1. EXECUTIVE SUMMARY

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Contents

1.1. INTRODUCTION 1-1

1.2. THE STELLARATOR CONCEPT 1-2

1.3. SCOPE OF THE STUDY 1-3

1.4. REFERENCE CONFIGURATION 1-4

1.5. TECHNICAL ASSESSMENT 1-9

1.5.1. Plasma Physics 1-9

1.5.2. Engineering 1-9

1.6. SUMMARY AND CONCLUSIONS 1-10

1.7. ORGANIZATION OF THE REPORT 1-10

REFERENCES 1-11

1. EXECUTIVE SUMMARY

1.1. INTRODUCTION

The Stellarator Power Plant Study (SPPS) examines the key issues posed by the stellarator approach to future magnetic fusion central-station electric power plants. While the tokamak remains the primary approach to magnetic confinement of fusion plasmas in the various worldwide fusion R&D programs, the stellarator approach portends a viable, complementary alternative. From among the broad range of configurational options encompassed by the stellarator class, a modular-coil, four-field-period was selected as a baseline configuration for the present study, representing a consensus recommendation of the U.S. stellarator physics community of an attractive baseline concept, coupled with modern magnetic-fusion engineering design practice, incorporating safety and environmental considerations, consistent with the recent ARIES [1] tokamak conceptual designs.

The SPPS was launched and technical work was largely completed prior to the recent fundamental restructuring of the U.S. fusion program [2] in the face of budgetary reductions. Concept innovation and attention to alternative (non-tokamak) magnetic fusion concepts, with a view toward improved concepts for energy applications, are emphasized as themes of the strategy of the new program. Also, given the several strong stellarator programs outside the U.S., an opportunity to strengthen international collaboration has been afforded by this work. The results of the SPPS contribute to the assessment of the stellarator approach as a promising alternative concept [3].

Intrinsically steady-state, the stellarator requires no net plasma current and avoids problems associated with (tokamak) current-drive requirements, control with high bootstrap current fraction, major disruptions, and positional control systems and instabilities. While sharing a list of generic concerns with the tokamak, specific reservations regarding complexity and relatively low system power density have traditionally been attributed to the stellarator. These issues are addressed by the SPPS to elucidate those high-leverage factors influencing the attractiveness of this class of power plant.

Generally, conceptual power-plant design studies extrapolate from the present knowledge base to characterize the future economic and environmental features of competing approaches. Such studies are increasingly guided by an awareness of the needs [4] of the potential customer, be it a traditional utility or a non-utility generator (NUG) that might emerge in a (future) deregulated electricity market. The customer stresses competitive economics, public acceptance, and regulatory simplicity as desireable generic features of any suitable generating option. The conceptual power-plant design study provides a basis for the comparative assessment of feasible alternatives and contributes to the quantitative decision-making process characteristic of capital investment in genertating capacity or, even, of competing R&D programs.

The SPPS addresses a widely perceived need to consider nontokamak alternative power-plant options in the post-ITER [5] period. The SPPS assumed a deuterium-tritium (DT) burning plasma and the need to incorporate a lithium-bearing, tritium-producing blanket system. The steady-state coil system was based on superconducting technology. Passive-safety considerations are reflected in low-activation materials choices and limits in (afterheat) power density. Maintenance issues are addresed in the context of assuring high plant availability and consequently favorable economics.

1.2. THE STELLARATOR CONCEPT

The stellarator is a toroidal magnetic confinement approach producing closed magnetic confining surfaces by means of external conductors exclusively. The stellarator is one of the earliest approaches to magnetic fusion [6–8]. A number of review articles discussing the stellarator concept have been published [9–13].

The term "stellarator" (inclusive of torsatron and heliotron configurations) refers to a class of helical (*i.e.*, toroidal plus poloidal field components) magnetic fusion devices that produces confining magnetic fields by currents in external coils only. Relative to the tokamak, the stellarator promises disruption-free, steady-state operation with reduced recirculating power due to the absence of current-drive requirements. Such advantages may be offset by more complicated coil, blanket/shield, and divertor (*i.e.*, impurity control) configurations.

