Executive Summary

Magnetic plasma confinement in toroidal geometries is a proven approach for advancing the science of fusion energy and is promising as a basis for future fusion power plants. Most of the research to date has concentrated on two types of configurations, tokamaks and stellarators. Tokamaks, which produce the confining poloidal magnetic field by toroidal currents in the plasma, have demonstrated excellent short-pulse plasma performance in "compact" geometries, with aspect ratios (ratio of the plasma major radius to the average minor radius) usually less than 4. Stellarators use three-dimensional magnetic fields generated by coils to produce some or all of the confining poloidal magnetic field. Stellarators have demonstrated levels of performance approaching those of tokamaks, generally at aspect ratios in the range of 6-12.

Since stellarators can produce all components of the magnetic field directly from external coils, they are intrinsically well suited for steady state operation and do not require external current drive systems. The physics properties of toroidal plasmas are determined by the shape of the MHD equilibrium, and the stellarator's three-dimensional shaping offers many more degrees of freedom than are available with axisymmetric configurations. This additional flexibility can be used to tailor the equilibrium to obtain desired physics properties. Historically, it has been a challenge for stellarators to provide acceptable drift-orbit confinement, allowing adequate fast-ion confinement and low neoclassical transport losses. A strategy for 3D drift-orbit confinement is "quasi-symmetry", which is based upon work by A. Boozer [1] showing that drift-orbit topology and neoclassical transport depends only on the variation of $|B|$ within a flux surface. This was used by J. Nuehrenberg [2] and P. Garabedian [3] to develop stellarators that, while three-dimensional in Euclidean space, have a direction (either helical or toroidal) of approximate symmetry of $|B|$ in (Boozer) flux coordinates. Quasi-symmetric configurations have drift-orbits similar to equivalent symmetric configurations, and thus similar neoclassical transport. Also, rotation in the quasi-symmetric direction is undamped, as in a symmetric configuration.

The Compact Stellarator Opportunity

The Compact Stellarator Opportunity is to build upon the advances in understanding of both stellarators and tokamaks and to combine the best features of both. We wish to use the stellarator's externally generated helical field and 3D shaping to provide MHD stability without external current drive or feedback systems, designing plasmas that have the potential to be free of disruptions. From tokamaks, we wish to use the excellent confinement, ability to stabilize and manipulate turbulent transport, and lower aspect ratio (compared to classical stellarators) to reduce development costs and system size. Three-dimensional shaping is used to passively stabilize the modes that limit the β and pulse-length of toroidal plasmas (particularly the external kink and neoclassical tearing modes). Quasi-axisymmetric design provides good orbit confinement, allows undamped flows to stabilize turbulence, and is best suited for higher β limits and lower aspect ratio.

An experimental facility, the National Compact Stellarator Experiment (NCSX), is proposed to support the research needed to develop the compact stellarator through its proof-of-principle (PoP) stage. The NCSX is the principal element in a national proof-of-principle program, first proposed in 1998 [4], to develop the physics of compact stellarators. It will advance U.S. plasma science goals by broadening the understanding of turbulence and transport, macroscopic stability, wave-particle interactions, and plasma boundary physics, using its unique capabilities to vary configuration parameters. It will acquire the physics data needed to evaluate the attractiveness of the compact stellarator as a concept for fusion energy, a 10-year milestone established by the Fusion Energy Sciences Advisory Committee in 1999. The breadth of the questions proposed for study by NCSX, and particularly the study of high-β plasmas requires a proof-of-principle experiment, with sufficient size, magnetic field strength, heating power, power and particle handling capabilities, diagnostics, and flexibility.

NCSX Scientific Mission and Design Goals

The mission of the NCSX research is to investigate the effects of three-dimensional plasma shaping, of internally- and externally-generated sources of rotational transform, and of quasi-axisymmetry on the stability and confinement of toroidal plasmas. It will contribute to the national fusion program's MFE Goal #1 to advance understanding of plasma and enhance predictive capabilities. It will contribute in important ways to answering some of the major outstanding challenges in plasma science, as set forth in the Fusion Energy Sciences Advisory Committee report, "Opportunities in the Fusion Energy Sciences Program." [5]

Macrostability major challenge: What are the fundamental causes and nonlinear consequences of plasma pressure limits in magnetically confined plasma systems, and how can a fusion system's plasma pressure and hence power density be optimized, with minimum off-normal events?

