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COMPARISON OF OPTIONS FOR A PILOT PLANT FUSION NUCLEAR MISSION*

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A fusion pilot plant study was initiated to clarify the development needs in moving from ITER to a first of a kind fusion power plant, following a path similar to the approach adopted for the commercialization of fission. The pilot plant mission encompassed component test and fusion nuclear science missions plus the requirement to produce net electricity with high availability in a device designed to be prototypical of the commercial device. Three magnetic configuration options were developed around this mission: the advanced tokamak (AT), spherical tokamak (ST) and compact stellarator (CS).

With the completion of the study and separate documentation of each design option a question can now be posed; how do the different designs compare with each other as candidates for meeting the pilot plant mission? In a pro/con format this paper will examine the key arguments for and against the AT, ST and CS magnetic configurations. Key topics addressed include: plasma parameters, device configurations, size and weight comparisons, diagnostic issues, maintenance schemes, availability influences and possible test cell arrangement schemes.

1. INTRODUCTION

The pilot plant (PP) study was initiated at PPPL in 2010 to explore the mission and design space for $Q_{eng} > 1$ devices which could convincingly demonstrate the credibility of fusion energy to the public. Three configurations, spanning the spectrum of current MFE designs, were investigated as candidate options: the advanced tokamak (AT), spherical tokamak (ST), and compact stellarator (CS). A paper by J.E. Menard [1] provided comprehensive coverage of the pilot plant study defining the details of the operating points for each option along with component sizing issues, radial build details, blanket and magnet system issues, candidate maintenance schemes, tritium consumption requirements, physics scenarios and an assessment of research needs. An engineering design overview of each option was provided in a paper by T. Brown [2] addressing basic configuration and maintenance schemes and component details, with prescribed build and space allocations.

The PPPL pilot plant study has reached a stage that allows an assessment of the ability of each candidate option to meet the pilot plant mission, to identify what R&D is needed to support further advancements and the degree of technical risk undertaken in pursuing an option. Where appropriate, comparative details between options will be provided. The pilot plant study funding level allowed development of only high level configuration and component design details with the majority of effort focused on the ST and AT options. The CS option was sized using a system code with configuration details developed around an upgraded ARIES-CS [3] design enhanced with improved maintenance features and downsized to the prescribed stellarator pilot plant operating design point.

2. STARTING CONDITIONS

2.1. Overview of Pilot Plant Mission and Requirements

The defining mission of the pilot plant is the production of net electricity Q_{engr}~1 with the accompanying task of meeting the fusion nuclear science mission to test and develop materials required to make fusion energy a reality. The pilot plant designs incorporate power plant relevant technologies to the extent possible to satisfy physics and technology pre-requisites for a first-of-a-kind power plant. The devices were designed for tritium sustainability necessitating machine sizes that would afford producing, extracting and processing the tritium required to operate the plant. Finally, the pilot plant options were configured for a steady state neutron, thermal, mechanical, material environment sufficient to address the multiple effect and integrated phenomena of a FW/blanket/shield/VV nuclear core prototypical of DEMO with the capability of achieving a 30% average operating availability.

2.2. Requirement Derived Component Features

Superconducting magnet systems were sized to take advantage of reduced cycles for steady state operating conditions, allowing magnets in the confined space of the inner bore to be designed with higher overall current densities than currently used on ITER. No assessment was made to evaluate magnet size reductions that might be attributed to improved quench protection with reduced S/C cable copper content or considerations for grading the conductor. The copper TF magnets for the ST option were sized to achieve net electricity Q_{engr}~1 with the inner TF leg center post (CP) dimensions and topology defined to minimize power consumption consistent with keeping it within stress limits.

The strategy followed for the blanket system [2][4] was to install "low-tech" robust, highly reliable versions of a baseline DCLL GEN-II blanket operated in a de-rated mode, limiting the temperature to ~ 450 C and to achieve temperature fields as uniform as possible to minimize thermal stresses. Upgraded blankets would be incorporated after experience is gained from earlier versions and from results from the test blanket modules that would be tested within the original base blanket.

In moving from an experimental device to the harsher environment of a pilot plant mission it is expected that diagnostic systems would impact the design. To minimize the impact of the diagnostics on the machine design, it was assumed that measurements are required for control and evaluation functions only and that additional dedicated systems to support a detailed scientific program are not included. Cautiously optimistic assumptions were made about diagnostic developments that are on-going in the diagnostic field, especially in the preparations for ITER, and which should be available by the time the detailed design of the pilot plant will be undertaken. A comprehensive study was undertaken by A. E. Costley to investigate diagnostic aspects dealing with the three pilot plant options [5].