The fundamental stellarator configuration and the anticipated higher plasma aspect ratio, $A^\star \equiv R_T/r_p \simeq 8.5$, relative to the tokamak also result in different overall performance as monitored by the usual figures of merit (*e.g.,* power density, stored magnetic energy, peak coil magnetic field strength, neutron wall load, and projected cost of electricity, COE).

The future payoff of the stellarator approach is a possible post-ITER candidate for an attractive demonstration (Demo) power plant and subsequent commercial power plant, fueled with deuterium-tritium (DT), with attractive features and performance relative to the tokamak. Generally, the development of suitable structural materials and nuclear technologies can be anticipated to proceed under the (tokamak) ITER program, without unnecessary duplication for stellarator applications.

1.3. SCOPE OF THE STUDY

The Stellarator Power Plant Study (SPPS) can be said to follow from a letter proposal [14] made in 1990, proposing a three-stage study: (1) Optimization of the stellarator (physics) concept; (2) Integrated systems optimization; and (3) Realistic engineering development. A charter for a multi-year study was developed to "identify and assess the feasibility of critical issues and their consequences for development of the stellarator concept as a steady-state fusion reactor. The approach will be similar to that used in the ARIES and TITAN RFP reactor studies: an integrated physics, engineering, reactor component, and cost optimization. The same assumptions and algorithms can be used to compare different stellarators and tokamaks. The activities during the first year will focus on physics optimization and selection of one or more stellarator coil configurations for more detailed engineering design evaluation."

The U.S. SPPS began in 1993 with the investigation of candidate configurations. Interim results were reported [15–17]. Following investigations that included consideration of a continuous, helical coil configuration [16], a consensus emerged around the selection of the modular helias-like heliac (MHH) as the reference approach. Some preliminary results are summarized elsewhere [18], but the present report represents the most comprehensive documentation of the work.

The Engineering Phase, conducted in CY1994-5, adapted technologies from ARIES tokamak power-plant conceptual designs, where applicable, and considered the key configurational aspects of the stellarator. The latter phases of the SPPS effort concentrated on the development of reference power plant parameters, conceptual engineering of key subsystems, and integration of the Fusion Power Core (FPC), encompassing the magnetic-coil set, nuclear systems (*i.e.*, shields and tritium producing blanket), and plasma-facing components (*i.e.*, first wall and divertor surfaces).

Limited in scope and depth relative to the ARIES tokamak-based studies, the SPPS focussed on the consensus identification of a reference configuration, *i.e.*, the modular, helias-like heliac (MHH), and applied engineering assumptions, constraints, and extrapolations consistent with the ARIES tokamak studies to address key issues for the stellarator approach. Projected Cost of Electricity [COE (mill/kWeh)] estimates are reported as a figure of merit. The SPPS does not purport to be a fully integrated engineering design.

Participants in the study included a broad representation of the stellarator physics community, as well as a subset of the multi-institutional ARIES Team, augmented by specialized industrial support. Many other scientists attended the SPPS project meetings and contributed to the development of the concept. They are identified in the acknowldegements.

The SPPS technical work was coordinated by a series of workshops and project meetings involving the participants, as follows:

- Feb. 3, 1993 USDOE, Germantown, MD
- Feb. 16-17, 1993 Oak Ridge, TN
- May 5-7, 1993 Los Angeles, CA
- Jul. 24, 1993 Madison, WI
- Nov. 2, 1993 St. Louis, MO
- Jan. 5-7, 1994 Oak Ridge, TN
- Mar. 13, 1994 Dallas, TX
- Mar. 17, 1994 La Jolla, CA
- Jul. 12, 1994 Chicago, IL
- Sep. 12-13, 1994 Lisle, IL
- Nov. 6, 1994 Minneapolis, MN