In NCSX:

- What are the beta limits and limiting mechanisms in a low aspect-ratio stellarator? Can pulse-length-limiting instabilities, such as external kinks and neoclassical tearing modes, be stabilized by external transform and 3D shaping?
- How do externally-generated transform and 3D shaping affect disruptions and their occurrence?

Transport and Turbulence major challenge: What are the fundamental causes of heat loss in magnetically confined plasmas, and how can heat losses be controlled, in order to minimize the required size of a fusion power system?

In NCSX:

- Can the collisionless orbit losses typically associated with 3D fields be reduced by designing the magnetic field to be quasi-axisymmetric? Is flow damping reduced? Is the resulting transport and confinement similar to actually axisymmetric systems? How does the transport scale in a compact stellarator?
- Do anomalous transport control and reduction mechanisms that work in tokamaks transfer to quasi-axisymmetric stellarators? Do zonal flows saturate turbulent transport in a quasi-axisymmetric stellarator at levels similar to tokamaks?

Wave-Particle interactions major challenge: What are the fundamental causes and nonlinear consequences of wave interactions with non-thermal particles, which can be used both to minimize any negative consequences of fusion products in magnetically confined plasmas, and ultimately to take advantage of the free energy represented by the fusion product population?

In NCSX:

- How do the Alfvenic-eigenmode spectrum and stability of a quasi-axisymmetric stellarator differ from those of a tokamak or a non-symmetric stellarator?

Boundary Interactions major challenge: What are the fundamental mechanisms of parallel transport along open magnetic field lines, and how can the heat flux along these field lines be dissipated before its strikes material surfaces?

In NCSX:

- How do stellarator field characteristics such as islands and stochasticity affect the boundary plasma and plasma-material interactions? Are 3D methods for controlling particle and power exhaust compatible with good core confinement?

The compact stellarator offers innovative solutions to some of the critical problems that lie on the path to an attractive fusion power plant. A key challenge for magnetic fusion energy research is finding a compact, high-beta toroidal plasma configuration that can be efficiently sustained without disrupting. The compact stellarator complements the advanced-tokamak and currentless-stellarator strategies. It uses helical fields from external coils, combining three-dimensional plasma shaping and quasi-axisymmetric design to produce a moderate aspect ratio configuration. The NCSX will contribute to the understanding and improvement of toroidal confinement by using its unique capabilities to gain a better understanding of the physics:

- Beta limits and limiting mechanisms.

- Neoclassical transport reduction by quasiaxisymmetric design.
- Anomalous transport reduction by flow-shear control, using reduced flow damping by QA design.
- Stabilization of equilibrium islands and neoclassical tearing modes with stellarator magnetic shear.
- Power and particle exhaust with a compact stellarator boundary.
- Three-dimensional shaping effects on nonlinear magnetohydrodynamics.
- Conditions for disruption-free operation at high-beta with bootstrap current and external transform.

The physics benefits of the compact stellarator solution, passive stability and tokamak-like confinement including the ability to manipulate the turbulent transport with flows, could outweigh the additional costs associated with more complex coils. The NCSX will provide the data needed to make a proper assessment and accomplish program milestones.

Relationship to Other Programs

The U.S. stellarator program will complement the world stellarator program, producing results that are important for fusion science and that will not be obtained otherwise. The Japanese LHD and German Wendelstein programs feature large experiments with superconducting magnets that will advance stellarator core transport and stability physics, boundary physics, and long-pulse sustainment. However, the LHD and W7-X devices have plasma aspect ratios of 6 and 10.6, respectively. The drift-optimized W7-X was explicitly optimized to suppress the bootstrap current. Neither will test the effects of quasi-symmetry on the confinement physics, compatibility with the bootstrap current providing a significant fraction of the rotational transform in the design optimization, or reductions in the aspect ratio of stellarators down to values (~4) approaching those of tokamaks. The NCSX will address these issues, making unique contributions to world stellarator research.

