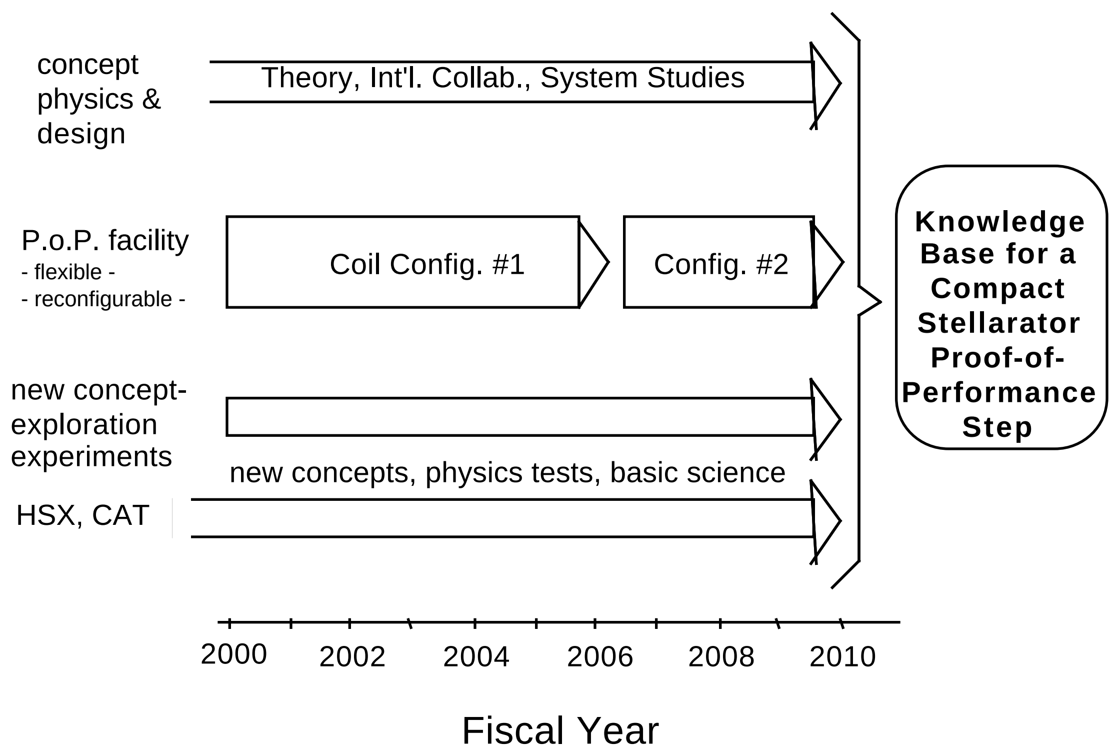
STELLARATOR PROOF

OF PRINCIPLE PROGRAM

Compact-Stellarator Proof-of-Principle Program



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U.S. STELLARATOR PROOF-OF-PRINCIPLE PROGRAM PLAN

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EXECUTIVE SUMMARY

A unique opportunity exists for the U.S.advance the world fusion program by developing compact stellarators. This would capitalize on exciting recent theoretical and experimental developments that have provided the foundation for this initiative, and would fill an important gap in the world stellarator program. The U.S. stellarator community proposes a proof-of-principle (PoP) *program* to develop the knowledge base needed for compact, high-beta, good-confinement stellarators and a decision on a compact stellarator Proof-of-Performance step that would allow extrapolation to burning plasma conditions. The proposed program is based on development of a hybrid magnetic configuration that combines the best features of stellarators (external control; immunity to disruptions; and no need for current drive, feedback stabilization, rotation drive, or a close conducting wall) and advanced tokamaks (compact, high beta, good confinement). The result should be a fusion power system considerably more attractive than either the advanced tokamak or the large-aspect-ratio stellarator.

A Unique Opportunity Exists for High-Performance Compact Stellarators. Stellarators have the potential for an attractive reactor featuring steady-state disruption-free operation, low recirculating power, and good confinement and beta. The new Large Helical Device (LHD) and Wendelstein 7-X (W7-X) are ~$0.5-1 billion-class stellarator experiments designed for a level of performance that allows extrapolation to burning plasma devices. They will provide data on divertors, high-power heating, steady-state operation, and superconducting coils that will be relevant to all stellarator concepts. Thus the international programs in Japan and Europe have judged the stellarator ready not only for Proof-of-Principle, but for Proof-of-Performance tests. However, the most developed foreign stellarators extrapolate to large reactors (for example, the W7-X-based HSR has major radius $R_0 = 24$ m). These stellarators, including smaller experiments, have plasma aspect ratios ranging from 5 to 11; low aspect ratios ($A = R_0/\langle a \rangle < 5$) are unexplored. Here $\langle a \rangle$ is the average radius of these inherently noncircular plasmas. None of the foreign stellarators takes advantage of the bootstrap current, magnetic symmetry, or quasi-omnigeneity to create a compact stellarator configuration. This gap in the world stellarator program can be filled by the U.S.

Two promising transport optimization strategies for compact stellarator design (with $A = 2$-4) have been developed theoretically: quasi-axisymmetry (QA) and quasi-omnigeneity (QO). Quasi-symmetric stellarators conserve a component of the canonical momentum (as do tokamaks) and have neoclassical transport properties that are tokamak-like. Quasi-axisymmetric stellarators can have aspect ratios and bootstrap currents typical of tokamaks, so they resemble tokamak-stellarator hybrids. Like tokamaks, they can have a deep magnetic well and high beta limits for ballooning, even at a low aspect ratio. Although the last closed flux surface appears non-axisymmetric in real space, the Fourier spectrum of $|B|$ in magnetic coordinates (Boozer coordinates), upon which the particle drift orbits and neoclassical transport depend, has a dominant axisymmetric component with non-axisymmetric components of only a few percent at the plasma edge. The U.S. Helically Symmetric Stellarator (HSX) will be the first experimental test and exploration of quasi-symmetry.

The QO concept: (1) achieves reduced neoclassical losses by approximately aligning the collisionless trapped particle drift orbits with the magnetic surfaces; and (2) provides a larger fraction of the rotational transform by external coils, reducing the fraction of the rotational transform that is created by the bootstrap current. This may ease startup, reduces the sensitivity of the equilibrium to changes in the bootstrap current, and may reduce susceptibility to disruptions. The variation of the field strength within a magnetic surface can be more complicated than in quasi-symmetric configurations since no particular symmetry is imposed. The low-aspect-ratio QO configuration has similar features to the large-aspect-ratio ($A \approx 11$) W7-X configuration. The large non-axisymmetric terms in the |B| spectrum and the lower fraction of bootstrap current distinguish QO stellarators from QA-stellarators.

Both the QA and QO concepts make use of the bootstrap current, but to different degrees, to create a configuration with <1/3 the aspect ratio of the currentless W7-X stellarator. The new QA and QO stellarator configurations are aimed at volume-average betas ($\langle\beta\rangle$) at least as high as the 5% value projected for LHD, W7-X, and the ARIES-RS tokamak reactor. Both look attractive for compact stellarator reactors, but each has distinct complementary advantages. Both must be developed experimentally to establish the needed scientific base for the program's ultimate success. A determination of the optimum strategy to pursue is one of the U.S. stellarator program's goals.

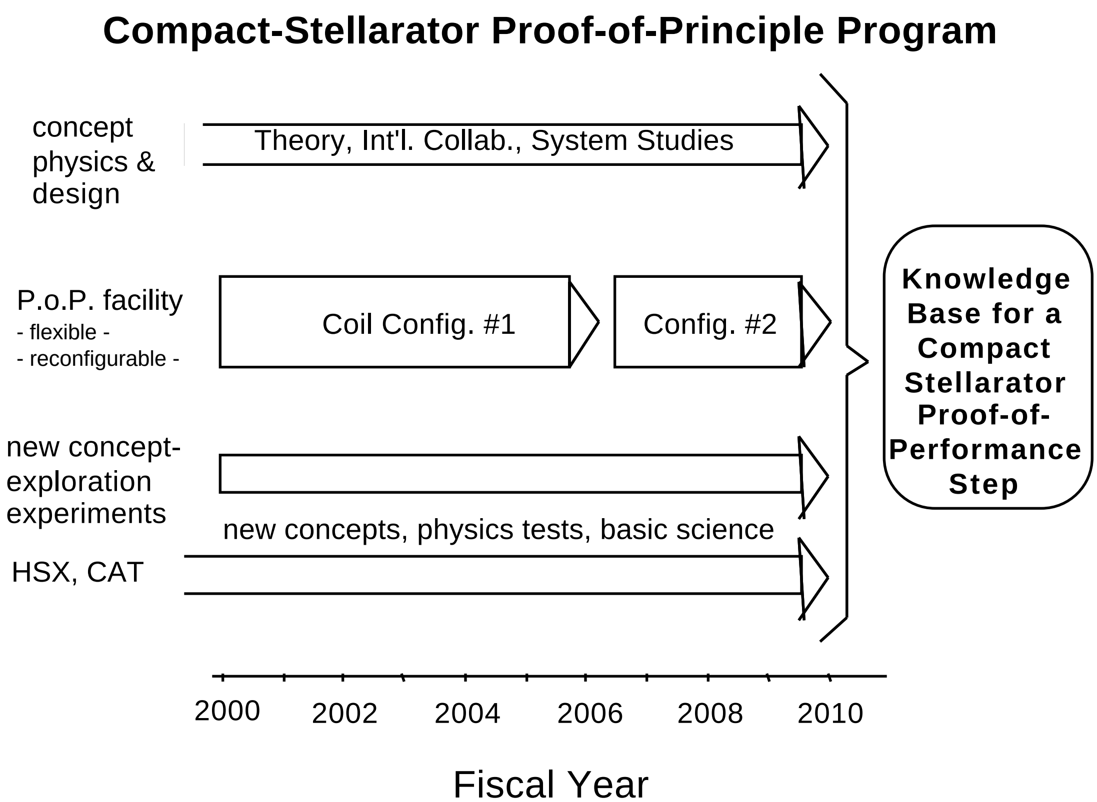

Fig. 1. Road map for the proposed stellarator PoP program.

The Stellarator PoP Program.

The program that the U.S. stellarator community proposes, illustrated schematically in Fig. 1, follows the model for PoP programs defined in the 1996 FESAC/SciCom Alternates Review. It is a coherent, integrated program to develop the knowledge base needed to assess compact, high-beta, good-confinement stellarators and consists of: (1) a new proof-of-principle facility, the National Compact Stellarator Experiment (NCSX); (2) a new concept exploration experiment,

(3) the existing Helically Symmetric Experiment (HSX) at the University of Wisconsin and a modification of the Compact Auburn Torsatron (CAT) at Auburn University; (4) stellarator theory focusing on concept optimization and key stellarator physics issues; (5) collaboration with the international stellarator program in specific areas; and (6) system studies to guide concept optimization tradeoffs. The NCSX PoP facility will be reconfigurable to ensure that experimental tests of improved configurations emerging from the program as a whole can be tested expeditiously. All elements of the program are necessary to adequately develop the compact stellarator concept.

The six PoP program elements cross-link with each other to provide a well-integrated program. The NCSX PoP facility would be the focus of the PoP program. It will be reconfigurable to ensure that experimental tests of the new developments arising from the total stellarator program can be conducted expeditiously. HSX, CAT, and the new concept exploration experiment will allow tests of new optimized configurations, basic concept optimization physics studies, and extension of 3-D plasma science. The results from these experiments and from international collaborations, theory, and system studies will help optimize the configuration to be developed in later phases of the PoP facility. In order to minimize cost, it is planned to construct the PoP facility by modifying an existing device, the PBX-M tokamak, and using its supporting infrastructure. The QA concept is likely somewhat more easily compatible with the PBX-M constraints, so it has been chosen as the initial PoP configuration. The new concept exploration experiment, the Quasi-Omnigenous Stellarator (QOS), will be proposed to test the basic principles of the QO optimization strategy.

This program will build on the substantial existing data base in stellarators and tokamaks and will make important contributions to the world fusion program. If this plan is carried out, the resulting knowledge base will be sufficient to permit comparisons with steady-state tokamak-based power plant designs and will provide a basis for proceeding to the following step, a Proof-of-Performance program to study more reactor-relevant plasmas in a compact stellarator configuration.

The new experimental elements described here, a PoP facility, a new Concept Exploration experiment, and modifications to the existing CAT device, are not presently at the stage of fully converged designs, appropriate for discussion at the level of a Physics Validation Review or a Conceptual Design Review. Instead the coordinated scientific goals of the experiments have been laid out, and the various elements of the designs have been brought to the point where the community has confidence that fully converged designs needed for machine proposals can be developed on the necessary schedule. For example the listed budgets imply conceptual design reviews in about one year, and start of detailed design and construction in FY 2000.