2.3. Configuration Concepts and Design Philosophy

A primary design goal for the PP design activity was to define configuration arrangements for each option that had the potential to achieve high operating availability, differentiating it from present experimental devices. Design studies advancing high availability have been undertaken for a number of years within the fusion community incorporating a range of configuration arrangements with varying maintenance concepts [6-18]. Reasonable availability was evaluated in an ARIES-CS stellarator study [12][13] which incorporated an ITER style port maintenance scheme, assuming a tenth of a kind plant and optimistic maintenance assumptions. However, the predominate approach used to achieve high availability within the ARIES studies are configurations which incorporate the horizontal removal of a small number of very large in-vessel components, i.e. entire sectors. Moving away from the ITER style small port maintenance approach has also been justified within EU studies in their pursuit of a vertical maintenance approach based on segmentation of the blanket in large modules, i.e., the MMS (Multi-Module Segment) concept for DEMO [11]. Both horizontal and vertical maintenance schemes have been studied within JAEA DEMO reactor studies striving to define possible high availability configurations [16][17].

The ST configuration followed a vertical maintenance approach of earlier ST device studies [14][15][18] but with added variations brought on by the pilot plant mission and design choices. Specific design choices include: locating a vacuum vessel inside the TF coils, incorporating discrete TF coil legs that connects with a single turn TF centerpost with Felt metal sliding joints, defining a robust PF coil arrangement to achieve plasma shaping with some coils embedded inside the TF centerstack. Both the ST and AT options were designed for double null divertor operation, a departure from the approach used in ITER.

The requirement that drives all PP designs is the need to develop configuration arrangements that can effectively integrate the device core, auxiliary systems, test cell and maintenance operations to promote high availability operations – that eventually can achieve the 90% values present in current fission power plants.

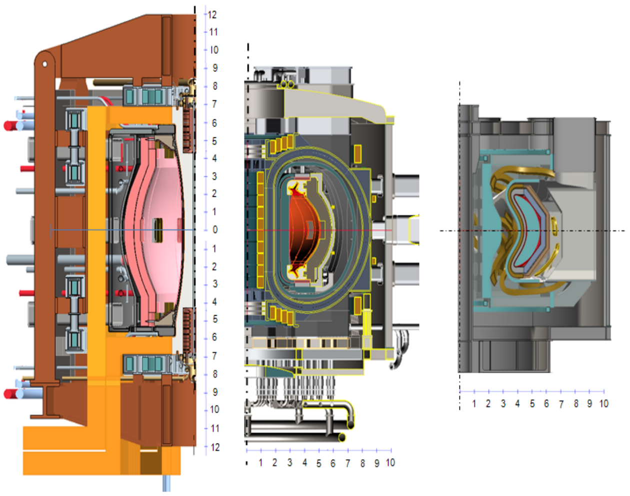
3. STUDY RESULTS

3.1. Parameter Summary and Size Comparisons

A size comparison among the three pilot plant options is illustrated in the half elevation views of Figure 1. Looking at gross measurements of machine height and overall diameter the ST is 24-m by 22-m, the AT is 17-m by 20-m and the CS is 15-m by 22-m with measurements taken from the outside of the ST support structure and from the outer dimensions of the cryostat for the AT and CS options. Table 1 summarizes the parameters of each device for two values of thermal efficiency η_{th} = 0.3 and 0.45; thermal efficiencies meant to span the range expected for candidate blankets.

TABLE 1 Power Plant Parameters

	η_{th}	A = R_0 / a	R_0 [m]	κ	B_T [T]	I_p [MA]	q_{95}	q_{cyl}	f_{BS} or iota from BS	n_e/n_G	H_{98} or H_{ISS04}	β_T [%]	β_N	P_{fus} [MW]	P_{aux} [MW]	Q_{DT}	Q_{eng}	<W_n> [MW/m^2]	Peak W_n [MW/m^2]
AT	0.30	4.0	4.00	2.0	6.0	7.7	3.8	2.40	0.69	1.00	1.22	4.8	3.7	674	79	8.5	1.0	2.2	3.1
AT	0.45	4.0	4.00	2.0	6.0	7.7	3.8	2.40	0.57	0.90	1.13	4.1	3.2	510	100	5.1	1.0	1.7	2.4
ST	0.30	1.7	2.20	3.3	2.4	20.0	7.3	2.80	0.90	0.70	1.35	39	6.1	1016	50	20.3	1.0	3.1	4.9
ST	0.45	1.7	2.20	3.3	2.4	18.0	7.8	3.00	0.86	0.70	1.34	31	5.3	645	60	10.8	1.0	2.0	3.1
CS	0.30	4.5	4.75	1.8	5.6	2.1	1.5	-	0.23	-	1.75	6.9	-	529	12	44.1	2.5	1.8	3.6
CS	0.45	4.5	4.75	1.8	5.6	1.7	1.5	-	0.19	-	1.60	5.7	-	313	18	17.4	2.4	1.1	2.2