A low-aspect-ratio quasi-axisymmetric experiment, CHS-qa, is being designed by a group at Japan's National Institute for Fusion Science (NIFS), though not yet formally proposed. The CHS-qa configuration design effort, so far, has evaluated plasma configurations assuming there is no net plasma parallel current [6]. The NCSX is optimized around a high-β configuration with self-consistent bootstrap current profiles, and targets the issues of β limits and high-β disruption avoidance. Thus the NCSX and CHS-qa would be complementary

The national compact stellarator program also couples to and builds upon the programs on existing or already-approved concept-exploration experiments at the University of Wisconsin

(HSX) and Auburn University (CTH). The HSX will investigate transport reduction by quasi-helical (QH) symmetry, while the CTH will investigate low-beta stability issues in current-carrying stellarators. In addition, it is expected that a stellarator theory and experimental collaboration program will parallel the project's activities.

As part of the proposed U.S. compact stellarator program, a concept-exploration-level experiment, the QOS, is also being proposed by Oak Ridge National Laboratory as a complement to NCSX. The QOS experiment will investigate the confinement physics of quasi-poloidal (QP) symmetry. Since QP symmetry implies a large toroidal viscosity, the QOS experiment will investigate the role of poloidal flows in confinement enhancement, while NCSX with its low toroidal flow damping will focus more on flow-shear stabilization mechanisms. While the QOS will push down to very low aspect ratio (~2.6), where equilibrium quality is a key issue, the NCSX will push up in beta (4%), where the key issues are stability and disruption immunity. The scientific contributions and design characteristics of the QOS will complement those of the NCSX in order to broaden the knowledge base for compact stellarators.

The proposed NCSX program also complements the Advanced Tokamak (AT) and Spherical Torus (ST) programs, which are developing an approach to sustainment based on the use of nearby conducting walls and active feedback control of unstable modes and current profiles. The NCSX investigates an innovative alternative, where instabilities are stabilized by the design of the three-dimensional shape of the plasma and control is maintained using external coils to shape the plasma.

NCSX Physics Design

In order to carry out its research mission, the NCSX plasma and facility has been designed with the following goals:

- Passive stability to the external kink, neoclassical-tearing, ballooning, and vertical instabilities to $\beta \geq 4\%$, without the need for nearby conducting walls or feedback systems.
- Good quasi-axisymmetry to provide closed drift-orbits and low flow-damping, sufficient to allow balanced neutral-beam injection (for control of parallel flows and beam-driven currents).
- $\beta \approx 4\%$ stable equilibria consistent with steady state without the need for external current drive.
- Aspect ratio ~ 4
- Good flux surface quality at high beta, without taking into account island width reductions due to neoclassical effects.

- Experimental flexibility to vary the plasma shape, external transform, pressure, and current to carry out the research mission.

The NCSX is designed around a computed QAS plasma configuration (Figure 1) It has three periods, an aspect ratio $R/\langle a \rangle = 4.4$, and strong axisymmetric shaping, as well as the three-dimensional shaping that is clearly evident in the figure. The shape is optimized for minimum ripple and marginal stability to several modes at $\beta = 4\%$. The rotational transform profile (Figure 2) increases monotonically, except very near the edge, from about 0.4 ($q \approx 2.5$) to 0.65 ($q \approx 1.5$). The bootstrap current provides about one-fourth of the rotational transform at the edge, while the remainder is provided by coils. The plasma as designed has good magnetic surfaces all the way to the edge except for a small removable island chain at the $\iota = 0.6$ surface.