The PoP Facility. The NCSX will meet the 1996 FESAC/SciCom criteria for a PoP experiment. It will provide sufficient plasma performance and machine capability for integrated testing of a compact stellarator configuration with high beta and bootstrap currents that can form the basis for extrapolation to more reactor-relevant performance. The PBX-M vacuum vessel constraint leads to a quasi-axisymmetric (QA) plasma configuration with $A \approx 3.3$ for NCSX, which will provide the basic physics data needed for lower-aspect-ratio designs, e.g., the $A \approx 2.1$ QA configurations which may provide greater compactness. Scientific questions to be resolved with NCSX include:

- Can a high-beta configuration, including bootstrap currents and external transform, avoid disruptions?
- What are the beta limits and limiting mechanisms?
- Can neoclassical transport be reduced by proper configuration design?

- Can turbulent transport be controlled, leading to enhanced global confinement? (for example, by flow shear or magnetic configuration)
- Can transport and stability be controlled through external magnetic configuration control?
- Are neoclassical islands and tearing modes suppressed by bootstrap current and the proper choice of magnetic shear, as indicated by theory and tokamak experiments?

NCSX conceptual design will be completed in mid-FY-99; construction will start in FY-2000. The total project cost (TPC) is estimated at $35M. Operation is planned to start in FY-2003 at a cost of ≈$20 million per year for facility operations, physics research, and facility enhancements. Research preparation activities will build up during the construction period (see table below) to facilitate an effective transition into operations.

New Concept Exploration Experiment. A new concept exploration experiment, QOS, is needed to test the basic optimization principles of quasi-omnigeneity, which complements the quasi-symmetry being tested in HSX and NCSX. The QOS experiment (with A ≈ 3.6) is needed to: (1) provide an initial data base on QO-specific issues that can feed into the optimum design of a PoP-level QO configuration which could be tested as the second magnetic configuration of the PoP facility, and (2) broaden the scientific base provided by the QA PoP and HSX into low-aspect-ratio non-symmetric confinement configurations. The primary focus of the proposed research program is

(1) reduction of neoclassical transport via nonsymmetric quasi-omnigeneity, and the effect of radial electric fields on confinement;
(2) reduction of energetic orbit losses in non-symmetric configurations;
(3) reduction of the bootstrap current (cancellation due to different B harmonics), its relative independence of β, and its compatibility with confinement improvement and the QO optimization;
(4) exploration and development of methods to affect anomalous transport, such as by producing sheared $E \times B$ flow, and understanding flow damping in non-symmetric configurations.

The Total Project Cost is ≈$6.5 million, similar to that of HSX. Operating costs would be ≈$2.5 million per year.

Existing Concept Exploration Experiments. The existing HSX and the Compact Auburn Torsatron (CAT) provide important information at moderate aspect ratio. The HSX, which will begin operation this year, will be the largest experimental element in the U.S. stellarator program. The quasi-helically symmetric (QHS) HSX has a single dominant helical component in the magnetic field spectrum. The unique characteristic of HSX is that it has $A = 8$, yet the toroidal curvature is the same as a device with $A > 400$. Since neoclassical transport depends only on the spectral components of the magnetic field amplitude, HSX has transport analogous to that of a tokamak. HSX has a high "effective" rotational transform (≈3) which provides several benefits: small Pfirsch-Schlüter currents, leading to high *equilibrium* beta limits; small poloidal gyroradius, leading to very good confinement of trapped particles; and neoclassical transport that can be smaller than in a comparable tokamak.

The primary objectives of the HSX program are to: (1) investigate the reduction of neoclassical transport in QHS configurations and the role of anomalous transport; (2) demonstrate a reduction in the direct loss of deeply trapped particles due to QHS; and (3) show that QHS leads to decreased viscous damping of rotation on a flux surface. The initial resources needed for this effort are $1.6 million per year (the current level); further resources will be required as the program evolves.

An upgrade to the existing CAT experiment is proposed for study of disruptions and stability in current-carrying plasmas in a medium-aspect-ratio (A = 5.6) stellarator. The upgrades consist of (1) addition of an ohmic transformer and capacitor bank power supply to drive a plasma current for stability studies, (2) an increase in the magnetic field to obtain more relevant plasma parameters, and (3) implementation of ICRF plasma generation and heating to provide target plasmas for current stability studies. The resources needed for this effort are $0.32 million (vs. the current ≈$0.2 million) per year, plus $0.4 million for upgrades in the first 2-3 years.

Theory. Stellarator theory has three fundamental objectives in the context of the Proof-of-Principle program: (1) extension of the framework for the interpretation of experiments, (2) development of techniques for extrapolation of the results from the Proof-of-Principle to the Proof-of-Performance level of experiments, and (3) invention of even better stellarator configurations. A secondary objective is the general development of three-dimensional plasma theory. The future development of the compact-stellarator knowledge base, and the world-wide development of the stellarator, will continue to require a strong theory effort, addressing: (1) MHD equilibrium, (2) magnetic island formation, (3) MHD stability, (4) neoclassical transport and drift orbits, (5) microstability and anomalous transport, (6) divertor and edge physics, (7) waves and heating, and (8) optimization of magnetic configurations. Theory progress in all these areas will be important for the new U.S. experiments testing compact-stellarator concepts as well as existing experiments such as HSX and the large foreign experiments. The cost will be about $3.5 million (vs. the current ≈$1.2 million) per year . Much of the needed increase can be achieved through changes in the research focus of existing theorists which will occur naturally as the U.S. stellarator program grows, but graduate students and postdoctoral fellows will also be needed.

International Collaboration. Collaboration with the larger international stellarator program on selected topics is an important element of the U.S. stellarator PoP program because it provides information needed for stellarator concept improvement that is not otherwise available in the U.S. program. The wide range of stellarator configurations accessible on foreign stellarator experiments allows study of higher aspect ratio, degree of helical axis excursion, magnetic-island-based divertors, and the consequences of a modest driven plasma current. Physics issues of particular importance are heating, transport, enhanced confinement modes, beta limits, and particle and power handling.

Collaboration on theory development and on tools for concept optimization are also an important element of the U.S. stellarator program. The areas of most interaction are MHD equilibrium with magnetic islands (Japan), MHD stability (Germany, Switzerland, Spain, Japan), bootstrap current (Japan), and concept development (Germany, Japan).

The resources needed for the experimental collaborations are $1.5 million (vs. the current ≈$0.8 million) per year. The theory collaborations would be funded through the Theory budget.

System Studies. Integrated physics and engineering systems studies are important for assessing the reactor potential of compact stellarators and guiding the experimental program on the QA and QO devices. The capabilities needed have been developed in previous U.S. stellarator reactor studies and in the ARIES tokamak reactor studies. The earlier U.S. Stellarator Power Plant Study (SPPS) showed that the reactor size could be reduced from $R_0 \approx$ 24 m (for the W7-X-based HSR) to R_0 = 14 m for the SPPS reactor, based on an early compact-stellarator concept. New studies will explore further size reductions based on the concepts being tested in the compact-stellarator program.

The resources needed, averaging $1 million per year and alternating between 2 years of scoping studies at the $0.3 million per year level and 2 years of the more detailed ARIES-type studies with a specific QA or QO coil configuration at $1.7 million per year, would be funded as part of the ARIES program and *not* as part of the proposed program.

The PoP Program is Ready to Proceed. The extensive data base created by higher-aspect-ratio currentless stellarators and by advanced tokamaks provides much of the knowledge base and motivation needed for a compact-stellarator PoP program. Stellarators are now second only to the mainline tokamak in their level of physics understanding and parameter achievement. For example, W7-AS (with R_0 = 2 m, $\langle a \rangle$ = 0.18 m, A = 11) obtains T_e = 5 keV, T_i = 1.6 keV, β = 1.8%, and τ_E > 50 ms, in different operating regimes.

New theoretical tools and understanding gained in designing HSX and W7-X have led to the compact stellarator concepts with reduced neoclassical losses to be developed in the PoP program. Anomalous transport has been reduced in present stellarator experiments and transport reduction techniques developed for tokamaks can be applied to stellarators. The newer designs optimized to reduce neoclassical transport are particularly suited for confinement improvement due to electric field ($E \times B$) flow shear.

The increased understanding of stellarator physics has provided capable tools and new optimization strategies for designing high-performance stellarator plasmas. Codes are now available to calculate fixed- and free-boundary stellarator equilibria, evaluate magnetic island formation, assess ballooning and external kink-mode stability, calculate the bootstrap current profile, and calculate neoclassical transport losses in stellarator configurations. Sophisticated tools for designing coils for a given plasma configuration are indispensable for modern stellarator design. The physics tools that are essential for proceeding with the PoP program are in place.

Modern computational and manufacturing techniques have made possible the fabrication of complex stellarator coils to high precision, as demonstrated by successful construction of the modular-coil stellarators W7-AS and HSX. The complex W7-AS and HSX vacuum vessels were fabricated from flat plates and using explosive forming, respectively, now well-developed techniques. Thus, the engineering tools for the program are also in place.

Summary. This program will build on the substantial data base in stellarators and tokamaks and will make important contributions to the world fusion program. If the proposed plan is carried out, in ten years the resulting knowledge base will be sufficient to permit comparisons with steady-state tokamak-based power-plant designs and will provide a basis for a decision on proceeding to the next step, a proof-of-performance program to study more reactor-relevant plasmas in a compact stellarator configuration, possibly with D-T capability.

The proposed program is an exciting opportunity for the U.S. to develop the compact stellarator concept as part of the innovative Fusion Energy Science Program.

Cost and Schedule. The program elements and associated costs are listed in Table 1.

Table 1. Compact-Stellarator Proof-of-Principle Program Budget Profile

	FY 98	FY 99	FY 00	FY 01	FY 02	FY 03	FY 04	FY 05
NCSX Proof-of-Principle Experiment								
Construction TPC ($35M)		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	1.8	3.9	10.0	14.0	15.9	20.0	20.0	20.0
QOS Concept Exploration Experiment								
Construction TPC ($6.5M)		0.5	1.1	1.8	2.1	1.0		
QOS Operations		0.1	0.1	0.1	0.1	1.0	1.3	1.5
QOS Enhancements		0.0	0.1	0.1	0.1	0.4	1.2	1.0
QOS Total	0.3	0.6	1.3	2.0	2.3	2.4	2.5	2.5
Helically Symmetric Exper.	1.6	1.6	1.7	1.9	2.0	2.0	2.0	2.0
Compact Auburn Torsatron	0.2	0.5	0.5	0.3	0.3	0.3	0.3	0.3
Theory	1.2	2.0	2.5	3.0	3.5	3.5	3.5	3.5
International Collaboration	0.8	1.0	1.3	1.5	1.5	1.5	1.5	1.5
Program Total	5.9	9.6	17.3	22.7	25.5	29.7	29.8	29.8

1. COMPACT STELLARATORS

The U.S. stellarator community proposes expansion of the national stellarator program to a Proof-of-Principle level to build upon recent innovations in stellarator design and advances in toroidal plasma physics understanding. The goal of this program is to develop the knowledge base needed to evaluate compact high-beta confinement-optimized stellarators as attractive fusion power plants and motivate their further study at the Proof-of-Performance scale. These stellarator designs offer the combined advantages of the advanced tokamak, as envisioned in the ARIES-RS study [1], with the strong advantages of stellarators: lack of disruptions and no need for external current drive. If successful, this program will produce plasmas similar to those needed to project to ARIES-RS, but with much lower recirculating power and without the need for disruption handling, feedback stabilization, or rotation drive.

This stellarator program should be a key component of the U.S. Fusion Energy Sciences Program, complementary to the strong stellearator programs in Japan and Europe, which are not focussed on compact designs. The demonstrated advantages of stellarators potentially provide ready solutions to some of the most challenging problems in achieving attractive power plants, like ARIES-RS. As such, stellarators present an attractive development path for demonstrating that such performance is achievable. It is important to demonstrate this expeditiously to establish the credibility of fusion as a possible solution to the energy challenges of the next century.

1.A. Stellarator Advantages

A stellarator plasma is not toroidally symmetric and the choice of its three dimensional (3-D) shape provides significant design freedom that can be used to optimize the configuration for desired fusion performance or for studying particular plasma physics properties. In a reactor, the confining poloidal magnetic field is generated using modular coils, similar to tokamak toroidal field coils but with out-of-plane deformations. Stellarator configurations have distinct advantages:

1. The need for an externally driven plasma current is eliminated or greatly reduced, reducing the recirculating power in a reactor.

2. Disruptive discharge termination and quench of the plasma current are not observed in stellarator experiments, even at the predicted β limit. Even in stellarators experiments with large plasma currents (edge iota > 0.6), the external fields stabilize the configuration preventing disruptions when the externally generated transform is at least 0.15 [2]. This allows stellarators to robustly access their full β limit in steady state.

3. Empirically, the density in stellarators is limited only by power balance, not by disruptions or edge instabilities as in tokamaks. For equivalent configurations, the empirical density 'limit' is higher in stellarators, [3] allowing an optimal reactor burn-point. Since edge current drive is not required (as in an advanced tokamaks), operation with high edge density and radiating layers can be used.