1.4. REFERENCE CONFIGURATION

The stellarator magnetic-coil configuration can be produced by continuous, helical coils. The most recent example of this approach is that of the Large Helical Device (LHD) [19, 20], presently under construction in Japan. It is possible to project this basic configuration to commercial power plant scale [21]. Alternatively, modular-coil configurations, represented by that of the Wendelstein 7-X [22], which is also under construction in Germany. A corresponding power plant design has been developed [23]. The coil configuration adopted for the present SPPS is also of the modular variety and a variant of the Wendelstein/Helias type. The baseline MHH configuration has four field periods, produced by 32 modular, non-planar coils of four distinct types, as illustrated in Fig. 1.4-1.

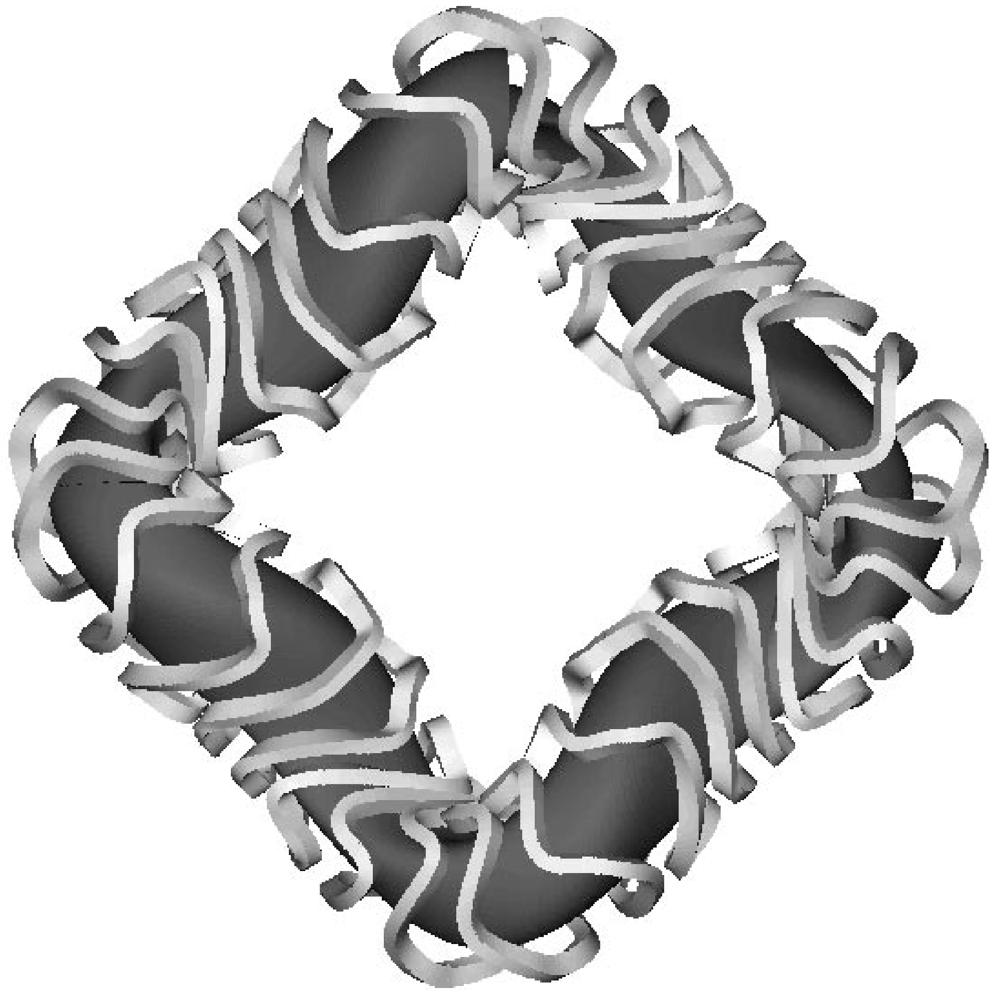

Figure 1.4-1. MHH modular-coil configuration, showing the last-closed flux surface of the plasma and the blanket/shield envelope interposed between the plasma and the coil set. The scale of the configuration is not specified here.