ST – 2.2 m R_0 AT – 4 m R_0 CS – 4.75 m R_0

3.2. Component concept details

An effort was made to develop each PP configuration assuming near term technologies, at least in defining the basic device core components (magnets, VV, cryostat, supports, maintenance approach and bulk shielding). The blanket and divertor systems were assumed to be upgraded during the PP life time.

3.2.1 ST

Defining a single turn PP ST center post required special attention and review of design approaches considered in other ST designs [14][15][18]. A review of the low cost fabrication approach assumed for the 10th of a kind commercial ARIES-ST power plant was made along with discussions with principle investigators. An independent review of the ARIES-ST magnet system fabrication methodology by a service industry provider of additive manufacturing solutions was also solicited to evaluate near term feasibility issues [19]. The industry reviewer commented that near term application and cost projections provided in the ARIES-ST report [20] were very optimistic and that if a TF pilot plant system at ~60% size of ARIES-ST were needed in 10-15 years it would require a substantial development program that should be started as soon as possible. Technical issues were also raised regarding the viability of developing integral coolant holes considering the heat and size of the material being added and concern about the final material properties and costing details quoted for developing a spray cast outer return leg shell.

To keep within the criteria of using near term manufacturing techniques, a plate assembled centerpost design with discrete return legs and sliding joints was adopted for the ST option. The plates of the centerpost incorporate longitudinal holes that run the length of the plate, an approach proposed in other ST neutron source concepts [21]. A radial coolant option is being evaluated as an alternate approach which could prove to be more efficient and more applicable for a device the size of an ST power plant. To meet plasma equilibrium requirements a pair of PF coils were located within the ends of the flared centerpost (CP). Plenums are located at the top and bottom of the centerstack to supply and return water from the TF centerpost and the embedded coils, designed using Bitter plates. The Bitter plate coils are canned in a copper alloy structure which contains matched drilled coolant holes to interface with holes emanating from the plates of the TF centerpost. A separate coolant supply is provided to the centerpost and each PF centerstack assembly with a common return system. The TF centerpost would be dispersion strengthened copper-alloy Glidcop with furnace brazing of the entire centerpost/PF containment structure for a leak free system. The outer PF coils would be superconducting. Figure 2 shows the details of the ST centerstack design along with a local view showing the coolant header, sliding joint coolant services and mechanical connection details between the CP and the external structure; supports needed for the sliding joint. A thermal-hydraulic analysis has been carried out on the centerpost of a companion 1.6-m ST-FNSF device with identical design features with results showing acceptable thermal stress conditions [22]. The 525-tonne PP Centerstack assembly contains the CP, 10-cm VV/shield, FW and divertor modules located at each end.

The ST configuration allows the independent removal of the Centerstack (with divertors) and the full blanket assembly as a vertical lift as well as the removal of individual divertor modules in a radial direction through horizontal ports. The capability to remove