4. The 3-D shaping provides control of the rotational transform ($\iota\!\!\!-$ = 1/q) and shear profiles externally, allowing designs with ‘reversed shear’ across the entire profile. This is not possible in axisymmetric configurations like tokamaks, and increases MHD stability and stabilizes tearing modes and islands, so long as the bootstrap current and d$\iota\!\!\!-$/dr are both either positive or negative. Experimental evidence from tokamaks confirms this effect, in that neoclassical tearing modes are never

observed inside of the shear reversal point in reversed-shear plasmas, but are frequently_observed in standard shear cases, and outside of the shear-reversal point in reversed-shear plasmas.

Due to these advantages, including solutions to many problems challenging tokamak reactors, stellarator configurations offer the possibility of an attractive development path for fusion energy. This is why stellarators are an important part of the world fusion development program. However, at present the world program is focused on currentless, high-aspect-ratio ($A > 5$) configurations that extrapolate to large reactors. The incorporation of bootstrap current and new transport optimization strategies can lead to more compact high-beta stellarator designs with lower aspect ratios, which would make available a wider range of reactor design solutions. There is a need and an opportunity to develop these ideas for compact stellarators, which is the aim of this proposal.

1.B. Innovation Has Created an Opportunity: Compact Stellarators

Earlier stellarator designs and experiments led to significant concerns about confinement at low collisionality, the achievable beta limit, coil complexity, and the aspect ratio (and thus reactor size). Through continuing research, these issues have been understood and largely resolved.

Development of new theoretical tools has led to stellarator designs with good neoclassical confinement. Earlier designs had large helical-ripple-induced neoclassical losses at low collisionality. These losses are greatly reduced (in some cases to below those in a comparable tokamak) in newer stellarator designs such as the quasi-helically-symmetric approach used in HSX [4], and the quasi-axisymmetric [5] and nonsymmetric quasi-omnigeneous configurations [6] being developed for low-aspect-ratio stellarator candidates in the U.S.

Quasi-symmetric stellarators [5] conserve a component of the particle canonical momentum (as do tokamaks) and have neoclassical transport properties that are tokamak-like. Although the geometry of a quasi-symmetric stellarator is fully three dimensional, the field strength has a continuous symmetry, either toroidal or helical. Exact quasi-symmetry is not possible, but quasi-symmetry can be approximated with sufficient accuracy to insure excellent neoclassical confinement. The HSX stellarator [4], under construction at the University of Wisconsin, will be the first experimental test of quasi-symmetry and is of the quasi-helical type. The quasi-helically symmetric (QHS) approach can have neoclassical transport even better than a tokamak because the high effective transform can lead to very small drift orbit widths. Quasi-axisymmetric (QA) stellarators can have aspect ratios and bootstrap currents typical of tokamaks, so they resemble tokamak-stellarator hybrids. Like tokamaks, they can have a deep magnetic well and high beta limits for ballooning, even at low aspect ratio. Although the last closed flux surface is not toroidally symmetric in real space, the Fourier spectrum of $|B|$ in magnetic (Boozer [7]) coordinates, upon which the particle drift orbits and neoclassical transport depend, has a dominant axisymmetric component with non-axisymmetric components of only a few percent at the plasma edge. QA configurations differ from QHS configurations in that QA configurations have smaller plasma aspect ratio (typically ~3), higher beta limits, larger bootstrap current, and the rotational transform produced by the bootstrap current is in the same direction as that produced by the external coils in the QA configuration (thus stabilizing islands), but in the opposite direction in the QHS configuration.

Non-symmetric omnigeneous stellarators [6], or quasi-omnigeneous (QO) configurations achieve acceptable neoclassical losses by approximately aligning the surfaces on which the approximate second adiabatic invariant J* is constant with the magnetic surfaces. The variation of the field strength within a magnetic surface can be more complicated than in quasi-symmetric configurations since no particular symmetry is imposed. Examples of stellarator designs of this general type are the large-aspect-ratio W7-X and the SMall Aspect Ratio Toroidal Hybrid (SMARTH). QO stellarators have reduced bootstrap currents, due to cancellations between the toroidal and the helical or bumpy field terms in the |B| spectrum, and can be designed to localize the trapped particles in regions of good curvature, reducing turbulence drive. The large non-axisymmetric terms in the |B| spectrum and the smaller fraction of bootstrap current distinguish QO stellarators from QA stellarators.

Anomalous transport has been reduced in present stellarator experiments (improved confinement modes). The international stellarator confinement scaling (ISS95 [8]), based on a large set of data from the world stellarators, is given by

$$\tau_E^{ISS95}(\mathrm{s}) = 0.079\langle a(\mathrm{m})\rangle^{2.21} R(\mathrm{m})^{0.65} P(\mathrm{MW})^{-0.59} n(10^{19}\mathrm{m}^{-3})^{0.51} B(\mathrm{T})^{0.83} \iota(2r/\langle a\rangle)^{0.4}. \quad [1\text{-}1]$$

The energy confinement time $\tau_E = H\tau_E^{ISS95}$, where H is the confinement improvement factor. For the low-shear W7-AS, which is relevant to the low-shear QA and QO configurations, H averages 1.3-1.4 and is 2.5-3 for discharges with good wall conditioning and low recycling, which has led to $\tau_E > 50$ ms [9]. Techniques developed for confinement improvement in tokamaks can be applied to stellarators; the new designs optimized to reduce neoclassical transport should be particularly suited for confinement improvement due to electric field ($E \times B$) flow shear.

Both the QA and QO optimization strategies have significant bootstrap currents at low aspect ratio. This provides additional rotational transform that raises the equilibrium and ballooning stability limits and relaxes some of the current density and torsion constraints on the coils, improving the designs. Combined with reversed shear, the bootstrap current stabilizes islands and tearing modes, making the flux surface configuration more robust. However, the parallel current can destabilize MHD modes, particularly the kink mode, and thus could reintroduce the disruptions observed in tokamaks. While the linear stability of these modes can be calculated and included in the design, the non-linear saturation and its consequences have not been characterized and experimentally verified. Experiments on W7-A showed immunity from disruptions with modest amounts of externally generated rotational transform (~0.15) in addition to that from the Ohmic current (~0.5) [2]. Related results were observed in the CLEO stellarator [10], where a modest amount of reversed external transform completely stabilized disruptions in low-q tokamak operation. These results were obtained at high aspect ratio and low beta. Experiments including significant bootstrap current are needed to investigate the role of these instabilities and the empirical range of disruption-free operation at low aspect ratio and high beta.

The new compact stellarator concepts are designed using theoretical predictions of stability limits to Mercier, ballooning, and kink modes. This is a conservative approach because present stellarators exceed simple estimates of beta limits, and novel configurations show promise of higher beta. The simple equilibrium beta limit criterion based on the shift δ of the magnetic axis equal to half the average plasma radius $\langle a\rangle$ has been exceeded in W7-AS where $\delta/\langle a\rangle = 2/3$ was obtained without a significant change in the plasma behavior, even when the outer 1/3 of the plasma was theoretically resistively unstable [11]. In addition, the Mercier instability criterion has been exceeded over most of the plasma radius in the CHS experiment at the highest volume-average beta values achieved in stellarators ($\beta = 2.1\%$) without a significant change in plasma confinement [12]. In both cases the achievable beta is limited by the available heating power and transport rather than by an observed stability limit. The new QA and QO stellarator designs will be designed to have equilibrium and stability beta limits >5%.

New computational techniques and modern numerically controlled machines have made tractable the fabrication of complex stellarator coils to a high degree of accuracy, as demonstrated by the successful construction of the modular-coil stellarators HSX (R_0 = 1.2 m, B_0 = 1.35 T) and W7-AS (R_0 = 2 m, B_0 = 3 T). These techniques are now being used for the large superconducting-coil W7-X stellarator (R_0 = 5.5 m, B_0 = 3 T). The complex W7-AS vacuum vessel was also fabricated from flat plates using similar techniques and explosive forming was used to fabricate the HSX vacuum vessel.

The remaining concern, that stellarator power plants may be too large, has now been addressed in concept by the U.S. program. The most developed confinement-optimized stellarator reactor concept, HSR [13], has R_0 = 22-24 m. Recent theoretical developments in the U.S. of new confinement-optimized configurations hold the prospect that a low-aspect-ratio stellarator can be developed with good confinement and higher beta. This should result in a more compact stellarator reactor. The U.S. can have the leading role in development of compact stellarator configurations.

At present, both strategies for transport optimization for compact stellarators (QA and QO) are promising and have complementary advantages. The QA designs have a direction of quasi-symmetry, allowing plasma rotation and rotational control of the radial electric field for turbulence suppression. Both make use of the bootstrap current, but the QO designs have reduced bootstrap currents simplifying configuration control. Both strategies must be explored experimentally to develop the scientific basis for the program's ultimate success.

1.C. Key Issues for Compact Stellarators

There are a number of key issues in the design of compact stellarators that will determine their attractiveness for future power plant designs.

1) Can a $\beta \sim 5\%$ configuration, with self-consistent bootstrap currents and external transform avoid disruptions? What are the configuration requirements (e.g. edge shear) to avoid disruptions?
2) Can neoclassical transport and orbits losses be reduced sufficiently by the QA and QO optimization strategies?
3) Can turbulent transport be controlled to give sufficient confinement for an attractive reactor, either through flow-shear control or magnetic configuration design?
4) What is the ultimate beta limit and the limiting mechanisms?
5) Is there a workable edge design giving control of particle and heat exhaust in a reactor?

The successful resolution of these issues must be experimentally explored and demonstrated in order for the compact stellarator concept to go forward. Issues (1)–(4) are the focus of the experimental part of the proposed program. Issue (5), the design of a reactor particle and heat exhaust system, is being actively studied in the international program and is an important area for international collaboration, and will be studied at reduced priority in the U.S. program.

I.E. Relationship to the International Stellarator Program

Development of the stellarator concept is being actively pursued in several countries. The largest new fusion facilities are stellarators: the Large Helical Device (LHD) now operating in Japan and the W7-X under construction in Germany are $0.5-1 billion facilities with superconducting coil systems. These experiments are supplemented by more moderate-size ($30-100 million scale) research facilities presently in operation in Japan (CHS and Heliotron E), Germany (W7-AS), and Spain (TJ-II). The proposed U.S. stellarator proof-of-principle program would complement the existing world stellarator program by adding an important element – research on compact stellarator configurations – that will be of value to all countries, but specially to the U.S. energy market, which is most sensitive to the unit size of future power sources.

LHD and W7-X are designed to demonstrate steady-state disruption-free stellarator operation and a level of performance (volume-average beta $\beta \geq 5\%$, ion temperature T_i ~ 10 keV, energy confinement time τ_E of hundreds of ms, etc.) that allows extrapolation to devices capable of burning plasma operation. The physics studies will focus on: ion heating and transport, neoclassical transport, the role of electric fields in confinement improvement, enhanced confinement modes, beta limits, practical particle and power handling, profile and configuration optimizations, and steady-state performance. These issues are crucial to demonstrating the viability of the stellarator confinement concept and extrapolation of stellarator performance. LHD and W7-X will provide unique data on both helical and magnetic-island-based divertors, high-power plasma heating, and superconducting-coil operation that will be relevant to all design approaches. Collaboration on these experiments will therefore be an important element of the U.S. stellarator program.

However, LHD and W7-X have plasma aspect ratios of 6 and 10.5, respectively, and extrapolate to very large reactors. In addition, W7-X was explicitly designed to minimize the bootstrap current, while LHD is expected to have bootstrap current much smaller than a comparable tokamak because of the combination of toroidal and helical curvature. Thus, neither will explore the potential advantages of designs with significant bootstrap currents.

The wide range of stellarator configurations accessible on the world's medium-scale stellarator experiments, CHS, Heliotron E, W7-AS, TJ-II, H-1, etc. allows study of a range of aspect ratios (from 5 to 11), helical axis excursions, magnetic-island-based divertors, and the consequences of a modest net plasma current. These experiments will make significant contributions to understanding of the stellarator concept and provide information of importance to the U.S. program. However, none of the foreign stellarator experiments incorporates magnetic symmetry or plasma current in their design strategies, and none has an aspect ratio smaller than 5, so these are important opportunities of special interest to the U.S. program and available for exploitation by innovative stellarator research in the U.S. program.

1.F. Synergy with the Tokamak Program

Stellarators share much of the physics basis of other toroidal systems, such as tokamaks, and can build upon and contribute to developments in the larger tokamak program. Many of the instabilities and configuration issues (e.g. transport control, divertors) expected in stellarators have been extensively studied in tokamaks. This shared physics basis aids the comprehensive analysis and evaluation of stellarator configurations, building upon the knowledge gained in both configurations. It also gives confidence that the flexibility of stellarators can be optimized to provide novel solutions to the challenging problems that have arisen in the study of the tokamak, and particularly the steady-state advanced tokamak, such as high-β disruption control and the need for edge current drive consistent with divertor operation.