The main coils for NCSX are the modular coils shown in Figure 3. They resemble toroidal field coils with out-of-plane deformations. Three coils are extended radially to allow tangential access for neutral beams and diagnostics. Not shown but also included are toroidal field coils, poloidal field coils, and trim coils for configuration flexibility. The coils are designed to produce good magnetic surfaces and to reconstruct the physics properties of the reference plasma. The additional coils provide experimental flexibility to test the physics, for example the ability to vary the rotational transform, the shear, and the beta limit, while maintaining good quasi-symmetry. The design is robust in that the coils can provide good configuration properties

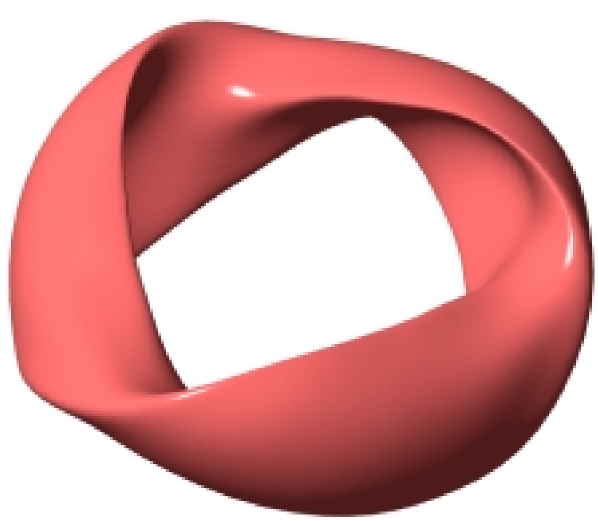

Figure 1. NCSX Reference Plasma Configuration

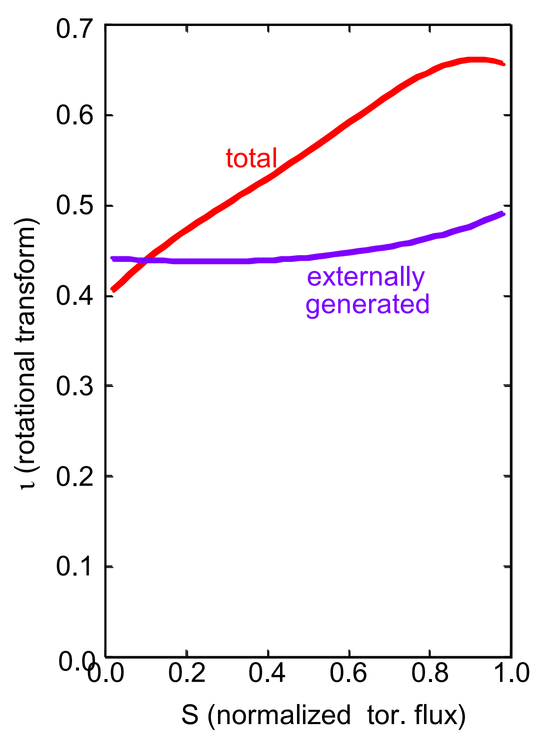

Figure 2. Rotational Transform Profile

over a wide range of β, I_P, and profile shapes.

The NCSX design is robust in its equilibrium evolution during a pulse. Startup simulations advancing from an initial vacuum state to a high-beta target state along a stable path, consistent with planned equipment capabilities, have been carried out. Free-boundary equilibrium calculations show that the coils produce good magnetic surfaces in the vacuum and high-beta states. This is a result of substantial improvements in design concepts and tools since robustness was identified as a technical issue by the Fusion Energy Sciences Advisory Committee in August, 1999. In sharp contrast to the reference plasma configuration presented in 1999, the present configuration has good magnetic surfaces out to the edge. Residual islands have been eliminated in the design by a newly-developed methodology which makes small resonant perturbations in the coil geometry to reduce island widths as calculated by the PIES free-boundary equilibrium code. A system of trim coils is included in the design to provide a capability for reducing islands over the range of equilibria needed for startup and flexibility. As a further measure the configuration is designed with "reversed shear" so that neoclassical effects should reduce the widths of any islands.