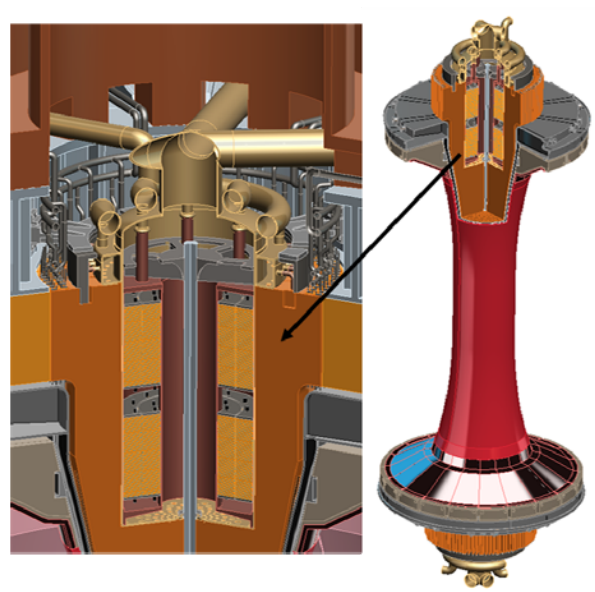


Fig. 2 ST Centerstack shown assembled with interfacing services and supports

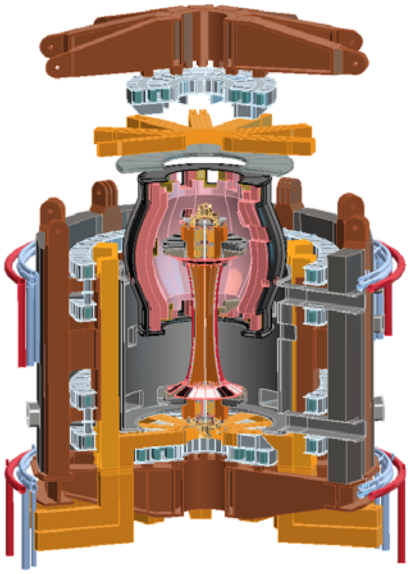


Fig 3. ST assembly exploded view

divertor modules independently was added to allow faster change out during the early operation/testing phase. Figure 3 shows an exploded view of the ST device highlighting the removal of the Centerstack and blanket assembly. To gain access to the blanket assembly or Centerstack requires the removal of all upper level components (pin connected support, TF horizontal legs, S/C PF cryostat, structure, VV upper lid) along with disconnection of services of all removed components. The full ST blanket assembly weight came within the 1500 tonne lift target set for the pilot plant study.

3.2.2 AT

The general arrangement of the AT option is shown in Figure 4 with the device core located in a test cell facility and maintenance casks attached to a bioshield roof. The AT design incorporates the advanced tokamak pilot plant physics parameters in a configuration developed to enhance access and foster operational availability [2][23].

A vertical maintenance approach was selected because of improved access and integration features with the device facility. Poloidal field (PF) equilibrium current sizing was also found to favor the vertical maintenance approach, when comparing PF arrangements needed for maintenance (horizontal vs. vertical). The in-vessel components are subdivided into twelve inboard shield modules, outboard blanket/shield modules and twelve blanket/shield modules located beneath the TF coils. The expanded TF and vacuum vessel allows sizing of continuous sub modules capable of accepting large ports. Divertor components are further sub-divided and can be

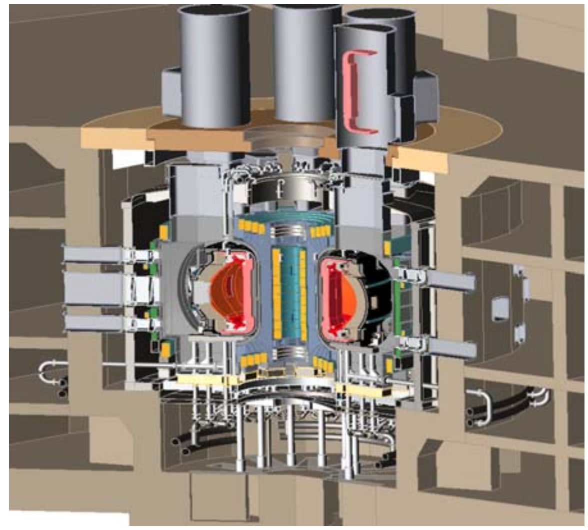


Fig. 4 AT device core shown with a test cell

maintained independent of the sector module. A semi-permanent inboard shield forms a strongback for supporting disruption loads, providing shielding for gaps between sectors and an alignment system for plasma components. Instead of supporting the internal blanket/shield modules from the vacuum vessel a lower base platform is included that also serves as a coolant plenum to service the FW/blanket modules. Replacement of all components would be from above, consistent with the expected initial vertical machine assembly process. Space is available to concurrently service half of the in-vessel components at any one time. An enlarged TF coil was used to minimize the number of segmented plasma components and allow increased space on the outside of the blanket for maintenance. The overarching design philosophy has been to expand the maintenance coverage of the in-vessel components from below, around the peripheral of the device and from above in an effort to meet the high availability goal.