The SPPS projects a 1,000-MWe(net), DT-fuelled central power station, with key parameters summarized in Table 1.4-I. The target net electrical power output (design electrical rating) was selected to be the same as that of the ARIES and other recent conceptual designs to permit direct-cost (and other) comparisons. An alternative approach would be to postulate a common balance of plant (*i.e.,* same gross electrical power), letting the net power output of the stellarator increase as a consequense of the lower recirculating power fraction, and gaining a cost advantage as a result of so-called 'economies of scale'. There is engineering margin in the coil design and high-heat-flux surfaces to accomodate this option.

Recirculating power is low for the stellarator because of no current drive is required, as for tokamaks. System power density is typically lower because of the requirement for nominal stand-offs at higher plasma aspect ratio. The engineering configuration is complicated by the helical plasma geometry, but fundamental choices of materials and coolants are shared with the tokamak approach. A critical design constraint is the desire to minimize the standoff between the inboard plasma edge and the coil winding pack consistent with adequate tritium breeding in the lithium-bearing blanket and sufficient neutronic shielding of the coils. The reference blanket configuration invokes a vanadium-alloy structure (with CaO-coated coolant channes to reduce MHD pumping power) with low-activation properties that is cooled by flowing liquid lithium, which serves as the tritium-breeding material and the primary coolant. Operation of the blanket at high temperature allows an advanced thermal cycle with gross efficiency, $\eta_{TH} = 0.46$, as for the ARIES tokamaks. A nominal plant factor, $p_f = 0.76$, reflects an availability equal to that taken for the ARIES tokamaks. A more detailed comparative assessment would have to be mounted to make differential estimates of forced and scheduled outages, if the plant factor were to be used as a discriminator between stellarators and tokamaks.

The incorporation of low-activation materials allows the SPPS design to be passively safe under typical accident scenarios. Cost credits, reflecting the Level of Safety Assurance (LSA) [24], contribute to lower direct costs and lower COE projections, relative to a device relying on active safety measures [*e.g.*, Emergency Core Cooling System (ECCS) as for present fission plants].

The SPPS Fusion Power Core is illustrated conceptually in Fig. 1.4-2. Major systems are included, but such items as diagnostics, fuelling systems, coil busbars, *etc.*, that tend to obscure the device itself, are omitted in the interest of clarity. With artistic license, the plasma (last closed flux surface) is shown despite the vacuum having been broken during a maintenance withdrawal of a corner sector containing several modular coils and the underlying systems.

Table 1.4-I.
Major Parameters of the SPPS Power Plant

Number of field periods	4
Number of modular, non-planar coils	32
Plasma major toroidal radius, R_T (m)	13.95
Plasma half-width, a_p (m)	1.16
Plasma aspect ratio (*cf.*, tokamak), $A \equiv R_T/a_p$	12.08
Plasma aspect ratio, $A^\star = R_T/r_p$	8.54
Circularized (average) plasma radius, r_p (m)	1.63
Plasma volume, V_p (m^3)	734.7
Plasma beta	0.05
Lackner-Gottardi confinement multiplier, H	2.3
On-axis magnetic-field strength, B_o (T)	4.94
Peak magnetic-field strength at coil, B_c (T)	14.5
Stored magnetic energy, W_B (GJ)	80 (est.)
Fusion power, P_F (GWth)	1.73
Thermal conversion efficiency, η_{TH}	0.46
Thermal power, P_{TH} (GWth)	2.29
Gross electrical power, P_{ET} (GWe)	1.05
Net electrical power, P_E (GWe)	1.0
Recirculating power fraction, ϵ	0.052
Plant capacity factor, p_f	0.76
Total direct cost (B$)$^{(a)}$	2,249.
Total capital cost (B$)$^{(b)}$	4,340.
Cost of electricity, COE (mill/kWeh)$^{(c)}$	74.6

(a) Assumes 10th-of-a-kind learning-curve and certain safety-related cost credits.

