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Results of Compact Stellarator Engineering Trade Studies

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RESULTS OF COMPACT STELLARATOR ENGINEERING TRADE STUDIES *

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Abstract-- A number of technical requirements and performance criteria can drive stellarator costs, e.g., tight tolerances, accurate coil positioning, low aspect ratio (compactness), choice of assembly strategy, metrology, and complexity of the stellarator coil geometry. With the completion of a seven-year design and construction effort of the National Compact Stellarator Experiment (NCSX) it is useful to interject the NCSX experience along with the collective experiences of the NCSX stellarator community to improving the stellarator configuration. Can improvements in maintenance be achieved by altering the stellarator magnet configuration with changes in the coil shape or with the combination of trim coils? Can a mechanical configuration be identified that incorporates a partial set of shaped fixed stellarator coils along with some removable coil set to enhance the overall machine maintenance? Are there other approaches that will simplify the concepts, improve access for maintenance, reduce overall cost and improve the reliability of a stellarator based power plant? Using ARIES-CS and NCSX as reference cases, alternative approaches have been studied and developed to show how these modifications would favorably impact the stellarator power plant and experimental projects. The current status of the alternate stellarator configurations being developed will be described and a comparison made to the recently designed and partially built NCSX device and the ARIES-CS reactor design study

1.0 Stellarator Metrics and Design Philosophy

To be successful a fusion power plant must meet physics performance requirements in an arrangement that offers attractive cost of electricity with reasonable plant availability. High availability is important for fusion power plant operations, similar to fission plants, because of its inherent high capital cost. The mechanical systems, design configuration and maintenance of a fusion power plant is considerably more complex than a fission power plant and it is not clear that time of operation would significantly improve the fusion power plants availability (to approach a benchmark 90% availability) if it were not designed for high operational availability in its initial design configuration. A design philosophy and physics / engineering metrics should be set that fosters a configuration arrangement with design features that increases the power plant availability, improves operating reliability, decreases capital costs and reduces the assembly schedule time.

Past fusion power plant studies have shown that plant availability can be maximized with a configuration design that allows the removal of large integrated plasma facing component segments that require frequent maintenance. A six year international fusion design study (1981-87) called the International Tokamak Reactor (INTOR) conducted near term power plant design studies centered on the Tokamak configuration with high operating availability as one of its design goals. The configuration design developed in this study was based on torus segmentation for horizontal access to remove large sectors between TF coils. A similar configuration arrangement was followed in the ARIES-AT design. In a further ARIES maintenance comparison study[6] the comparative advantage of the large sector removal verses through-a-port maintenance was documented. The ITER configuration follows a port based maintenance approach (depicting a “ship-in-a-bottle” arrangement). It was not designed for high operational availability, leading to inefficient remote maintenance operations and maintenance outages that range from several moths to multiple years. Even with the advent of time the ITER configuration will impede a significant improvement in its operational availability. The ARIES stellarator design sturdy, ARIES-CS, did evaluate a port based maintenance approach and deemed it acceptable, with expected plant availability in the high 80’s; however optimistic assumptions were invoked. The study assumed that aggressive maintenance research and development programs would be implemented to accomplish a robotic maintenance system that would quickly and efficiently inspect, diagnose, repair, remove, replace, and inspect all components of the power core. The assumption here is that fully automated, autonomous maintenance machines will efficiently accomplish the remote operations. Rather than relying on an untested development program, a preferred approach is to establish a design philosophy and component metrics that would avoid this path. To have the best chance of meeting a 90% operation availability in a Stellarator power plant, three principle design goals need to be followed in the configuration design:

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1. Provide straight (or nearly straight) outer modular coil legs. Incorporate trim coils or magnetic materials, e.g., S/C pucks to magnetically reform the flux surface or size the machine at a higher aspect ratio.

2. Position the modular coil back legs back far enough to allow sufficient access between the straightened modular coil back leg to extract the limited lifetime plasma facing components; removing each sector straight back.