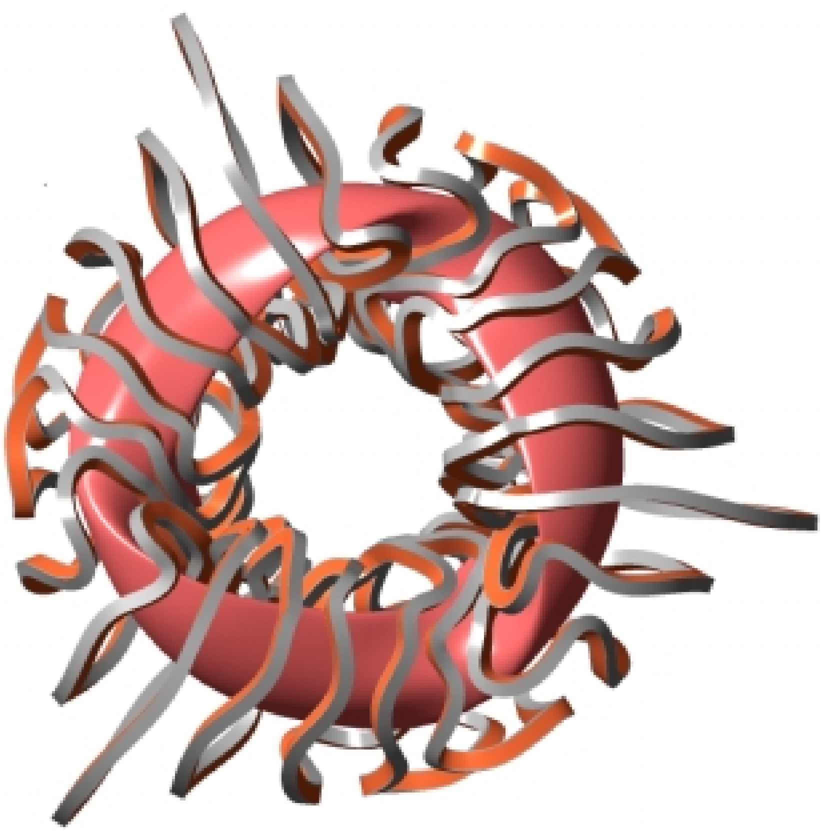

Figure 3. NCSX Modular Coils

The dimensions and performance parameters of NCSX are chosen to meet mission requirements at minimum cost:

- Major radius R=1.4 m.
- Magnetic field 1.2-1.7 T (>2 T at reduced rotational transform).
- Magnet flattop pulse length 1 s at B = 1.2 T, 0.2 s at B = 1.7 T.
- Neutral beam heating power 3 MW initially, upgradable to 6 MW.
- Radiofrequency heating power: upgradable to 6 MW

Plasma-facing component are designed to remove plasma heat losses without overheating material surfaces, control neutral recycling, and minimize impurities. Because of the complexity of the three-dimensional magnetic field, it is expected that the design of the optimum power and particle handling system will be a long-term research program. The NCSX will use carbon as the material for plasma-facing components, which will be bakable in-situ to 350°C. A range of internal structure, including neutral beam shinethrough armor, limiters, baffles, divertors and pumps are expected to be implemented over the life of the experiment. Initially, an internal frame structure will be installed that can support graphite-weave panels in a range of shapes covering any part of the first wall. This design is robust and provides flexibility to modify the first-wall design to meet program needs. It will also allow plasma shape flexibility and experimental investigations of boundary physics issues, as well as other physics studies.

Fueling will be provided by a programmable multi-species gas injection system which can provide feedback control on the density. High vacuum will be provided by an existing turbomolecular pumping system

A total of 12 MW of auxiliary heating can be accommodated by the NCSX design, 6 MW of tangential neutral beam injection (NBI) and 6 MW of radio-frequency heating. Initially the facility will be equipped with 3 MW of tangential NBI using two of the four existing PBX-M neutral beamlines arranged in a balanced (1 co-, 1 counter-) configuration. The remaining two can be added to upgrade the NBI power to 6 MW. Two radiofrequency heating options are available as potential upgrades. High frequency fast wave heating utilizes fast magnetosonic waves at high harmonics of the ion cyclotron frequency, which minimizes ion damping while producing strong damping on the electron population. The operating frequency chosen for NCSX is 350 MHz. Ion Bernstein wave mode conversion heating uses a fast magnetosonic wave in the 20-30 MHz range, excited at the boundary of a multiple-ion species plasma. After conversion to the slow wave, damping can be on electrons or ions, depending on the ion temperature and species mix.

