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An Overview of Pilot Plant Designs Based on the Advanced Tokamak, Spherical Tokamak and Stellarator*

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Abstract— **A fusion pilot plant study was initiated to evaluate the potential benefits of following the fission development path as an approach for the commercialization of fusion. In such an approach, a fusion pilot plant would bridge the development needs in moving from ITER to a first of a kind fusion power plant. The pilot plant mission would encompass the component test and fusion nuclear science missions yet produce net electricity. In the first phase of the study scoping designs were developed for three different magnetic configuration options: the advanced tokamak (AT), spherical tokamak (ST) and compact stellarator (CS). Critical component features have been added to the designs that impact the general arrangement and maintenance characteristics of each device. The requirements specified in defining the pilot plant challenge the machine configurations developed for each option. Developing multiple options with a consistent set of requirements enables a uniform comparison of configuration and component issues that drive each design. This paper will provide an engineering design overview of each option, address open issues and assess where further work is needed to meet the pilot plant objectives.**

Keywords- advanced tokamak (AT); spherical tokamak (ST); compact stellarator (CS), component test, pilot plant

I. INTRODUCTION

A potentially attractive next-step toward fusion commercialization is a pilot plant which produces net electricity in a minimum size facility with a configuration that is directly scalable to a power plant. Three configurations are being investigated as candidate options: the advanced tokamak (AT), spherical tokamak (ST), and compact stellarator (CS); devices that span the spectrum of current experiments. The tokamak presently has the most well-developed physics basis, the ST offers a potential lower cost copper TF coil option and the CS offers disruption-free operation with low recirculating power. A recent paper [1] presented initial study details that covered a range of configuration issues including: radial build details, blanket and magnet systems, maintenance schemes, tritium issues, physics scenarios and a brief assessment of research needs. System code sizing studies were performed for each option to establish the starting design point for each configuration. 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. A preliminary set of top level system requirements have been defined to guide the design effort. The requirement to achieve high availability singularly drives each configuration to subdivide in-vessel components into a small number of large segments and to provide the access space to remove them. The pilot plant targets power plant relevant technologies with configuration features of a commercial fusion device so that through its design, construction and operation relevant experiences can be gained.

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}	$\langle W_n \rangle$ [MW/m^2]	Peak W_n [MW/m^2]
AT	0.30	4	4	2	6	7.7	3.8	2.4	0.59	0.9	1.2	4.6	3.6	553	79	7.0	1	1.8	2.9
AT	0.45	4	4	2	6	7.7	3.8	2.4	0.5	0.8	1.1	3.9	3	408	100	4.1	1	1.3	2.1
ST	0.30	1.7	2.2	3.3	2.4	20	7.3	2.8	0.89	0.7	1.35	39	6	990	50	19	1	2.9	5.2
ST	0.45	1.7	2.2	3.3	2.4	18	7.8	3.0	0.85	0.7	1.3	30	5.2	630	60	10.5	1	1.9	3.4
CS	0.30	4.5	4.75	1.8	5.6	1.7	1.5	-	0.2	-	2	6	-	529	12	42	2.7	2	4.0
CS	0.45	4.5	4.75	1.8	5.6	1.7	1.5	-	0.2	-	1.6	6	-	313	18	17	2.7	1.2	2.4

In sizing the devices initial sets of radial build data were established that reflected physics requirements, assembly issues, component tolerances and technology constraints. The build data were identified as starting values that would be reviewed and updated as the design details evolved. The planned evolution for each design option involves a three-

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step process: first, developing a basic configuration arrangement that meets a planned maintenance scheme with prescribed component features, following build space allocations; the second phase, adds critical configuration-driving details which impacts the maintenance or space allocations and the final phase reassesses build information, reruns system codes, resizes each candidate option and performs a high level availability assessment. The pilot plant study is currently midway into the second phase. Design details that have been developed along with a general status assessment will be presented.

II. Component Features and Design Philosophy

Pilot Plant design options are evolving beyond the initial scoping study phase to the point where configuration driving details for in-vessel systems, diagnostics, heating systems and their services are being added to the designs. Basic design philosophies are also being established to guide the component details. It is assumed that a preliminary design phase for a pilot plant would start in about 10 - 15 years. Therefore the design bases for the component details (where applicable) will be derived from ITER or other current experimental devices with the addition of expected improvements in technology or design. Background ideologies used in defining some of the pilot plant component details are listed below.