3. The configuration arrangement should foster simplicity in the design of all components and services with the design driver being reliability and remote maintenance of a fully integrated system.

The machine capital cost and assembly schedule can be improved by reducing component shape complexities and eliminating interferences in assembled components and in the machining process of the modular coils. Design metrics include:

1 Avoid interlocking modular coil windings. This will reduce the modular coil/vacuum vessel period assembly time and costs. Incorporate trim coils, magnetic materials (e.g., S/C pucks) or higher aspect ratio to facilitate this.

2 Define smooth, non-erratic, modular coil windings. This will reduce the complexity of the winding support structure and reduce its machining costs. Coil winding times may also be improved.

3 Define modular coil winding shapes that allow adequate machine head access to help control machining cost and schedules.

4 Define simpler geometries for the blanket/FW system and vacuum vessel with adequate space for in-vessel components and clearances. This will reduce its fabrication cost and also simplify the geometric features of the plasma facing components; also reducing their capital costs.

5 Do not let the components shapes be unilaterally driven by the output of physics magnetic codes. Iterations need to be made between engineering and manufacturing to arrive at acceptable component geometries.

6 Relax magnetic field accuracy requirements wherever possible; over specification increases costs.
   a) Limit the regions where minimum assembly tolerances are needed.
   b) Limit the regions where 1.02 magnet permeability is needed; specify where it is needed and be specific where it can be degraded.

7 Consider making the modular coils "moldable" with a 2-part mold (like W7AS and W7X) when developing a stellarator power plant design. This arrangement appears not to be practical for an experimental configuration the size of NCSX due to the lack of space in the interior region.

8 Define a divertor system with full operational reliability and availability consistent with all plasma facing components.

2.0 Stellarator Congfiguration Devleopment

A number of design options need to be investigated in order to determine relevant design features that will lead to improvements in the stellarator configuration. In parallel to the engineering design process physics methodology used to define the magnetic windings needs to be reevaluated and augmented to respond to engineering constraints that will be imposed by an availability driven stellarator configuration. Different design concepts will be investigated later this FY in pursuit of an improved stellarator configuration.

One configuration arrangement that was explored at the advent of the NCSX project was looking at the potential benefit of incorporating monolithic High Temperature Superconductors (HTS) for stellarator applications [1]. The diamagnetic properties of the bulk HTS material can be used to provide simple mechanisms for field-shaping where the bulk superconducting materials provide diamagnetic control of stellarator fields through magnetic dipoles aligned against the ambient field. An initial study was undertaken to look at the far right spectrum where the bulk HTS material replaces the traditional conductor shaped modular coil winding system, although a less radical departure from the nominal winding design may be of greater interest in future design studies.

3.0 HTS Stellarator Configuration

A pure HTS puck stellarator configuration has been developed using the ARIES-CS design point and component features as a point of departure. The ARIES-CS design was developed using a port-based maintenance scheme whereby replacement of the blanket modules is done through a limited number of designated maintenance ports. Incorporating HTS pucks in the design process allows magnetic forming materials to be placed as tiles on a structural substructure, effectively allowing a traditional modular coil winding to be electrically split in the poloidal direction at any location. With the ability to electrically disjoin a traditional winding provides one way for a stellarator to be designed with large openings that provide access to remove interior plasma facing components, once restricted by highly shaped back legs of the modular coil winding.

The general arrangement of the HTS puck stellarator concept is shown in Figure 1 illustrating the major details of the core components.