(b) Assumes 6-yr construction lead time.

(c) Levelized COE, reported in (constant) 1992 U.S. dollars (1 mill = 0.001 dollar).

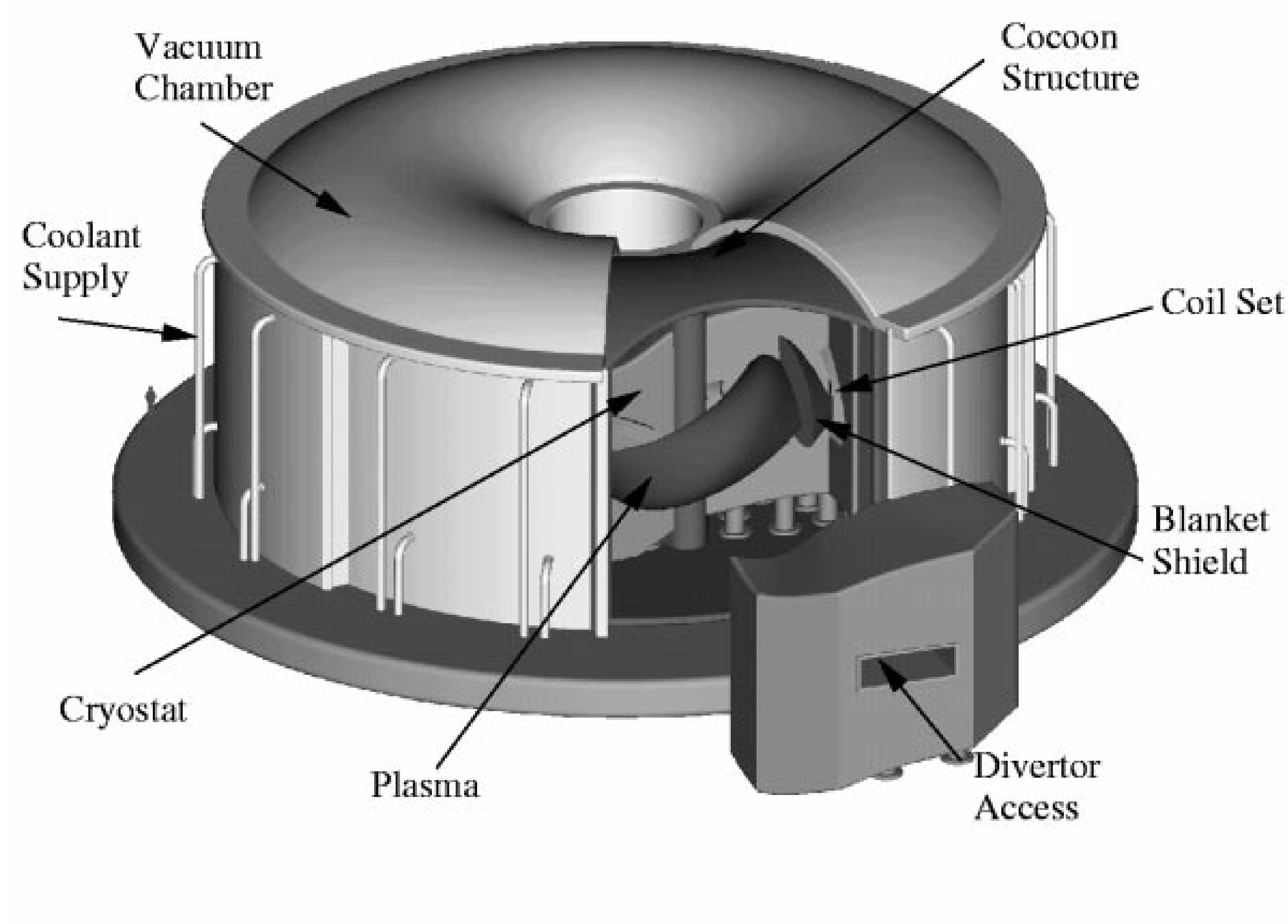

Figure 1.4-2. Isometric view of the SPPS Fusion Power Core.