A. Magnet Systems

For superconducting magnets discussions are in process to determine if allowed current density values can be increased over what presently exists on ITER or W7-X. There are a number of design choices that can be made to reduce a coil size. The use of aggressive quench protection could reduce the S/C magnet copper content and thereby the cable size; grading a TF magnet can be beneficial but defining an effective structural support system and layer joint arrangement needs to be developed; reducing the number of cycles allows operation at higher stresses and can result in reduced structural area. Any increase in magnet current density values over those used in present experimental devices will be documented.

B. Blanket /Divertor Systems

The strategy being followed for the blanket system is to initially install a “low-tech” robust, high reliable version of the baseline breeding blanket design. Essentially, the base blankets would be the GEN-II blankets operated in a de-rated mode. The plan is to degrade the performance of the blanket system in the initial operating phase by limiting the liquid metal (LM) and He exit temperatures to ~ 450 C to achieve a uniform temperature field (as much as possible) to minimize thermal stresses. The low exit temperature would allow replacing SiC flow channel inserts by sandwich inserts (steel-Alumina-steel) in case the SiC-inserts cannot be developed in time. However, the LM exit temperature can be raised only after base blankets with SiC inserts have been installed. To avoid an early exchange in the base blankets, it is desirable to have the SiC inserts ready and installed for the start of the power plant operation but to operate the plant in an initial phase at lower performance and take advantage of the results gained from the more advanced test blanket modules (TBM) in test ports before the base blanket LM exit temperature is raised to its design value. The consequence of a starting with reduced exit temperature limits the potential for generating electricity unless the helium of the BRAYTON cycle power conversion system is heated with burning gas from ~ 420 C to ~ 650 C. With later versions of the base blankets, the pilot plant will achieve the design value of electricity generation. In order to get valuable statistical results about blanket reliability, and to have a transition to the blankets with the higher performance needed in the DEMO plant that follows, the pilot plant base blankets will need to be replaced a few times. The upgraded blankets will employ the experience gained from previous versions of it together with the results from the TBM’s that would be tested together with the original base blanket. The TBM serves as fore-runners of advanced base blanket versions. Advanced test blanket modules will be located at the horizontal midplane at 3-4 ports to enable "reliability growth testing" of the blanket concepts. Dedicated coolant supply pipes and instrumentation will service the TBM. The design goal is to limit the down-time for the replacement of a TBM and allow parallel operations on all blanket test ports (especially for liquid metal breeder blankets). The specified size of the TBM is 1.5 m - 2 m high by 0.5 m - 1 m wide.

The blanket geometry is required to be conformal to the plasma shape to maximize the tritium breeding ratio (TBR). The goal of the pilot plant is reach a TBR > 1, at least after a short initial phase with low tech base blankets. Where the low tech blanket system initially might impact the ability to reach the TBR goal, it is hoped that coupled with the installed advanced blanket modules, it would take only a few 100g of purchased tritium per year to make up any shortfall. It is expected that the divertor systems will have a higher maintenance and repair rate than the blanket system, thus it is desirable have the capability to repair or replaced them as separate units. A second in-vessel replacement feature that tokamak candidate design options will strive to incorporate is the ability to replace local FW panels in areas where runaway electrons might cause damage or other areas where FW surfaces are prone to have high failure rates or shorter operational lifetimes.

An open issue to be resolved during this power plant study is how to handle more advanced versions of the original blanket concepts. Should flexibility be added to the pilot plant mission to allow installing base blankets of different concepts, such as He cooled solid breeder blankets and/or self-cooled PbLi blankets based on SiC-composites as structural material with the implied configuration impacts on physical size requirements, supplied services and auxiliary equipment?

C. Diagnostic Systems

As on existing machines, measurements will be required of the main plasma parameters and the condition of the first wall, and some of the measurements will be used in real-time control loops to ensure that the required plasma performance is achieved and to prevent operation where potentially damaging off-normal events can occur. The measurements will be provided by the diagnostic system. The implementation of the diagnostic system on a pilot plant is a significant challenge because of the relatively harsh environment - high levels of radiation, high thermal and mechanical loads - restricted access, and the combined needs of high reliability and maintainability. An initial dedicated study is being made for each of the pilot plant configurations.