The facility will be equipped at first with the minimum set of diagnostics needed to support shakedown of major machine systems and the first few phases of physics operation, including first-plasma, electron-beam mapping of flux surfaces, Ohmic plasma experiments, and initial heating experiments. It is expected that an expanded diagnostic system will be necessary to achieve the full range of NCSX physics goals, and that these will be added during the operating life of the facility. Experimental results from the initial operating phases will help optimize the selection of new diagnostic systems and their design characteristics.

NCSX Engineering Design

A pre-conceptual design has been developed for NCSX. The NCSX design is built around the 3-period reference plasma configuration, with a major radius of 1.4 m. The plasma is surrounded by a vacuum vessel with an internal structure that can support molded carbon fiber composite (CFC) panels that are bakable to 350°C. The design features 21 modular coils, 21 toroidal field coils, and 4 pairs of poloidal coils located symmetrically about the horizontal midplane. The coils are pre-cooled to 80K. A cryostat encloses all of these coils. The modular coils, TF coils, and vacuum vessel are assembled in 120° segments. Each segment features ports for heating, pumping, diagnostics, and maintenance access. A cutaway view of the stellarator core assembly is shown in Figure 4.

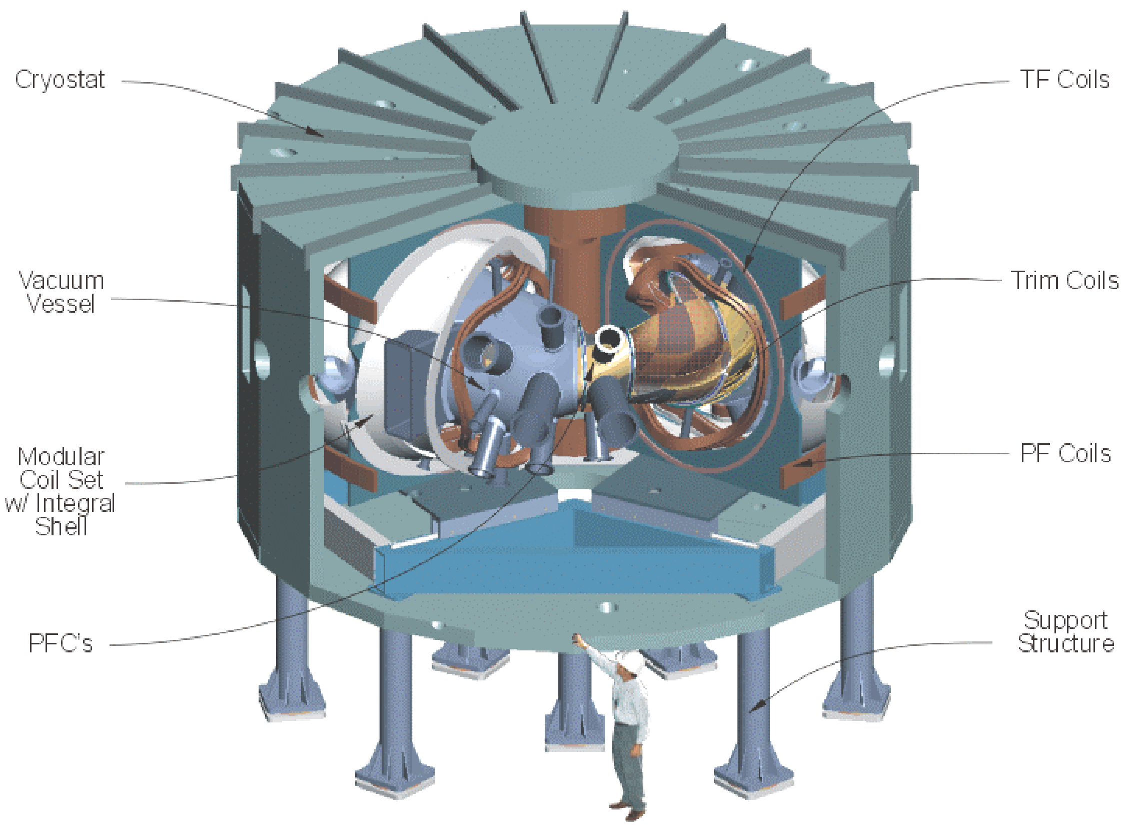

Figure 4. NCSX Stellarator Core

The NCSX will be assembled in the combined PBX/PLT test cell following removal of the PBX device. This location is well suited to NCSX. The test cell provides ample space for the device along with adequate crane capacity. The PBX/PLT computer and control rooms, which are contiguous to the test cell, will be refurbished and utilized. Many systems formerly used on PBX including the neutral beam, vacuum pumping, power supplies, and water systems will be reused. Power supplies located at D-site will also be used.