3.2.3 CS

The CS design was carried out as an engineering effort to identify concepts that would improve stellarator maintenance features; concepts that could be then tested and iterated with physics to arrive at a self consistent plasma magnetic / engineering compatible arrangement. The ARIES-CS design was used as a starting point and reconfigured in a concept similar to the design developed for the AT option; expanded vacuum vessel and vertical maintenance of in-vessel components. The Type-A and B modular coil back legs were straightened to provide access from above, local trim coils were added for plasma shaping and the general shapes of the blanket/shield modules were simplified (details shown in Figure 5).

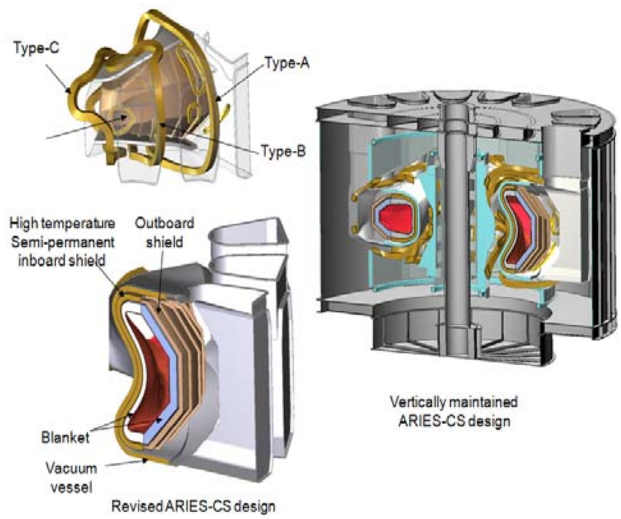


Fig. 5 SC configuration concept details

Concepts were also developed that used passive high temperature superconducting (HTS) tiles to help shape the plasma and provide large access for maintenance [3][24]. In addition to concept investigations, plasma studies were carried out looking at aspect-ratio scans for improvements on magnet complexity and computational searches to identify attractive quasi-symmetric configurations [25][26].

4. CONCLUSIONS

The engineering effort for the pilot plant study was focused on top level configuration issues that would impact high availability operations. Subsystem components were sized to expected requirements and configuration driving components and services were defined in an effort to identify design and R&D issues that would require further study. At this point replacement times for in-vessel components were not estimated to develop availability numbers for the different PP options; however, with the details developed a general conclusion can be made to determine if a configuration approach has merit in meeting the availability goal. With most of the effort concentrated on the ST and AT options, more informative results can be made for these options. Table 2 provides a comparison of some machine parameters and component sizes. The smaller major radius ST device has the lowest plasma surface area but the highest plasma edge heat load and corresponding value of $P_{alpha+aux}/R_0$, a metric used to measure divertor heating conditions. Comparable weights exist between most of the AT and ST components except for the TF systems. Here the ST windings were sized for reduced power consumption of the copper coils and the external structure, partially included with the building structure, was assumed to carry all the magnetic loads.

	ST	AT	CS
Major Radius, R_0, m	2.20	4.00	4.75
Plasma volumn, m^3	192	146	104
Plasma surface area, m^2	227	234	266
P_{fus} (MW) - 0.45 thermal eff n_{th}	645	510	313
P_{aux} (MW) - 0.45 thermal eff, n_{th}	60	100	18
$P_{alpha+aux}$ / S, MW/m^2	0.83	0.86	0.30
$P_{alpha+aux}$ / R_0, MW/m	86	51	17
Blanket/shield - tonne	1364	1370	
Divertor (upper) - tonne	40	76	
PF winding - tonne	910	921	
TF winding - tonne	5893	360	
TF structure - tonne	3033	915	
Cryostat -tonne	-	1055	

Table 2 Select Pilot Plant Comparison Data

Availability values have not been calculated at this time to gage the ability of each option to meet the target 30% average availability goal; however, there appears no evidence that any one option has a superior advantage. Although the ST configuration allows the removal of a complete blanket system, it will require disassembling major subsystems to gain access which can add risk in damaging the disassembled components. In moving from a pilot plant facility to a full power plant operation the AT and CS maintenance approach appear more feasible than the ST configured option due to the size of the components being handled and greater facility integration complexity. All three options will confront design issues dealing with the divertor. The ST has the highest divertor heat load and will need the integration of new divertor concepts (Super X, snowflake…) to bring the heat load down to manageable levels. The CS has more tenable divertor heat loads but can expect greater complexity in their design and maintenance features. Developing a viable current drive system for the ST and AT option is an open issue with respect to demonstrating a credible tokamak scenario with very high bootstrap fractions and economic efficiencies of external H&CD systems. H&CD arrangement and device/facility interfacing details are more onerous for the ST and AT options compared to requirements needed for a CS design. Figure 6 shows an arrangement of six 10 MW JT-60SA NNB injectors surrounding an ST pilot plant; 16.5 MW ITER NNB's also could be used. The AT option can have a different mix of H&CD systems ranging from 50 MW EC and NNB's in 4-5 ports to 3-4 ports assigned to 25 MW IC and 75 MW LH (depending on power densities).