A five-step process is being followed for each configuration. The physical parameters that have to be measured are determined from a consideration of the intended plasma operating mode(s); diagnostic systems are selected to provide the measurements; a first level integration with the main interfacing machine components - vacuum vessel, blankets, ports, divertor - is carried out; an assessment of the environmental effects on the performance of the integrated system is made; and the system is adjusted to try to minimize the environmental effects while at the same time maintaining the required performance. The output is a first level integrated system design, which, in principle, should meet the measurement requirements, but with some open design questions and identified areas where R&D is needed. To minimize the impact of the diagnostics on the machine design, throughout the process it is assumed that measurements are required for control and evaluation functions only - additional dedicated systems to support a detailed scientific program are not included - and cautiously optimistic assumptions are 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.

As in the case of other areas of the pilot plant development, the diagnostic activity is still ongoing. Nevertheless some significant findings are already clear. For example, for the ST, the blankets are currently configured as large poloidal segments. From the point of view of radiation shielding and cabling, the best location for the sensors of the magnetic diagnostics is on the back-wall but the coupling to the plasma will be significantly influenced by the presence of the (electrically conducting) blanket modules. Detailed calculations are required to determine the potential diagnostic performance with this arrangement and if it will be adequate for plasma control. An additional difficulty with the ST is the absence of a blanket on the inboard side. Any magnetic sensors in this location would be subject to high levels of neutron radiation and this would lead to the generation of spurious signals (prompt effects) and to changes in the characteristics of the sensors due to material damage (accumulative effects). It is probably unrealistic to think of sensors in such an exposed location. In principle, sensors could be mounted in the skins of the vacuum vessel, or integrated with the CS centre stack, but then currents in the vacuum vessel would significantly disturb the measurements, and the radiation level would still be high leading to spurious signals and potential sensor damage. Alternative means of making the measurements provided by these sensors, in particular measurements of the plasma shape and position are therefore highly desirable and constitute one area of diagnostic R&D that would be needed to support the ST pilot plant. The AT, which has a blanket on the in-board side, and the CS, which will not rely on these measurements for plasma control, will be less affected by this effect but have yet to be studied in detail.

Some measurements require optical systems, for example first wall and divertor viewing and, as experience on ITER has shown, these can be realized only by using optical labyrinths imbedded in shielding blocks and plasma facing mirrors. The mirrors are subject to erosion and deposition and so are sources of significant risk to the system performance. Mitigating methods for handling this risk are under development for ITER and will certainly be needed for a pilot plant.

It is clear that in order to achieve a credible integrated design of a pilot plant, diagnostics need to be considered at this early stage. Modifications in the machine design to minimize the diagnostic problems where these are significant should be made if possible. Areas where developments in diagnostics are needed will be a natural output of this work.

III. Pilot Plant Configuration Arrangments

The pilot plant study is at the midpoint of the second phase and the configuration development of each option is not at the same level of completeness. An ST has been defined with a vertical maintenance scheme and blanket details and services have been added. Two physical arrangements are being investigated for the AT option that involve either vertical maintenance or horizontal maintenance schemes with blanket details developed. A preliminary arrangement has been developed for the stellarator but further work is needed to resize it for the proper design point for the pilot plant.

A. Spherical Takamak (ST)

The ST device sizing follows the parameters established by J. Menard et.al. [1]. The configuration is driven by a collection of design choices that include: locating a vacuum vessel inside the TF coils, incorporating discrete TF coil legs that connects with a single turn TF centerpost, defining a robust PF coil arrangement to achieve plasma shaping and defining an arrangement that allows vertical access from above to remotely maintain the internal plasma core components. A secondary configuration design goal is the replacement of the divertor components through midplane ports without the need to remove the entire upper machine sections to gain access. To minimize resistive losses large cross-section return legs are incorporated and external superconducting PF coils are located outside the TF coil. For assembly and maintenance a goal was set to define an arrangement where components or assemblies to be handled would be less than 1,500 tonnes.