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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 $\beta \sim 5\%$ 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 ~ 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

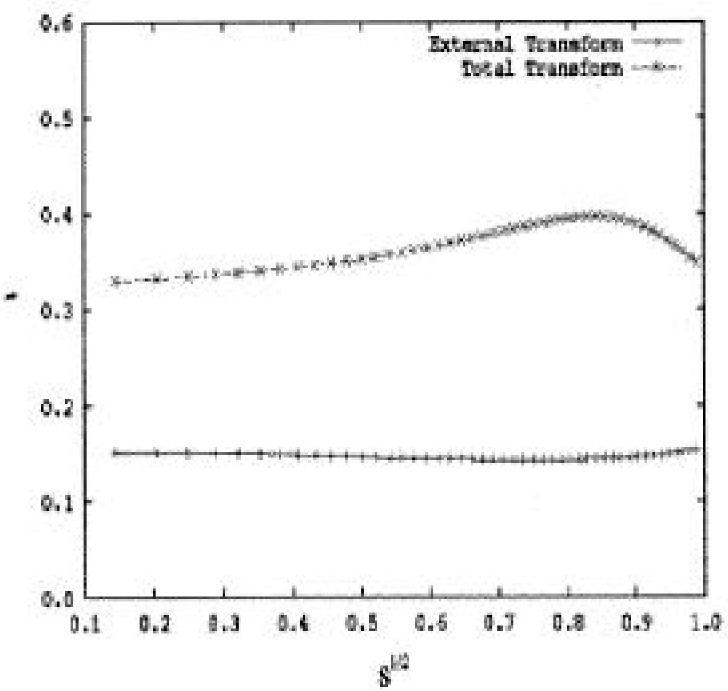

Fig. 2-1. Rotational transform profile for a two period configuration with R/ a 2.1, flat external transform profile. (Configuration I).

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.

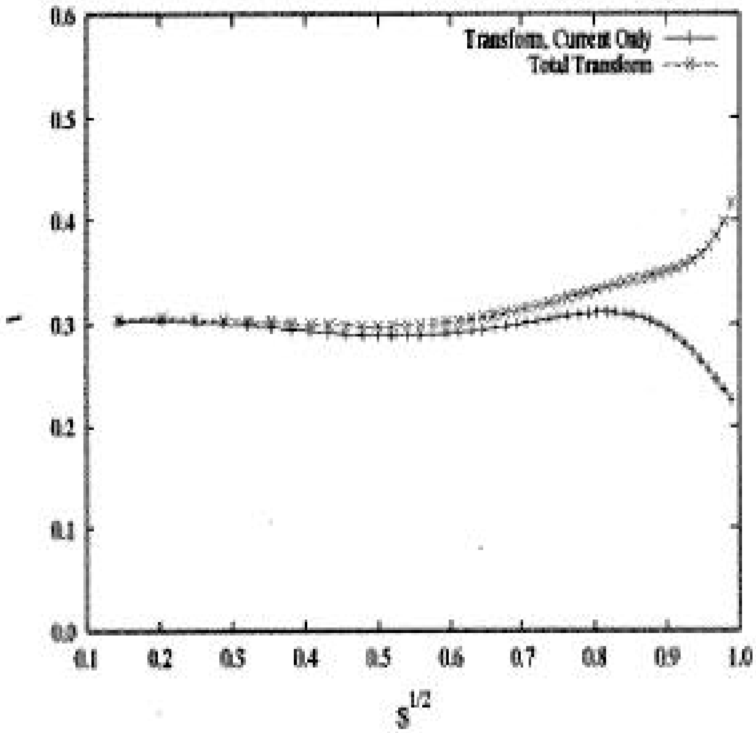

Fig. 2-2. Rotational transform profile for four period configuration with R/ a 2.1, monotonic rotational transform profile. (Configuration II)

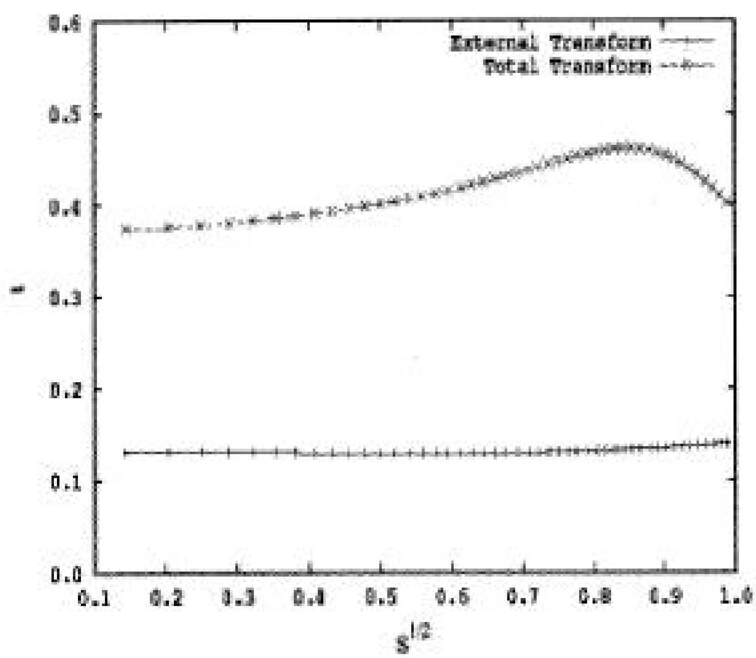

Fig. 2-3. Rotational transform profile for a three-period configuration with R/ a 3.3, flat external transform profile (Configuration III).

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

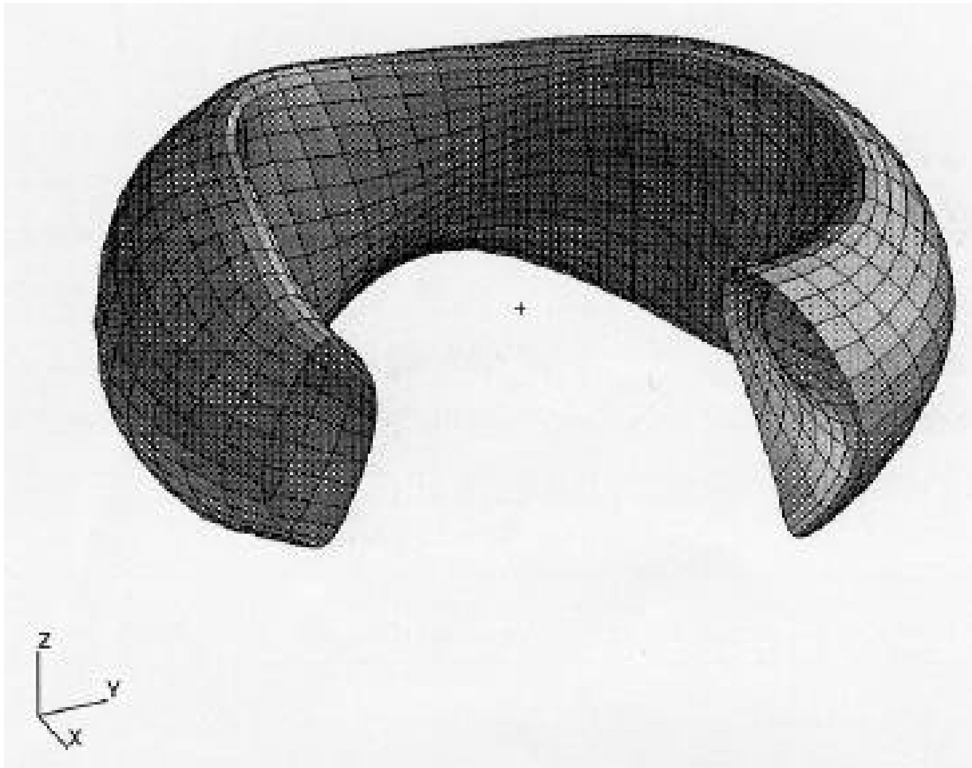

Fig. 2-4. Plasma boundary of quasi-axisymmetric configuration with R/ a 3.3.

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 $n \neq 0$ 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.

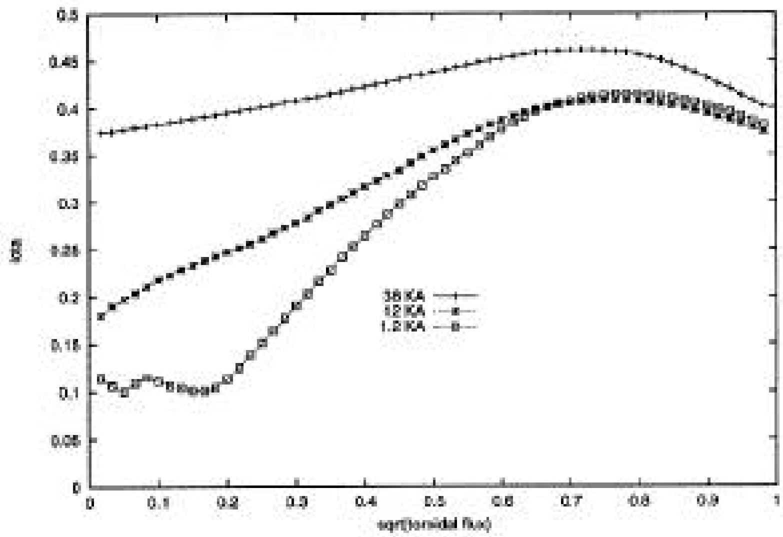

Fig. 2-5. Rotational transform profile for several values of the seed current in Configuration III.

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.

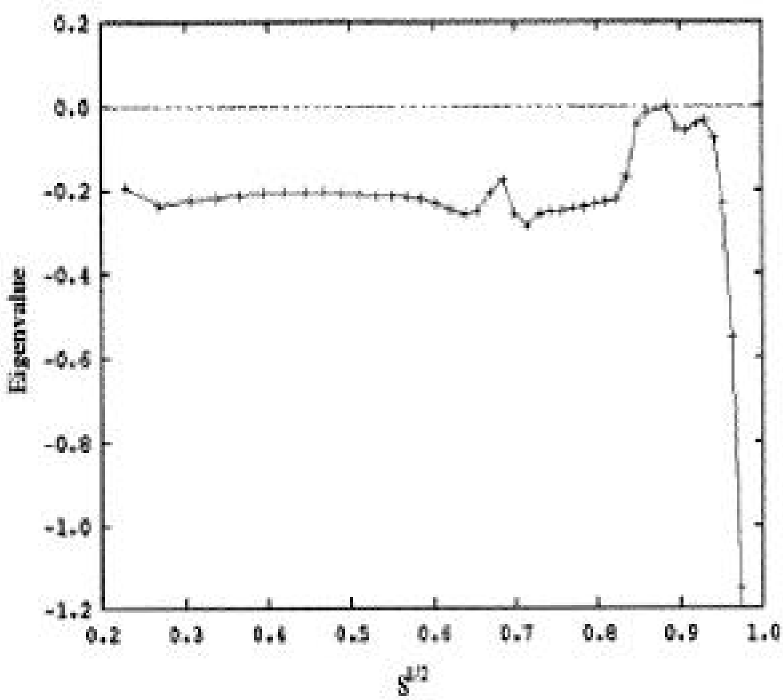

Fig. 2-6. Ballooning eigenvalues for Configuration III.

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 β = 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.

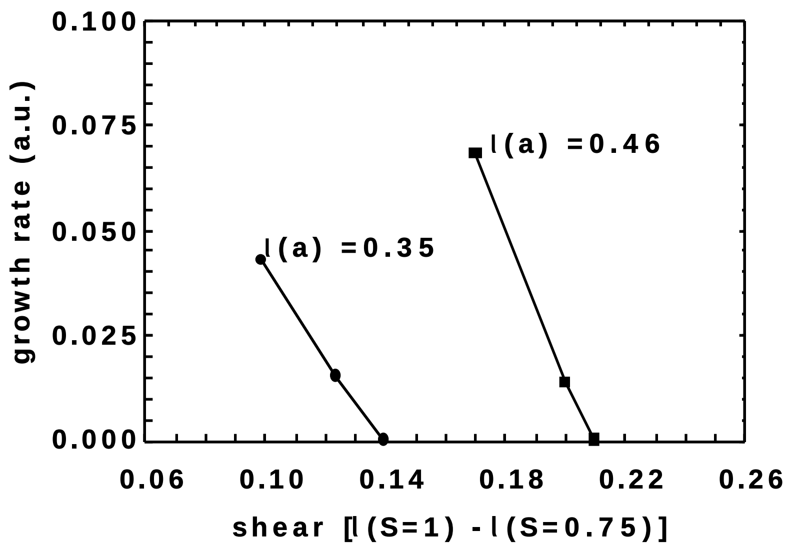

Fig. 2-7. Growth rate of the external kink mode as a function of shear at the plasma edge.

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.

