, F. Sano, T. Obiki, National Institute for Fusion Science Report NIFS-375 (1995).

[24] W.A. Cooper et al., Phys. Plasmas **3**, 275 (1996).

[25] H. Furth and C. Hartman, Phys. Fluids **11**, 408 (1968).