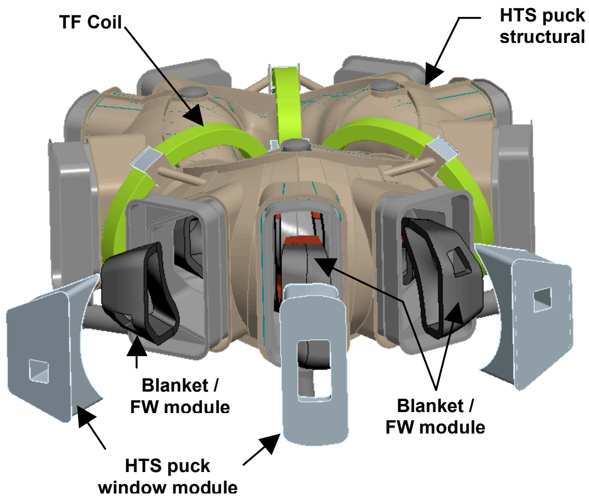

Figure 1. Isometric View of the HTS puck stellarator concept

50° K, Monolithic High Temperature Superconductors (HTS) are used to provide field-shaping against an ambient field provided by three superconducting TF coils. The three TF coils are arranged and sized to allow each of the three field periods to be assembled with a straight line radial motion. The HTS pucks are pre-attached to the puck shell structure. Figures 2 and 3 shows the local details of the vacuum vessel, the HTS puck structural shells and the modularization used in their assembly. The inboard HTS puck structural shell is made up from three pairs of sub modules assembled around the TF inboard leg and its bucking system support structure. Although details have not been developed, given the size of the overall assembly, there is adequate space for bolt access along the entire length of an exterior flange of the pucks structural shell. The installation of three vacuum vessel period segments is shown in Figure 3. A recessed area was provided in the inboard puck shell structure to allow room for the vacuum vessel pump ducts located at the bottom of the vessel, just inboard of the central horizontal port. A pair of outboard puck shell structures is assembled with radial motion centered on each of the two large angled ports and bolted together at their interfacing flanges.

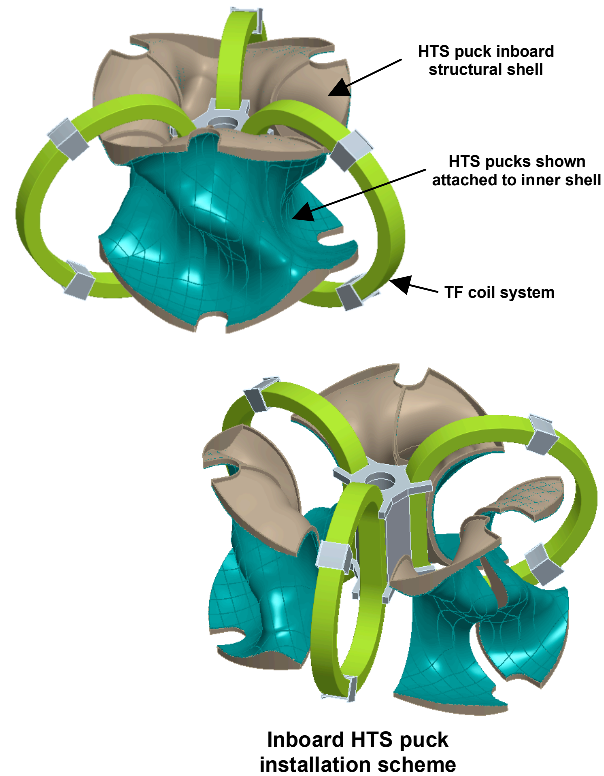

Figure 2. Isometric View of inboard HTS puck assembly scheme

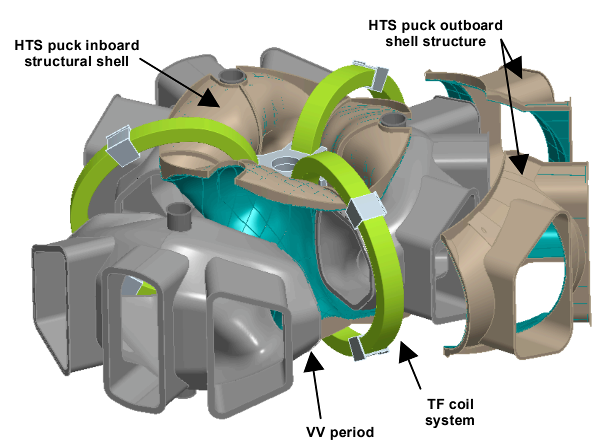

Figure 3. Vacuum vessel and outboard HTS puck shell assembly scheme

The geometry of the blanket / FW system and vacuum vessel was refined to provide the greatest amount of shape simplification in an effort to minimize their fabrication costs. Where magnetic fields and plasma shaping is most critical on the inboard surface a simplified conformal shell structure was generated. Along the top and outboard regions the geometry was shaped using straight lines and circular arcs. Figure 4 illustrates the developed geometry of the blanket / FW period.