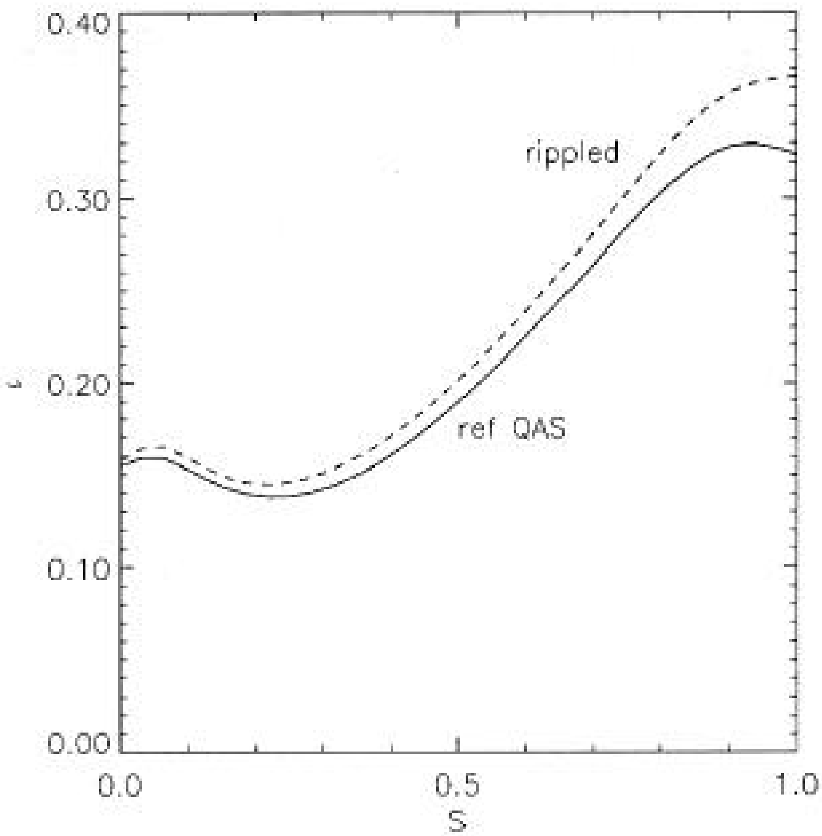

Fig. 2-8. Modification of the rotational transform profile of a quasi-axisymmetric configuration produced by a helical corrugation of the plasma boundary localized to the region near the outer midplane.

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.

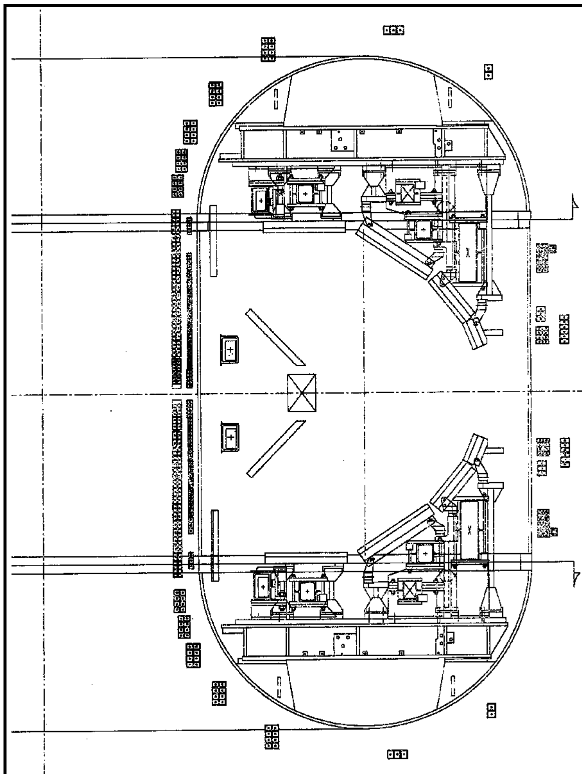
Figure 2-9 - Elevation View of PBX-M Tokamak

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.

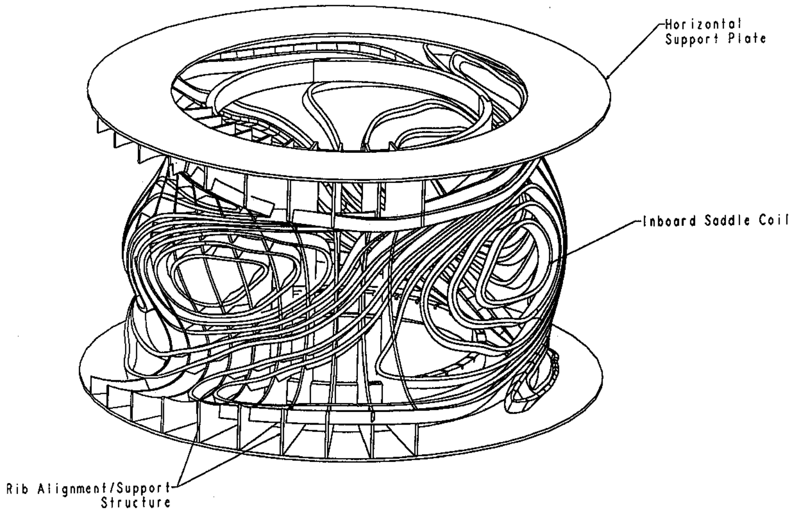

Figure 2-10 - Isometric View of Inboard Saddle Coils and In-Vessel Support Structure

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

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3. NEW CONCEPT EXPLORATION EXPERIMENT

3.1 A QO Concept Exploration Experiment Complements the QA PoP Test.

Two promising transport optimization strategies for a compact stellarator fusion power plant concept have been developed theoretically: quasi-axisymmetry (QA) and quasi-omnigeneity (QO). The determination of the optimal strategy to pursue is one of the program's goals. In assessments by the stellarator community, both concepts have been judged to have sufficient promise and developed enough for a proof-of-principle test. The QA concept, which is likely more compatible with the PBX-M vacuum vessel constraints, has been chosen for the initial proof-of-principle testing in order to minimize cost, as discussed in Sect. 2. But both the QA and QO concepts must be developed experimentally to provide the scientific base for the program's ultimate success.

A new concept exploration facility is needed to test QO optimization at low aspect ratio. Some physics issues for the QO approach [3-1] are generally similar to those for the QA configuration, but there are also important differences. The QO concept: (1) achieves reduced neoclassical losses by approximately aligning the particle drift surfaces with the magnetic surfaces, rather than relying on quasi-symmetry; and (2) provides most of the rotational transform by current in external coils, minimizing the fraction of the rotational transform that is created by the bootstrap current. This eases startup and reduces the sensitivity of the equilibrium to changes in the bootstrap current while reducing susceptibility to disruptions in a way similar to that for QA configurations. Conceptually, the QO configuration is closer to the currentless W7-X advanced stellarator configuration, but at 1/3 the aspect ratio, than it is to the QA configuration, which is closer to an advanced tokamak with a high-bootstrap-current-fraction combined with advantageous stellarator features, or to the quasi-helically symmetric HSX configuration.

Figure 3-1 shows flux surfaces at toroidal angles of 0° and 45° (a field period is 90˚) for a 4-field-period (N = 4) PoP-sized (R_0 = 1.5 m, B = 1.4 T) QO configuration with $R_0/\langle a\rangle = 3.6$ for both the vacuum configuration and $\beta = 7\%$. There is a magnetic well across 80-90% of the plasma cross section for $\beta > 1\%$, with shear in $\iota(r)$ at the edge stabilizing Mercier and ballooning modes. The computed β limit for Mercier modes, high-n ballooning modes, and external kink instabilities is >7%. Figure 3-2 shows the rotational transform profile $\iota(r)$ for the same conditions. The current profile used is consistent with the bootstrap current profile, and the magnitude is approximately self-consistent with the finite-β equilibrium; further calculations are underway to couple the equilibrium and bootstrap current. Because of the canceling effects due to the bootstrap current and the curvature-driven Pfirsch-Schlüter current. for this particular QO configuration, the ι profile at $\beta = 7\%$ is nearly equal to its vacuum value at all radii. This implies an invariance of the ι profile over a wide range of beta. Other QO configurations are currently being investigated that have significantly lower bootstrap current. Because $\iota(0)$ is typically >0.1 for QO configurations, it is possible to find QO configurations that require no external net driven current, even on axis, to obtain reasonable ι profiles.

Figure 3-3 demonstrates one of the unique features of QO configurations: use of the flexibility in the |B| Fourier spectrum to suppress the bootstrap current. A series of optimized N = 4 configurations with increasing levels of either the helical or bumpy |B| components lead to a bootstrap suppression up to a factor of 20 below that of an equivalent axisymmetric device. These calculations are not yet fully consistent with the initial equilibrium, but preliminary results indicate that the self-consistent calculation may lead to even larger suppression factors while maintaining QO transport

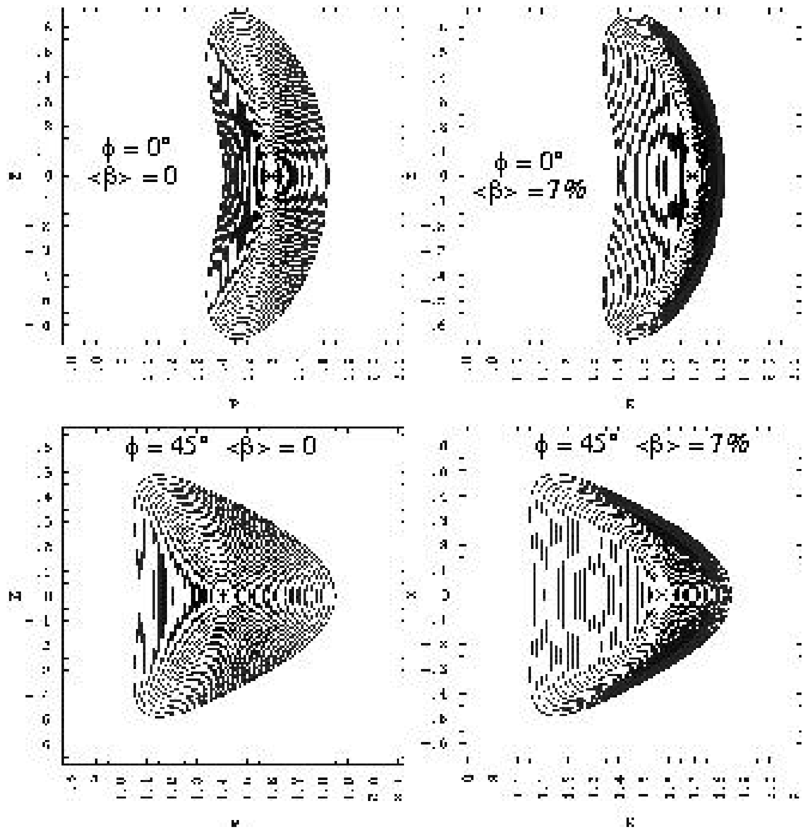

Fig. 3-1. Flux surfaces for a four-field-period QO configuration at toroidal angles of 0° and 45° for the vacuum configuration (left) and at a volume-average beta β of 7% (right).

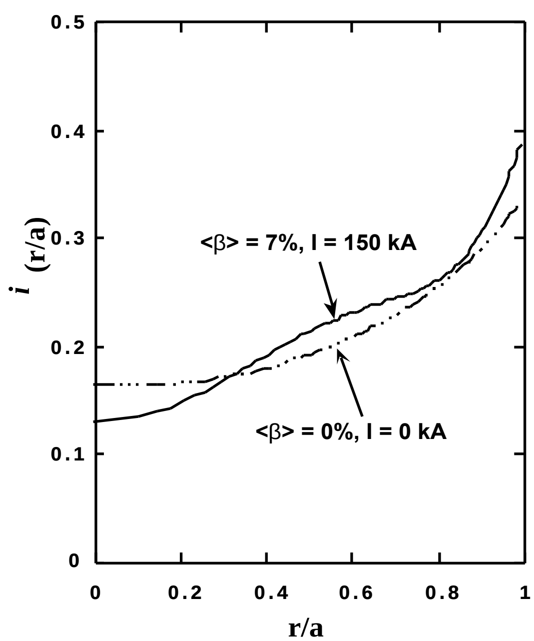

Fig. 3-2. The rotational transform profile $\iota(r)$ for a four-field-period QO configuration for the vacuum configuration and for β = 7%. (R_0 = 1.5-m PoP case)

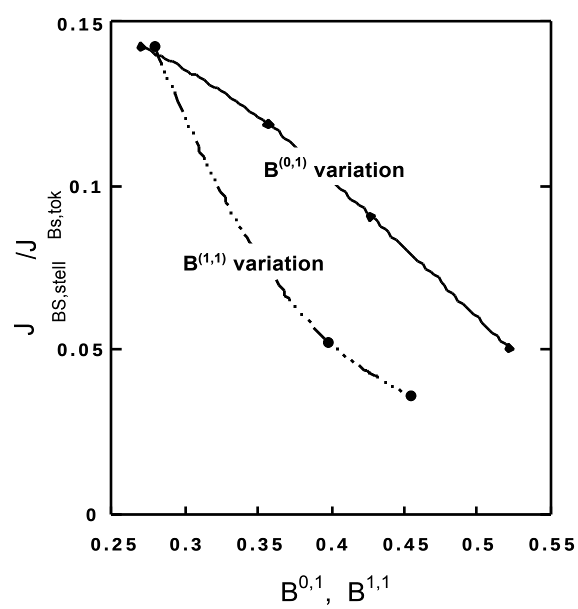

Fig. 3-3. Suppression of bootstrap current (relative to an equivalent tokamak) in 4-field-period QO devices in which the helical (1,1) or the bumpy (0,1) field component of *B* has been increased.

levels. Although the study in Fig. 3-3 was done with separate configurations, it is possible that some degree of variation of this type could be incorporated into a single coil set. In addition to devices with ɫ(r) ranging from 0.1 to 0.4, we have also recently found that QO configurations are possible with ɫ(r) ranging from 0.6 to 1. However, stability studies have not been performed for these configurations yet.