Plans for Completing the Design, R&D, and Construction

Following the Physics Validation Review, the review findings and recommendations will be incorporated into the project's plans for developing the design. The next step will be to update the reference NCSX design by September, 2001, incorporating results of ongoing design improvement studies. That update will be the basis for developing the conceptual design. A conceptual design review (CDR) is currently planned for April, 2002. It is expected that the CDR will formally establish the baseline design, cost, and schedule for NCSX. Approval to construct NCSX would be expected following a successful CDR.

R&D and physics analyses will be important to support the design development. The R&D activities will range from small-scale tests to establish design criteria to large-scale prototypes to establish manufacturing approaches and costs. Manufacturability input from industry will continue to be a feature of the design process. Physics analyses (e.g., coil-set flexibility, operating scenarios, boundary physics) will continue in support of the design process. Detailed engineering design is proposed to begin in FY-2003 with fabrication activities starting in FY 2004. For planning purposes it is assumed that First Plasma would occur at the end of FY 2006.

Cost and Schedule

Cost has been a prime consideration in establishing the design parameters for NCSX. The machine size and maximum toroidal field were established to keep the project cost at a target value of $55M in FY 1999 dollars, including contingency, while meeting the mission objectives. Assuming a four-year project (FY 2003-06), the cost in as-spent dollars is projected to be about $65M. This project has been categorized by DOE as a Major Item of Equipment (MIE) activity and the project cost defined accordingly. The preconceptual design work done to date supports our expectation that the proposed project can be carried out for this cost. Non-project pre-operational costs for NCSX research preparation and planning during the project period are projected to be about $4M in FY 1999 dollars ($4.6M as-spent).

Management Arrangements

The NCSX is jointly proposed by Princeton Plasma Physics Laboratory and Oak Ridge National Laboratory in partnership. These laboratories will lead the design, construction, operation, possible enhancements, and physics research for the NCSX project. PPPL has the lead responsibility for project execution. A management organization for the Project is established within the PPPL organization, reporting to the Department of Energy through the PPPL Director. ORNL provides major support, including leadership in key physics and engineering areas.

In carrying out the design and construction phases of the NCSX, PPPL will lead the project management team while ORNL has the lead responsibility for designing the stellarator core. The project will apply the management approach that has been developed and refined during the execution of numerous successful fusion projects. This approach is consistent with DOE Project Management guidelines and directives.

The physics and concept development phase of the NCSX project has been carried out by an integrated national team, with numerous U.S. institutions collaborating. The work has benefited from collaborations with foreign stellarator researchers (in Australia, Austria, Germany, Japan, Russia, Spain, and Switzerland). This approach has facilitated cost-effective knowledge transfer and resource sharing within the DOE system of laboratories and been effective in broadening national participation in the program.

The national team model will be continued as the project moves forward. The NCSX will be a national facility that will provide research opportunities for collaborators from many institutions. Collaborators will have opportunities to lead research areas, install innovative diagnostics and other research tools, conduct student research, and represent the project in scientific conferences.

Conclusions

Compact stellarators provide an important opportunity for the fusion program, offering unique capabilities to advance fusion science and innovative solutions to making magnetic fusion energy more attractive. The quasi-axisymmetric stellarator builds upon the advances in both tokamaks and stellarators and combines the best features of both. A sound physics basis has been established for a proof-of-principle experiment, NCSX, to further develop the physics. Dramatic advances in several aspects of the design have resolved earlier technical issues associated with the robustness of the equilibrium as it evolves from vacuum to high beta. A machine concept has been developed that shows that the NCSX scientific mission can be carried out in a practical and affordable facility. Plans are in place to move forward with the next phase of the project, conceptual design.

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