1) *General Arrangement:* The general arrangement of the ST pilot plant design is shown in Figure 1 illustrating the basic features of the design. A water cooled copper picture frame TF coil arrangement incorporates a flared single turn nonolithic TF centerpost connecting to ten large cross-section return legs. Felt metal sliding joints located at the coil interfaces allow the coil horizontal legs to be removed for vertical maintenance of the internal plasma components. To minimize the resistive losses, external superconducting PF coils located in vacuum enclosures are used for the outside PF ring coils. Copper PF coils are also located inside the upper/lower region of the TF coil centerpost to help perform divertor shaping. Considerations were given to locating these coils inside the TF geometry, but when balancing TF coil power, plasma shaping and shielding requirements; the preferred location was found to be within the center region of the TF center leg. The vacuum vessel is located within the bore of the TF coil system and is designed to have a section at the top to be removed to gain access to the plasma components located within the vacuum vessel. External TF coil support structure is used to support the magnet loads with multiple finger connections and pin joints used to connect the external structural members. An exploded view of the ST device is shown in Figure 2 illustrating the basic component features that will be involved in the assembly and maintenance of the blanket sytem independent of the TF centerpost. The ST configuration was altered from the initial study phase to allow the TF centerstack and outer blanket system to be removed independently. To accomplish this, the lower vacuum seal weld was moved radially inward and the base component features altered to provide space for an external welder/cutter.

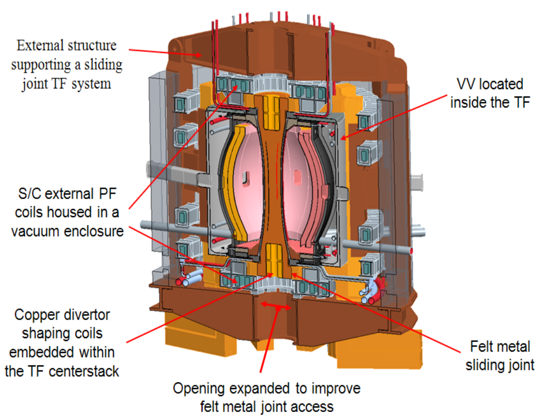

Figure 1. ST Pilot Plant Isometric View

2) *In-Vessel Coolant Services:* In-vessel coolant services for the blanket, inboard first wall and inboard

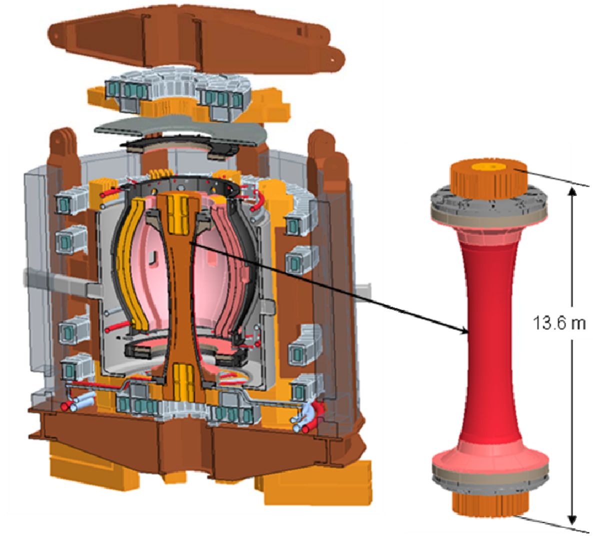

Figure 2. 3-D Maintenance View

vacuum vessel have been defined. The arrangement developed is integrated within the overall maintenance scheme with access space provided to perform pipe weld/cutting operations. The opening in the TF coil exterior support structure in the area of the centerstack felt metal joints was expanded to increase access to the joints and to the coolant services that will be defined in this phase of the design study.

3) *Auxiliary Systems:* Nine mid-plane neutral beam ports arranged in a pinwheel orientation, interface with JT60-SA NNBI systems which was found to be well matched to the ST plasma requirements. Diagnostic ports would be located above and/or below the beam ports. A single large mid-plane port would be available for diagnostic systems at the remaining open port. Three to four advanced test blanket modules will be located at the horizontal area between the NB ports.