Examination of fast ion confinement in an N = 3 QO device with R_0 = 1.4 m and B_0 = 1.35 T indicates reasonable central (*r*/*a* < 0.5) confinement up to ~100 keV. Above this energy, ions begin to be lost due to loss of magnetic moment conservation at large Larmor radius.

Figure 3-4 shows the outermost flux surface and an unoptimized set of modular coils that produces an equilibrium nearly the same as that shown in Fig. 3-1. In this optimization, the last closed flux surface was computed for its desired physics properties, the coils were found that created it, and the configuration that was actually produced by those coils was calculated.

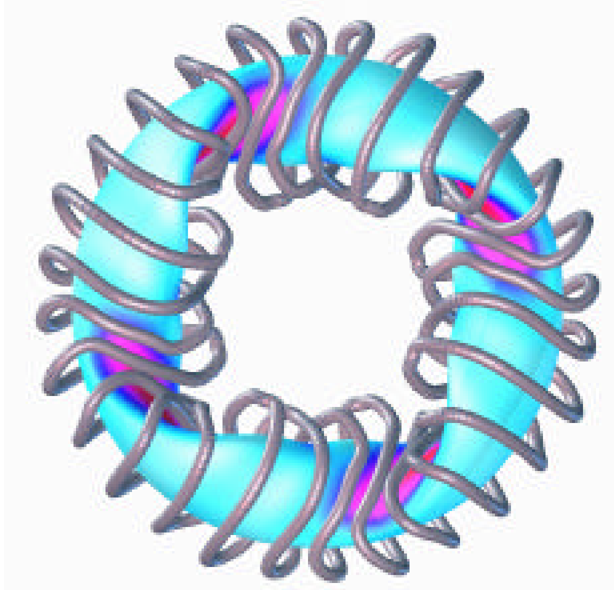

Fig. 3-4. The last closed vacuum flux surface and the coils that produce it for the four-field-period QO configuration in Figs. 3-1 and 2. Contours of constant |B| are indicated by the same color.

3.2 Physics Issues for a QO Concept Exploration Experiment.

A QO concept exploration level experiment is needed at this time to: (1) provide a data base on low-aspect-ratio QO-specific issues that can feed into the optimum design of a PoP-level QO test, and (2) broaden the scientific base provided by the QA PoP and HSX experiments into low-aspect-ratio non-symmetric stellarator configurations. Key issues in exploring the QO optimization approach at low aspect ratio are

(1) reduction of neoclassical transport via nonsymmetric quasi-omnigeneity, and the effect of radial electric fields on confinement;

(2) reduction of energetic orbit losses in non-symmetric configurations;

(3) reduction of the bootstrap current (cancellation by different B harmonics), its compatibility with confinement improvement and the QO optimization, and the independence of ɫ on β;

(4) production of sheared $E \times B$ flow and understanding flow damping in non-symmetric configurations, important for affecting anomalous transport;

(5) tests of the predicted dependence of the size of equilibrium islands on shear, the bootstrap current direction, and beta; and

(6) tests of methods for particle and energy-exhaust control compatible with QO stellarators.

These research areas complement studies done on the quasi-symmetric QA and HSX configurations and broaden our understanding of compact stellarators:

3.3 The QOS Experiment

The degree to which these issues can be addressed depends on the scope of the QO device. Table 3-1 gives device parameters for two concept exploration experiments: a QO stellarator and the existing HSX device. The major radius and magnetic field for QOS were chosen to limit the total project cost to $6.5 million, approximately the same as that for HSX. The average plasma radius is the same as that of the earlier U.S. Advanced Toroidal Facility (ATF), and larger than the 20-cm radius of W7-AS and CHS, but the major radius and magnetic field are half that of ATF.

Table 3-1. Comparison of Device Parameters for QOS and HSX

Device Parameter	QOS	HSX
Average Major Radius, R_0 (m)	1.0	1.2
Average Plasma Radius, $<a>$ (cm)	28	15
Plasma Aspect Ratio $R_0/<a>$	3.6	8
Plasma Volume (m^3)	1.55	0.53
Magnetic Field on Axis, B_0 (T)	1	1.35
Maximum Plasma Current, I_p (kA)	<150	---
Pulse Length (s)	0.2 - 1	0.2
Electron Heating	0.4-0.6 MW; 53/60 GHz	0.2 MW; 28 GHz
Ion Heating	>1 MW; ICRF	

Flexibility for varying the magnetic configuration properties will be provided by three sets of poloidal field coils, as on ATF, to vary the plasma current and the dipole (axis shift) and quadrupole (plasma oblateness) components of the poloidal field, and by small trim TF coils, as on HSX.

Plasma Performance. The practical difference between a concept exploration experiment and a PoP experiment is the plasma volume and amount of plasma heating available, which determines the range of beta and ion collisionality that is accessible, and the extent of the diagnostics. Table 3-2 gives the projected plasma parameters for two values of plasma density and heating power for a QOS concept exploration experiment, based on the ISS95 stellarator scaling with a confinement improvement factor H = 2, similar to that routinely obtained on the low-shear W7-AS stellarator. Up to an additional factor of 1.5 confinement improvement has been obtained in W7-AS. Achieving that level of confinement improvement, a target for QOS, would increase the plasma parameters in Table 3-2 by the same factor. The flat density profile and parabolic-squared temperature profiles characteristic of stellarators are assumed. The maximum density n_{max} is calculated from the value given by Sudo [3-2] with a multiplier of 1.2, as found in ATF; with ECH n_{max} is determined by the heating mode (2nd-harmonic X-mode or X-O EBW mode conversion [3-3]).

Table 3-2. Consistent Sets of Plasma Parameters for the QOS Experiment

Plasma Parameter	0.4-MW ECH		1-MW ICRF	
Line-Average Density, n_e ($10^{19}m^{-3}$)	1.6 (2^{nd} X)	3.2 (O-X EBW)	3.2	10
Energy Confinement Time, τ_E (ms)	11	16	9.4	17
Volume-average beta, β (%)	0.5	0.7	1.0	1.8
Central Electron Temperature, T_{e0} (keV)	2.3	1.6	1.6	0.7
Central Ion Temperature, T_{i0} (keV)	$<<T_{e0}$	$<<T_{e0}$	0.8	0.7

Research Program, Cost and Schedule. Initial operation is limited to electron heating with 53- or 56-GHz ECH and a base set of standard diagnostics. Plasma heating upgrades (ICRF antenna) and a more sophisticated set of diagnostics will be funded out of the operating budget. Operation with 0.4-MW ECH power allows significant investigation of the QOS research areas (1), (3), (4), and (5) at temperatures up to 2.3 keV and β up to 0.7%, but mainly for electrons. The later increase in heating power, which allows ion heating and higher beta, and upgrades to the diagnostic capability, will extend these investigations to more relevant parameters (and to energetic ions) and allows study of areas (2) and (6). Steerable ECH antennas allow studying on-axis and off-axis electron heating and an ICRF antenna allows studying fundamental, second harmonic, minority species, and IBW heating schemes. Although the β values expected are not enough to test predicted MHD limits, these beta values are sufficient to study bootstrap-current related issues.

Confinement improvement is a key research area for the QOS experiment. Because of the $P^{-0.59}$ dependence in τ_E^{ISS95}, a factor of 1.5 confinement improvement is equivalent to increasing P by a factor of 2.7 for β and $T_{e,i}$, and a factor of 5.4 in P for a factor of 2 confinement improvement. Low neoclassical losses allow tests of anomalous confinement improvement; spoiling neoclassical confinement allows testing the degree of neoclassical improvement. Neoclassical confinement for an N = 3 configuration, for B_0 = 1T and no ambipolar radial electric field, is ≈2-3 times better than the ISS95 stellarator scaling. Inclusion of the ambipolar electric field further improves neoclassical confinement by more than a factor of 3. The electric field can be modified through ICRF and biased probes to affect neoclassical confinement.

The QOS experiment requires an annual budget ~10% of that for the PoP facility, or ≈$2.5 million/year. The total project cost (TPC) includes all design, R&D, construction, site preparation, installation of 200-kW ECH, and a base set of diagnostics. About 3-4 years would be required for construction and commissioning the experiment after a final optimization and design phase. There are significant resources at ORNL, UTX, and PPPL that can be applied to the experiment to reduce costs: heating systems, power supplies, control systems, the standard set of tokamak diagnostics, as well as specialized diagnostics (heavy ion beam probe, phase contrast imaging, etc.). Extensive collaborations, nationally and internationally, and use of thesis students will allow extending the scope of the program.

Table 3-3. Cost Profile for QOS Project (M$)

Fiscal Year	**1999**	**2000**	**2001**	**2002**	**2003**	**2004**	**2005**
Construction TPC	0.5	1.1	1.8	2.1	1.0		
QOS Operations	0.1	0.1	0.1	0.1	1.0	1.3	1.5
QOS Enhancements	0.0	0.1	0.1	0.1	0.4	1.2	1.0
QOS Total	0.6	1.3	2.0	2.3	2.4	2.5	2.5

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4. ONGOING CONCEPT EXPLORATION EXPERIMENTS

A. Helically Symmetric Experiment (HSX)

The Helically Symmetric Experiment (HSX) [1,2] is the principal element in the U.S. stellarator program at the present time. Construction is nearly complete with operations to commence in FY 1998. The goal of the HSX experimental program is to test the improved confinement properties in quasi-symmetric configurations and to exploit its unique geometry to elucidate outstanding issues in toroidal confinement. The symmetry in the |B| assures neoclassical transport analogous to the tokamak and reduced by nearly two orders of magnitude from the conventional stellarator in the collisionless regime. The primary objectives of the physics program are:

- Verify reduction of neoclassical transport for quasi-symmetric configurations; quantify levels of symmetry necessary to achieve full benefits
- Demonstrate a reduction in the direct loss of deeply trapped particles
- Show that restoration of a direction of symmetry leads to lower viscous damping of plasma rotation on a flux surface

HSX is a quasi-helically symmetric [3] (QHS) device, and the only device of this type in the world program. The physical parameters of HSX are shown in Table 4.1. The symmetry is obtained by reducing the toroidal curvature term in the magnetic field spectrum through appropriate shaping of the plasma. HSX has the toroidal curvature of an aspect ratio 400 conventional device, while being a fully toroidal system of aspect ratio 8. The spectrum thus possesses a single dominant helical harmonic, with symmetry breaking terms well under 1%. QHS configurations have also been identified [4] numerically at aspect ratio 6 with only minimal increases in symmetry-breaking terms from those in HSX.

QHS configurations have an effective transform given by the number of field periods minus the actual transform, $|N - \iota\!\!\!-|$; for HSX with four field periods and near unity transform, $\iota\!\!\!-_{eff} \sim 3$. Thus, HSX will have neoclassical transport analogous to a $q = 1/3$ tokamak. The high effective transform has multiple benefits, which factor into the elements of the experimental program:

- Reduction of Pfirsch-Schlüter and bootstrap currents; small finite-beta effects on the magnetic field spectra and equilibrium
- Smaller banana widths with accompanying improved confinement of high-energy particles; HSX can fit as many banana widths within its 15 cm minor radius as a stellarator or tokamak with a much larger plasma cross-section.
- Anomalous transport should be reduced, based on data from L-2 and ISS95 scaling, which scale inversely with transform.

HSX is an extremely flexible device. The QHS field is produced by a set of 48 modular coils. A set of 48 planar, non-circular, auxiliary coils provides for variation in rotational transform, magnetic well depth, and spectral content. One configuration of the auxiliary coils (mirror-mode) breaks the quasi-symmetry and increases the transport back to the level of a conventional stellarator, with minimal effect on the plasma stability. In an alternate mode (well-mode), the plasma stability limit to Mercier and ballooning modes [5] can be varied by a factor of 3 (Mercier: 0.4% to 1.3% , ballooning: 0.7% to 1.7%), with only small changes in the neoclassical transport. In the mirror mode, direct losses are dramatically increased and the neoclassical electron thermal conductivity increases by 2-3 orders of magnitude. Finally, the parallel viscosity can be altered by 1-2 orders of magnitude to examine how changes in the plasma rotation and radial electric field affect confinement.