The dominant pros and cons for each option are listed below:

ST option

Pro

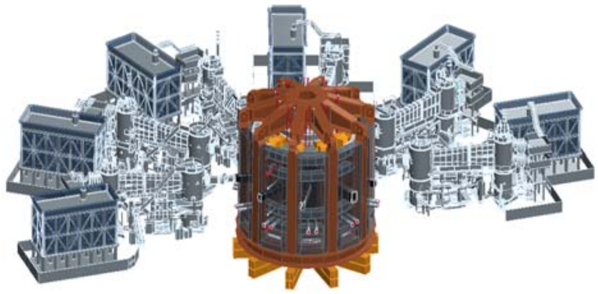


Fig. 6 JT60 NNBI interfacing with the ST device

- ST physics offers a special class of low-aspect-ratio wall-stabilized high-β high-bootstrap fraction tokamak equilibrium.
- The ability to assemble a full blanket system before installation in the device core simplifies alignment.
- The external assembled blanket system may benefit development of a simplified disruption support system.
- Low-aspect-ratio enables higher wall loads to be developed in a given size.
- Jointed TF coils allow the replacement of in-vessel components located within the TF boundary.

Con
- Low-aspect-ratio plasmas allow little inboard space for shielding, preventing the use of superconducting TF or OH magnets.
- Copper TF coils result in high circulating power and the need to size the device to compensate for its use.
- Lack of inboard space for shielding prevents the use of in-board plasma control diagnostics requiring the need to develop alternate plasma control solutions.
- The lack of inboard shielding results in the need to replace the TF centerstack with a frequency similar to the blankets.
- Jointed coils operated in steady state conditions may have higher failure rates; reliable steady state operation of jointed TF coils needs to be demonstrated.
- Copper TF coils sized for power balance and sliding joints results in heavy components and support superstructure.
- Maintenance of a full blanket assembly within the test cell and interfacing cask is complicated by the size of the component.
- To minimize power losses for large conductor currents requires power supplies (conventional or homo-polar generators) to be located very near the device, complicating interfacing details of competing auxiliary equipment and services.
- Developing a viable current drive system remains an open issue.

AT option

Pro
- The plasma can be sized to allow sufficient inboard space for blanket/shield, plasma control diagnostics and superconducting TF and PF coils.
- Plasma physics allows a device to be sized with wall loading and divertor heat loads that is more amenable to material limits.
- Continuous superconducting TF coils should afford high reliability operation with technology advancement offering further improvements.
- The configuration developed allows in-vessel components to be sized for easier integration with maintenance cask and facility.

Con
- Although an intermediate disruption support shell structure has been added to the AT option, the ability to survive disruption loads needs to be demonstrated.
- Developing a viable current drive system remains an open issue.

CS option

Pro
- The stellarator has the potential of solving two limiting impediments of the tokamak design - high-beta disruption-free operation without plasma control and operation with low circulating power without the need for current drive.
- Operates with lower surface and divertor heat loads.
- A design based on quasi-axisymmetric shaping results in a smaller stellarator device, more in line with tokamak sizing.

Con
- The coil system geometry used to form non-axisymmetric shaping of the plasma result in complex configuration designs with more complex maintenance approaches. Alternate concepts as described in the PP study need to be pursued.
- 3-D shaping results in more difficult design practices and more complex part manufacturing.

5. SUMMARY

The PPPL pilot plant study attempted to develop three candidate options around a common mission to better understand and evaluate technical issues of a prototypical magnetic fusion power reactor. Progress was made in defining configuration concepts for each option, understanding technical issues and defining areas where further development is needed to move forward. Further study of the PP ST option sized to meet a fusion nuclear science mission is warranted. However, the continued assessment of a superconducting AT (or CS) power plant device needs to be made, as the marginal cost associated with carrying out an FNS mission in a pilot plant facility may be very small.

All conceptual studies that make it through a final construction process can expect to see some changes in concept definition, component sizes, and overall cost projections – it is hoped though that these preliminary studies can be used to see the trends and define a path forward.

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