B. Advanced Takamak (AT)

The maintenance philosophy for the AT pilot plant design is being evaluated with a renewed study of the horizontal and vertical maintenance scheme. Candidate AT configurations are based on 16 and 12 TF coils and include heating systems, diagnostics, in-vessel coolant services, PF coils positioned for plasma equilibrium requirements, and maintenance systems. When the designs are complete an overall assessment will be made between the two design approaches to determine if a clear preference can be identified. The ARIES-RS [2] and AT [3] studies promoted horizontal maintenance of large torus sectors between expanded TF coils as a solution for high availability given reduced downtime for replacing components that have fewer external coolant services and mechanical interfaces. Recent European studies have looked at an advanced blanket maintenance concept for their DEMO using multi-module blanket segments (MMS) that involved toroidal in-vessel transport and vertical insertion through a limited number of remote handling ports [4][5]. The new MMS reactor integration and maintenance concept suggests promising features and potential high availability. Similar configuration studies were carried out much earlier on INTOR (an international tokamak reactor study) [6][7] held over the period 1981 thru 1987. It is clear from past reports that extracting a fewer number of very large integrated in-vessel components through large horizontal ports will have an availability advantage over removing an increased number of in-vessel components with more service connections through vertical ports when evaluated in the absence of auxiliary systems that interface with the fusion core at or near the midplane. However the assessment of a fully integrated fusion device may indicate there may not be a significant difference in operating availability between the two approaches. One issue being evaluated is the ability to define a vertical port of sufficient size that is compatible with a PF arrangement that can effectively shape a double null high triangularity plasma. The European vertical maintenance DEMO concept was developed with a single-null plasma following the ITER parameters.

1) *General Arrangement*: The general arrangements for the horizontal and vertical maintenance AT schemes are shown in Figure 3 as defined for a 12-TF coil system. The horizontal maintenance approach requires larger TF coils to allow extraction of an integrated in-vessel sector between TF coils. This increase coil size favors reducing the coil number to 12 coils from 16 as it does not impact ripple conditions. The TF coil size actually was increased over a ripple defined size for 12 coils due to the additional toroidal space needed between coils to extract the in-vessel module. To minimize the toroidal extent between coils, local shield segments are provided in the shadow of the TF coils. This arrangement follows designs of past studies developed for INTOR and ARIES. If the removed module becomes too large it could be subdivided into two segments, one incorporating a radial movement and one a toroidal plus radial movement or two unequal modules could be extracted between TF coils in a straight line motion but at different angles.