H-mode confinement characteristics are not necessarily predicted in the HSX physics program. The experimental program will, however, provide insight as to the mechanisms by which the $E \times B$ shear necessary for the quenching of anomalous transport can be maintained. Through use of the auxiliary coils, HSX can span the space between having a large plasma flow contribute to the radial electric field (in QHS mode) or having the non-intrinsically ambipolar electric field arise naturally when the symmetry is broken. Additionally, HSX will investigate the relationship between anomalous transport, effective transform, and the level of symmetry. Initial flow measurements will be made using passive spectroscopy utilizing the C^{+4} and other impurity lines. We are implementing a CHERS system over the next three years for more detailed flow measurements. Density profiles will be measured in HSX using a 9-chord interferometer being set up in collaboration with David Brower of UCLA. We are also working with Brower and Tony Peebles to set up a reflectometer for density fluctuations as we move more into the study of anomalous transport. In addition to our 8-point Thomson scattering system, we have an ongoing collaboration with Neville Luhmann's group at UC-Davis to implement a 2-D ECE imaging system on HSX for electron temperature profiles; this diagnostic will also provide electron temperature fluctuations for the anomalous transport studies.

HSX will use a 28-GHz gyrotron with a power output of 200 kW to heat the plasma electrons into the collisionless regime for this first part of the HSX program. This has been shown to be effective in other stellarators of similar size and field strength, such as L-2, and is ideal for carrying out the above investigations. The 28-GHz heating does restrict HSX to operation at 1 T or 0.5 T and to densities less than 10^{13} cm^{-3}. Heating in the ion cyclotron range of frequencies in stellarators has typically suffered from impurity problems. Poor confinement of the high-energy ions in the rippled magnetic field is suspected as the cause of the unsuccessful results. Experiments performed on CHS with ICRF heating [6] of ECH target plasmas has shown strong electron heating. The loss of high-energy trapped ions produced in an ion heating regime degraded the performance and limited the duration of the discharge. The high effective transform and good trapped-particle confinement in HSX should permit effective ion or electron heating in the QHS mode with ICRF, although we would need to operate at somewhat higher densities than CHS for ion heating to reduce central charge-exchange energy loss. One megawatt of ICRF on HSX would provide a wider range in the density-field operating space as well as provide direct data on effects of orbit confinement and impurity production with ICRF. Additionally, the ICRF in combination with the ECRH and auxiliary coils could allow tailoring of the radial electric field through differential loss mechanisms over the plasma cross section. The addition of the ICRF is envisioned as an extension of the main program after the primary goals above are accomplished.

In order to obtain maximum scientific benefit from the HSX experiment, tight coupling with the theory and computation effort (Chapter 5) is needed in the following areas: effects of small non-symmetric fields (how much symmetry is enough?), ballooning mode stability limits (can they be made low enough to test?), Fokker-Planck and delta-f modeling of ECH in HSX, and turbulence and anomalous transport predictions for HSX-type plasmas.

HSX is presently in the second year of a three-year grant period, with a funding level of $1.6 million. The initial HSX program can be achieved within this budget level. Base funding requests for years beyond the present period will include modest increases to cover escalation in costs. The funding profile is shown below for the next 8 years. Acquisition of RF systems ($0.5 million total) is included in the profile for years 2001, 2002 and 2003 and is based upon the purchase of all required components. The costs presented represent no major changes in the goals of the HSX program; adjustments would be made as dictated by changes in program goals and available funding.

Projected Budget Requests for Next 8 Years (in $M)								
FY	1998	1999	2000	2001	2002	2003	2004	2005
Request	1.6	1.6	1.7	1.9	2.0	2.0	2.0	2.0

The HSX Device		Estimated Parameters	
Major radius:	1.2 m	Heating power (28 GHz ECRH)	200 kW
Average plasma minor radius:	0.15 m	Pulse length	100 ms
Field Periods	4	Electron density	$< 10^{13}$
Rotational transform: axis	1.05	Central electron temperature	
Edge	1.12	(with 100 kW absorbed)	~ 1 keV
Magnetic well depth	0.6%	Energy confinement time (LHD)	2 ms
Magnetic field strength	1.37 T	Plasma electron β	0.3%
Magnet flat-top (full field)	0.2 s	ν^*_e	<0.1

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B. Compact Auburn Torsatron (CAT)

A major thrust of the proposed proof-of-principle program is the development of high performance, compact stellarators that operate with a significant fraction of the total rotational transform generated by the bootstrap current such that they represent hybrids between tokamaks and stellarators. The immunity from current-driven disruptions, and more generally, from tearing and kink modes, provided by the 3-D hybrid configuration is important to the success of these concepts. To provide timely experimental input on current-driven stellarator instabilities at the earliest possible opportunity, disruption and stability studies will be carried out in current-carry-ing plasmas in the existing Compact Auburn Torsatron (CAT), suitably modified for these investigations. These studies will support the PoP program by developing an understanding of current-driven instabilities in a flexible stellarator characterized by a combination of vacuum rotational transform, rotational transform from ohmic current, and pre-existing magnetic islands at the rational surfaces (which could lead to locked mode-type disruptions). An additional modification will allow ICRF heating and plasma generation issues to be investigated.

Experimentally, major disruptions have been shown to be suppressed in current-carrying stellarators with a vacuum transform as low as $\iota_o = 0.14$. [1,2] In the case of W-7A, the total transform was about 0.65. These results are generally consistent with calculations of tearing mode stability in stellarators [3-5] which also indicate the importance of the shear in providing stability. The importance of understanding and controlling disruptions in low aspect ratio stellarators motivates a detailed experimental investigation of the physics of stellarator disruptions to support the proof-of-principle effort. Because of the flexibility offered by two independently controlled helical coil sets, the vacuum rotational transform in CAT can be varied from $\iota_o = 0.08$ to 0.6, allowing a large range of rotational transform, making it an ideal test bed for these studies. The moderate shear of the vacuum transform is stellarator-like ($d\iota/dr > 0$), but can be reversed with sufficient plasma current. The CAT upgrade will perform three essential functions:

1. Addition of an ohmic transformer and capacitor bank power supply to drive a plasma current Ip ≤ 25 kA for stability studies. Multiple vertical field coil sets are already available for additional equilibrium control.
2. Increase of the magnetic field from 0.1 T (typ.) to 0.5 T with the use of motor generator sets transferred from MIT and the University of Wisconsin to obtain higher plasma densities and temperatures ($n_e \geq 1 \times 10^{19}$ m^{-3}; $T_e \geq 200$ eV).
3. Implementation of ICRF plasma generation and heating scheme (developed on CHS [6]) at a power level $P_{rf} \geq 100$ kW, frequency f ≥ 6 MHz with a Nagoya Type-III antenna to provide target plasmas for current stability studies. The use of non-resonant ICRF plasma generation allows operation over a range of magnetic field.

With the upgrade, ohmic rotational transforms $\iota_o \leq \pm 0.5$ can be obtained, allowing full exploration of external kink stability as well as internal resistive modes. The measurement of rotational transform profiles in the upgraded CAT device will allow determination of disruption threshold conditions in terms of external transform fraction and shear dependencies. The angle of the local magnetic field will be measured by polarization of Zeeman–split lines of injected neutrals (e.g. Li or He) using the present CW laser-induced fluorescence system on CAT. The results will be interpreted with the VMEC equilibrium code to obtain the rotational transform profile.

Following the stability studies, the ICRF capabilities of the upgraded CAT facility will be used to explore plasma generation, direct ion heating [7], and mode conversion electron heating [8] to explore methods that can be used to achieve finite β plasmas in other PoP program devices.

The extended CAT program proposed here requires a base operating budget of about $320k/year. In addition, about $200k is required in the first year of the program and $40k in the second year for the upgrades needed for disruption studies. An additional $150k is required for upgrades needed to carry out the ICRF studies.

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5. THEORY

Stellarator theory has three fundamental objectives in the context of the proof-of-principle program: (1) the creation of a framework for the interpretation of experiments, (2) the development of techniques for the extrapolation of the results from the proof-of-principle to the proof-of-performance level of experiments, and (3) the further optimization of stellarator configurations. A secondary objective is the general development of three-dimensional plasma theory. Theory has had a strong role in establishing the physics basis for compact stellarators, more so than for other fusion concepts. The future development of the compact-stellarator knowledge base, and the world-wide development of the stellarator, will continue to require a strong theory effort, addressing eight areas: (1) MHD equilibrium, (2) magnetic island formation, (3) MHD stability, (4) neoclassical transport and drift orbits, (5) microstability and anomalous transport, (6) divertor and edge physics, (7) waves and heating, and (8) optimization of magnetic configurations. Theory progress in all these areas will be important for the new U.S. experiments testing compact-stellarator concepts as well as existing experiments such as HSX and the large foreign experiments.

A. MHD Equilibrium

Several of the pioneering stellarator equilibrium codes were developed in the United States: BETA [1], VMEC [2], and PIES [3]. The VMEC code is used worldwide as the fundamental design and interpretation tool for stellarators. The PIES code, unlike BETA and VMEC, does not assume the existence of good flux surfaces, and is therefore capable of handling magnetic islands and stochastic field lines. The VMEC and PIES codes are the two major stellarator codes developed in the U.S. that are widely used internationally and whose continued development and support are the subject of active international collaborations. Since an expanded U.S. program will increase the use of these codes for concept development, experiment design, and data analysis, improvements in code performance and speed are needed. Ideas for further algorithm improvement in VMEC and PIES, to improve the convergence behavior with increasing resolution of VMEC, and to improve the speed of PIES [4], have been proposed and will be developed.

B. Magnetic Island Formation

Equilibrium beta limits in stellarators can be set by the formation of magnetic islands and magnetic surface breakup. This issue does not arise in axisymmetric equilibria, but the physics is closely related to that of tearing modes in axisymmetric systems. Important physical effects, well known in tokamaks, such as neoclassical tearing modes [5] and rotational shielding of resonant perturbations [6] can change, and in many cases improve, the quality of stellarator magnetic surfaces. Stellarator configurations with significant bootstrap current are of interest for the proposed U.S. stellarator program. The PIES code will be modified to include perturbed bootstrap current effects, which gives rise to the neoclassical tearing effects. The effects of rotational shielding will also be assessed. An algorithm that uses the output of the VMEC and BETA codes to estimate the quality of the surfaces would be of great value in scoping and optimization studies and will be developed.

C. Stability

After equilibrium the most fundamental issue in the design or interpretation of experiments is MHD stability. Low mode number ideal instabilities are presently studied using the European codes Terpsichore [7] and CAS3D [8]. Systematic calculations are required to determine the role of net toroidal current in configurations where the rotational transform is produced by both the plasma current (as it is in a tokamak) and external coils. Another important issue in stability is whether localized stability criteria such as Mercier and ballooning give predictions that are pessimistic. The predictions of localized stability analysis need to be compared with the results of stability codes that include kinetic effects, as well as with global MHD stability codes and with experiments.

D. Neoclassical Transport and Drift Orbits

The constraint of adequate confinement of the particle drift orbits is not trivially satisfied in a stellarator, in contrast to the situation in an axisymmetric tokamak. Two fundamentally different concepts for achieving good orbits in stellarators are known: quasi-symmetry [9] and nonsymmetric, or quasi-, omnigeneity [10]. A number of neoclassical transport issues remain to be addressed. An example is the development of transport theory for quasiomnigeneous (QO) systems in which the particle trajectories remain close to a flux surfaces but do not satisfy the symmetry assumptions upon which much of neoclassical theory is based. The quasisymmetric stellarators, both quasi-axial (QA) and quasi-helical (QH), always have some breaking of the symmetry of the drift trajectories [11]. Techniques that were developed to align the action ($J = \int mv_{\|}dl$) contours with the magnetic surfaces in QO stellarators will be used to minimize the effects of symmetry breaking in QA and QH stellarators. Some of these issues can be studied analytically, but the complex geometry of stellarators generally requires numerical calculations. The U.S. DKES code [12] is a variational spectral code which is in use worldwide for computing neoclassical transport coefficients in stellarators. The δf codes for neoclassical transport [13] pioneered in the U.S. should be developed into production codes, which would provide useful tools for the study of neoclassical effects and heating in the experiments. The output of these delta-f codes could be statistically refined by coupling to the DKES code.

E. Microstability and Anomalous Transport

The empirical scaling of transport in stellarators is similar to that in tokamaks. However, no stellarator has been designed to minimize anomalous transport, and the advancement of understanding in this area will be an important goal of the proposed program. The gyrokinetic [14], gyrofluid [15], and linear microstability codes [16] that have been developed for studying the microstability and anomalous transport in tokamaks will be adapted to stellarators. Not only are such codes required to interpret experiments but they can also address whether the broad range of magnetic configurations available in stellarators allows a significant reduction in the predicted transport rates. An important application will be the design of experiments that test critical physics issues of transport minimization. For example, microturbulence such as the ion-temperature-gradient (ITG) mode may break the ion longitudinal action invariant, and ease a design constraint on stellarators. Methods to utilize flow-shear stabilization of turbulence and anomalous transport will be studied.