1.5. TECHNICAL ASSESSMENT

The Stellarator Power Plant Study (SPPS) examines the key issues posed by the stellarator approach to magnetic fusion central station electric power plants. The remainder of this section examines the key findings of the study.

1.5.1. Plasma Physics

Physics issues for the MHH include equilibrium/stability at the reference beta value of 5%, the influence of magnetic islands on divertor performance, neoclassical transport in the presence of electric fields, suppression of the bootstrap current, and startup/control scenarios. MHH physics performance is projected to be adequate and robust.

The MHH configuration was chosen because its helias-like features led to good physics properties and its heliac feature, the large helical excursion of the magnetic axis that provides most of the rotational transform, allows the coils to be farther from the plasma, permitting a 2-m distance between the plasma and the coils for blankets, shields, structure, clearance, *etc.* for a 14-m average major radius. This new configuration has the best physics properties (beta limit, neoclassical transport, orbit losses, reduction of bootstrap current, *etc.*). However, its smaller plasma-coil separation leads to a larger major radius (larger blanket and shield area) and consequently higher cost.

1.5.2. Engineering

Engineering emphasis in the SPPS is on the coil design, neutronics optimization, integration of the blanket/shield/divertor systems into the MHH configuration, and assessment of a suitable maintenance approach. A suitable divertor/blanket/shield configuration can be accommodated in MHH-based fusion power plants and maintained. Preliminary, but serious, assessment of Nb_3Sn CICC graded, modular, nonplanar coils at 14.5 T is encouraging. The MHH as examined in the SPPS was found to be cost competitive with tokamak approaches. No unit-cost penalty was assessed for the complications introduced by the out-of-plane winding of the MHH coils. Such coils would be fabricated at a factory under 'clean' conditions and shipped to the plant site for installation, as for the ARIES tokamaks. For power-plant-scale installations, this approach to modularity is generally viewed as advantageous to the *in situ* winding of large continuous helical coils. Similarly, although life-of-plant coils are typically (and wishfully) assumed, the rare replacement of a failed modular coil is more acceptable than the repair/replacement of a continuous coil. Further reductions in the overall size of the fusion power core, with corresponding increases in the system power density, may be required to improve competitiveness.

1.6. SUMMARY AND CONCLUSIONS

Detailed conclusions may be found in the body of this report. In concise form, the following conclusions of the SPPS have been drawn:

- The Modular Helias-like Heliac (MHH) configuration, selected for emphasis in the SPPS, is one of a number of credible stellarator embodiments.
- The MHH physics performance is projected to be adequate and robust.
- A suitable divertor/blanket/shield configuration, based on structural/breeder/coolant choices in common with tokamak technologies, can be accommodated in MHH-based fusion power plants and maintained.
- Preliminary, but serious, assessment of Nb_3Sn CICC graded, modular, nonplanar coils at 14.5 T is encouraging.
- The reference SPPS MHH design point, with low system power density and low recirculating power, is found to be cost competitive with tokamak MFE designs.
- Interesting variations of the MHH configuration were not explored in detail, but could provide improved performance.

1.7. ORGANIZATION OF THE REPORT

This SPPS design report is devided into six major sections. Section 2 summarizes the physics basis of the Modular Helias-like Heliac (MHH). Section 3 summarizes the systems analysis activity supporting the selection of the reference design point. Section 4 provides a detailed discussion of the MHH coil configuration, internal design, and technical analysis. Section 5 describes the design of the SPPS fusion power core, emphasizing the issues of materials selection, neutronics, as well as configurational layout and maintenance. Section 6 examines several issues relating to safety and environmental aspects of the SPPS having broad application to the stellarator approach. Some detailed information appears in an Appendix.

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