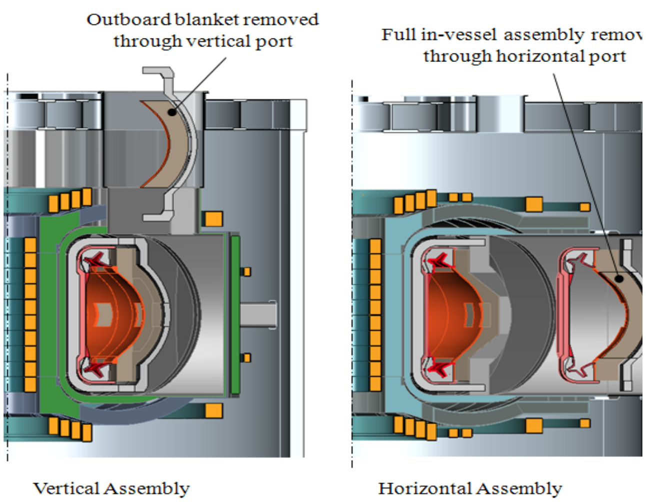

Figure 3. Horizontal and vertical maintenance schemes

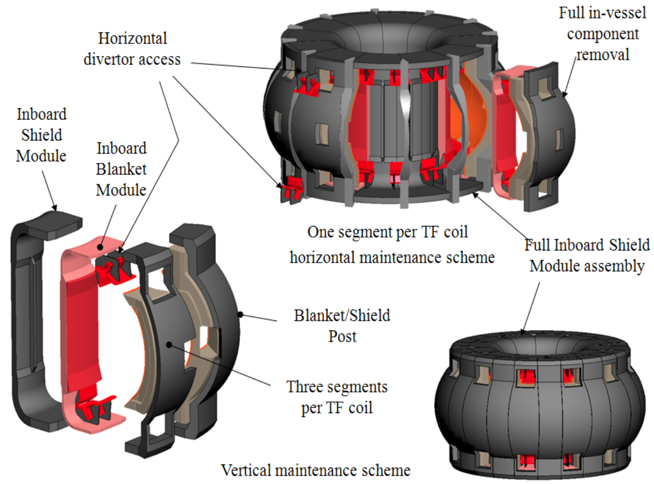

Figure 4. Blanket/shield system

2) *In-Vessel Component Arrangement:* The details of the in-vessel components developed for both maintenance schemes are shown in Figure 4 highlighting the different components that make up the blanket/shield system for both options. For the 12 TF coil arrangement with vertical access, there are three sub-modules associated with each TF coil to be removed: an outboard blanket/shield module, an outboard blanket/shield post, and an inboard blanket module. All outboard blanket/shield module interfaces have a separation gap formed by labyrinth geometry to reduce streaming conditions. Twelve inboard shield modules also make up a semi-permanent "C-shaped" ring structure that is used for gravity support and to support disruption loads. The horizontal sector maintenance option is shown with one integrated port module with semi-permanent shield posts located under each TF coil. The divertor system for both schemes are sub-divided into 24 upper and lower units, sized to allow them to be retracted independent of the blanket modules. One set can be extracted with radial motion; a second divertor module requires a toroidal movement before it can be retracted in the radial direction. Currently the inboard shield is separated from the inner blanket module in both concepts making it a semi-permanent structure which can act as a gravity support and potentially supporting disruption loads with a simpler interface. It is realized that this arrangement would defy following a minimum size path as additional shield material would be added to the inboard blanket module to allow the semi-permanent shield structure to be capable of being rewelded; alternately an inboard shield member could be developed with mechanical interfaces using shear keys and bolts. Rather than striving to develop a minimum size

device a goal of the pilot plant study is high availability operation which warrants investigating simplified design approaches. The European vertical maintenance Multi Module Segment system supports the high temperature components off a lower temperature vacuum vessel using long bending bars located along the inner VV surface and at the lower outside surface. ITER internal components are also supported off a very large vacuum vessel which in turn is supported from the superconducting TF coils. The blanket/shield structure for the pilot plant and for a commercial device will be larger, heavier units operating at high temperature. The approach being studied here is to reduce the structural size of the lower temperature vacuum vessel; designing it to only support vacuum loads and supporting it from the semi-permanent ‘C-shaped’ shield structure, which is supported from the facility floor.

C. Compact Stellarator (CS)

Disruption-free operation with low recirculating power is the mantra for the stellarator but simplification of the stellarator coil geometry is needed to realize this potential. A stellarator optimization study was undertaken [8] with the object of simplifying the stellarator configuration by balancing the needs of both physics and engineering. Using the ARIES-CS design as the starting point [9], two avenues were investigated for potential configuration improvements: 1) use of passive high temperature superconducting (HTS) tiles to help shape the magnetic fields and 2) incorporating removable local saddle coils to reshape the field to permit straightening of the modular coils back leg and allow removal of larger in-vessel segments. In addition to concept changes, plasma studies were carried out [10][11] looking at aspect-ratio scans for improvements on magnet complexity and computational searches to identify attractive quasi-symmetric configurations. It is envisioned that a combination of options will lead to unlocking the complexity of the stellarator configuration.

A vertical maintenance approach has been developed for the revised ARIES-CS design where internal plasma components are sub divided into inboard and outboard modules to be extracted vertically through upper ports; a concept very similar to the approach developed for the vertical AT design. Outboard blanket/shield modules have overlapping interfaces to minimize neutron streaming between assembly gaps. Although the stellarator will have the higher average major radius of the three candidate pilot plant options, the potential for no disruption loads, lower heating requirements, no current drive and reduced plasma control features can help to offset the larger physical size. A new stellarator systems code has been developed and will be used to define an operating point consistent with component build and space requirements needed extract in-vessel components. Increased aspect ratio and the inclusion of trim coils also will be evaluated.

IV. SUMMARY

The pilot plant study was initiated to see if the commercialization of fusion could follow a development path similar to fission. The study has taken a broad view of current experimental activities with the attempt to view each magnetic configuration option on a common ground. The pilot plant study is in the midpoint of its second phase with component design ideologies and critical component details being defined to establish well represented configuration designs for each pilot plant option. Open physics and diagnostic issues will be documented. Questions related to developing details of the major components, involving design basis, interfaces or requirements, also will be documented along with discrepancies found between the initial build data and the developed design details. The final phase of the study will be used to resolve open issues, reset the design point of the options if required, perform a high level availability studies and document the resuls.

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