F. Divertor and Edge Physics

A strategy for dealing with particle and energy exhaust at the edge of the plasma is fundamental to any fusion concept. Concepts for stellarator divertors exist and some computational studies have been made. Codes that model tokamak divertors do not address important issues for stellarator divertors. Thus, divertor theory that is appropriate for stellarator applications [17] requires significant development.

G. Wave Propagation and Heating

Waves and heating techniques in stellarators, as in tokamaks, can play crucial roles: pressure and current profile control, flow shear drive, control of electric field, heating and velocity space control of selected particle populations. Aside from ray tracing, none of the computational tools for RF propagation and absorption has been adapted for 3D geometry. Also the RF Fokker-Planck codes have not included the multiple classes of trapped particles found in stellarators. Development of these tools will be undertaken.

H. Optimization of Magnetic Configurations and Coils

All of the above studies, coupled with experimental results from the proof-of-principle and concept exploration devices, will allow development of new, better-optimized, stellarator concepts. The optimization efforts will have two aspects: the development of optimization criteria and the exploration of configurations satisfying these criteria. Improved optimizers will include more comprehensive and more accurate physics criteria. They will apply increasingly sophisticated search algorithms running on increasingly powerful (massively parallel) computers.

Both the plasma configuration and the coil design are important areas for optimization. The standard method for studying coils is the European NESCOIL code [18]. In guiding optimization efforts, a number of subsidiary studies are required. An example is the tradeoff between beta and aspect ratio in a stellarator. Theoretically one can achieve volume-averaged betas in stellarators above 30% at infinite aspect ratio [19], but how does this beta limit change as the aspect ratio is made smaller? Also, new methods have been developed for enhancing the ballooning stability of the QA and QO configurations, and these methods need to be studied to understand how they work and what their range of applicability is. In particular it is important to apply such techniques to quasi-helical symmetry, which is the basis of the HSX experiment. Quasi-helical symmetry has the desirable feature of very low parallel current but existing designs have low beta limits and larger aspect ratios than the most compact QA or QO stellarators.

To carry out the required theoretical research effort in all eight areas requires approximately 16 theorists in total, plus graduate students and postdoctoral fellows. Much of the needed increase can be achieved through changes in the research focus of existing theorists, which will occur naturally as the stellarator experimental program is strengthened. This resource estimate, equivalent to approximately $3.5 million/year, would allow the United States to address the theoretical issues that are critical to an innovative development of the stellarator concept.

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6. INTERNATIONAL COLLABORATION

Collaboration with the larger international stellarator program on selected topics is an important element of the U.S. stellarator PoP program because it provides information on stellarator concept improvement that is not otherwise available in the U.S. program. The international stellarator program is already at the *proof-of-performance* stage. It features billion-dollar-class facilities now operating in Japan (LHD) and under construction in Germany (W7-X, 2005) that are designed to demonstrate steady-state disruption-free stellarator operation and a level of performance that allows extrapolation to devices capable of burning plasma operation. These large facilities are supplemented by *proof-of-principle* ($30-100 million scale) experiments in Japan (CHS), Germany (W7-AS), and Spain (TJ-II).

A. Experimental Collaborations.

The wide range of stellarator configurations accessible on LHD, W7-AS, CHS, and TJ-II allows study of high aspect ratio configurations, degree of helical axis excursion, magnetic-island-based divertors, and the consequences of a modest driven plasma current, elements that are incorporated in the low-aspect-ratio QA and QO stellarator concepts. Areas of particular importance are ion heating and transport, neoclassical transport, role of electric fields in confinement improvement, enhanced confinement modes, beta limits, practical particle and power handling, profile and configuration optimizations, and steady-state performance. Study of these issues at higher aspect ratio (A = 5-11) and low bootstrap currents in foreign experiments complements the U.S. stellarator PoP program, which focuses on lower aspect ratio (A = 3-4) and larger bootstrap current.

LHD (Japan). The order of magnitude increases in plasma volume, heating power, and pulse length of LHD over that in existing stellarators allows studies of size scaling and stellarator physics at more reactor-relevant parameters. Collaborations on LHD, which just began operating, should be expanded soon to take advantage of the new opportunity.

(1) Ion heating, transport, and orbit confinement will be studied through analysis of the energetic ion distribution (a) toroidally and vertically at small to medium pitch angles and (b) toroidally at larger pitch angles. Study of fast ion behavior is necessary for understanding the effectiveness of neutral beam and ICRF heating, energetic particle orbit losses, and their reduction with magnetic configuration properties and ambipolar electric fields. Reduction of energetic orbit losses and the associated improvement in neoclassical confinement is important for optimization of QO stellarators with large magnetic field ripple. Understanding the ion temperature behavior, its scaling, and the ion confinement improvement associated with internal transport barriers is important to both QA and QO stellarators.

(2) Direct measurement of energetic particles on unconfined orbits will be done with an array of lost-particle detectors. This provides additional information on energetic orbit losses and their amelioration by tailoring the magnetic field spectrum and by ambipolar electric fields.

(3) An ECE array has been constructed for fast measurements of the electron temperature profile, which is important for studies of electron heating and transport.

(4) High-β configurations will be analyzed using magnetic loops, bootstrap current codes, and 3-D MHD equilibrium and stability codes. Understanding beta limits is central to optimization

of both QA and QO stellarators. Present stellarator experiments have not obtained $\beta > 2.1\%$, but $\beta \geq 5\%$ is predicted for LHD at a magnetic field of 1.5 T.

(5) An imaging bolometer array will be used to measure the amount and spatial distribution of impurity radiation, which is important in understanding the overall power flow in LHD.

W7-AS (Germany). Confinement improvement and divertors are being studied in W7-AS in magnetic configurations complementary to that of LHD. The U.S. is participating through

(1) pellet injection for central fueling and reduction of edge recycling to explore maximum performance (higher β, longer τ_E, higher density), profile shaping for confinement improvement, and central fueling with the island-based divertor system that will be installed in 1999.

(2), analysis of the consequences of a net plasma current (a key element of the QA and QO concepts) and of a magnetic-island-based divertor applicable to modular-coil stellarators.

CHS (Japan). CHS allows study of transport and beta limits at plasma aspect ratios as low as 5. The U.S. has contributed in a number of areas: (1) HIBP measurements of the electric field, critical for understanding neoclassical transport in stellarators; (2) minority-species and second-harmonic ICRF heating; (3) measurement of direct fast-ion orbit losses; and (4) assessment of the effectiveness of the local island divertor. These areas should be developed further on LHD.

TJ-II (Spain). The U.S. can also take advantage of the U.S.-supplied neutral beam heating system on the TJ-II flexible heliac to understand beta limits and transport in a stellarator with a large helical axis excursion, an important ingredient in U.S. stellarator configuration optimization.

H-1 (Australia). The H-1 flexible heliac allows studies of a stellarator with a large helical axis excursion in a configuration complementary to that of TJ-II.

A. Theory Collaborations.

Collaboration on stellarator theory and computational tools development can benefit U.S efforts in support of compact stellarator concept development. Effective areas for international collaboration are MHD equilibrium; Mercier, ballooning and kink stability; microstability; bootstrap current; transport; optimization techniques; coil design; and effects of magnetic islands.

IPP-Greifswald (Germany). This stellarator theory group has provided the U.S. with access to many computational tools, including stability, coil design, and optimization codes. Close collaboration on magnetic island development through the PIES code has also been carried on for many years. The development of the QA and QO compact-stellarator concepts has relied heavily on this collaboration. The U.S. program will benefit from continued interaction with this group.

CRPP-Lausanne (Switzerland). This theory group has state of the art codes for MHD stability. In particular their ballooning code has been used extensively in the design of QA devices. Their free boundary kink stability code is very fast and there is a three way collaboration with CRPP-Lausanne and IPP-Greifswald to verify crucial MHD stability calculations.

NIFS-Toki (Japan). The NIFS theory group has provided the bootstrap code used in the design of U.S. compact-stellarator experiments. In the future we will be collaborating on island formation by comparing PIES and the NIFS-Toki code, HINT.

Kyoto University (Japan). Joint work on interpretation of Heliotron E results has been carried out for many years, most recently on the understanding of sawtooth oscillations in

Heliotron E. The combined capabilities of linear stability codes at Kyoto and nonlinear codes at ORNL is useful in studies of the impact of profile modification by resistive interchange modes on the ideal MHD beta limits for stellarators.

CIEMAT-Madrid (Spain). ORNL and CIEMAT have collaborated on the conceptual design and physics studies for the two stellarators build at CIEMAT, the TJ-IU and TJ-II. Present collaborations with the U.S. are (1) development and use of the PIES code for free-boundary calculations of equilibria without the assumption of simply nested surfaces and (2) analysis and modeling of plasma edge fluctuations and turbulence.

Universidad Carlos III, Madrid (Spain). 3-D nonlinear codes have been developed with applications to turbulence. New codes are used to study ballooning stability, low-*n* mode stability using the Lagrangian averaging method, and the full 3-D stability of stellarator configurations.

NPFRF-Canberra (Australia). The theory group at the Australian National University are the leading experts on ballooning modes in stellarators. The focus of U.S. collaboration with this group will be the development of capabilities for computer optimization on ballooning stability.

Kurchatov-Moscow (Russia). This stellarator group is expert in transport and kink stability. The U.S. will continue collaboration on transport optimization using pseudo-symmetry.

The resources needed for the experimental collaboration efforts are $1.5 million per year.

7. SYSTEM STUDIES

Integrated physics and engineering systems studies are needed for assessing the reactor potential of compact stellarators and for setting goals for the QA and QO experiments to achieve in order to extrapolate to an attractive reactor. These capabilities have been developed in previous U.S. stellarator reactor studies and in the ARIES tokamak reactor studies. The most recent stellarator reactor example is the U.S. Stellarator Power Plant Study (SPPS) , a "scoping study" at a smaller scale than the typical ARIES study. A byproduct of the SPPS work was development of the quasi-toroidal Modular Helias-like Heliac (MHH) configuration on which the SPPS was based. The four-field-period MHH configuration has physics properties similar to the W7-X configuration, but allows reducing the reactor size from $R_0 \approx 24$ m (for the W7-X-based HSR) to $R_0 = 14$ m for the SPPS reactor. The possibility of further significant reductions in reactor size is a major motivation of the proposed U.S. stellarator program; the goal is another factor of 2 reduction in major radius.

The SPPS configuration extrapolated to a reactor power plant that was economically competitive with the second-stability ARIES-IV tokamak reactor assuming that stellarators have the same unit costs for components with complex geometry as tokamaks and that tokamaks have the same availability as stellarators. Reducing the plasma aspect ratio should lead to significant cost reductions through reducing the mass of the most expensive parts of the fusion reactor core (the first wall, blanket, shielding, and other components that scale with the plasma surface area).

The minimum size of a stellarator reactor. is set by the need for adequate space Δ between the edge of the plasma and the center of the coils for the plasma scrapeoff/divertor region, the first wall, the thick (≈2 m) blanket and shield assembly, the superconducting windings and assembly gaps, etc. For a typical stellarator coil configuration, Δ/R_0 is relatively small; the coils normally have to be close to the plasma because the higher order multipole components that produce the desired magnetic configuration decay away rapidly from the coils. More compact stellarators with larger Δ/R_0 would have a significant impact on the viability of stellarators as reactors.

Studies are needed to assess the potential advantages and design issues for QA and QO configurations as fusion power plants relative to conventional stellarators and tokamaks. Initial scoping studies are needed to examine the differences between QA and QO configurations as fusion power plants; in particular the design consequences of the higher degree of spatial non-axisymmetry for QO configurations and the startup and control consequences of the larger bootstrap current for QA configurations. An in-depth study (similar in scope to the ARIES studies) would clarify the trade-offs on more issues than were possible in the SPPS, and would greatly assist optimization of these new, more attractive, stellarator configurations.

The areas that need to be explored in detail for compact stellarator configurations include:

- cost/benefit tradeoffs for aspect ratio, beta limit, and confinement enhancement to guide the physics optimization efforts and the targeting of experiments;
- limits on acceptable orbit losses for α-particles and other energetic ions to help bound how much optimization of orbit losses is required;
- consequences of practical particle (including impurities) and power handling to help integrate the divertor geometry with the coil geometry;
- cost/benefit tradeoffs for plasma-coil spacing, access between coils, maximum field on the coils, degree of nonaxisymmetry, etc. to guide optimization of the coil design; and
- integration of reactor systems optimization with stellarator physics and configuration optimization to guide the development of self-consistent attractive reactor configurations.

A comprehensive compact stellarator reactor study would allow analysis of the benefits of reducing the aspect ratio for stellarators in much the same way that the recent spherical tokamak reactor study is allowing analysis of the benefits of reducing the aspect ratio for tokamaks.

The resources needed for this work average $1 million per year, alternating between 2 years of scoping studies at the $0.3 million per year level and 2 years of the more detailed ARIES-type studies with a specific QA or QO coil configuration at $1.7 million per year. However, this effort would be funded as part of the ARIES program and *not* as part of the proposed program.