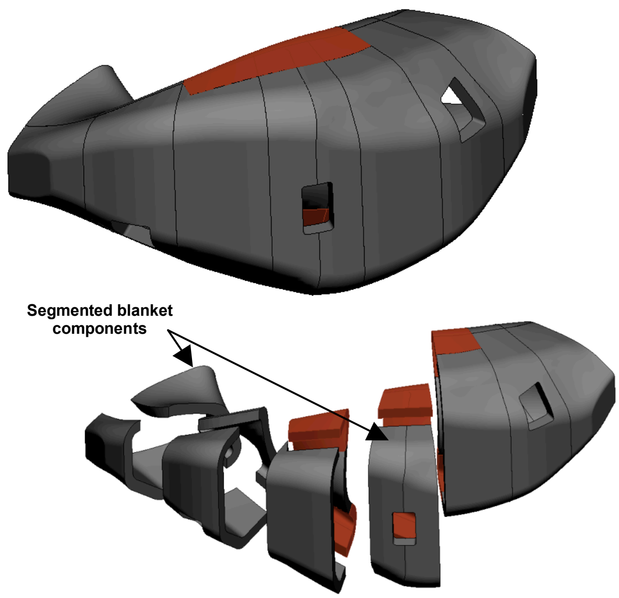

Figure 4 Blanket / FW components

The blanket/FW segmentation adopted in this study allowed every other sub module to be extracted straight back through the horizontal ports. This was predicated by maintaining the superimposed modular coil magnet position used in the ARIES-CS design. In a follow up design study this requirement will be changed to allow the magnet surface (or modular coils) to be positioned back far enough to allow each blanket segment to be extracted in a radial motion through individual vacuum vessel ports. HTS puck window modules (shown in Fig. 1) are inserted into the large horizontal ports to provide additional shaping fields. The puck window modules would operate at 50° K and have a cold-to-warm interface to carry the magnetic loads to the room temperature vacuum vessel. A double-walled vacuum vessel provides the vacuum boundary for the plasma components on the inside and the superconducting TF and PF system on the outside. A cross-sectional view of the HTS puck configuration is shown in Figure 5.

4.0 Concluding Remarks

A pure HTS puck stellarator configuration was developed using the ARIES-CS design point and component features as a point of departure and provided further insight into options that might be used to simplify the stellartor configuration.

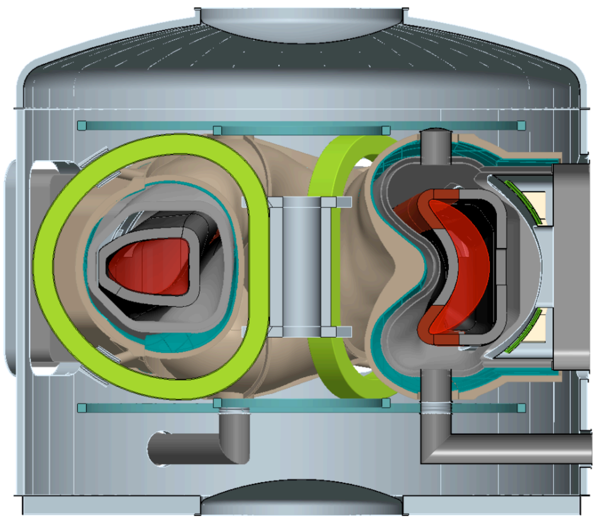

Figure 5 Section View through the HTS puck configuration

Follow-up studies will be carried out this FY to look at a configuration that incorporates modified modular coil winding shapes in conjunction with trim coils and magnetic materials to improve the maintenance characteristics of the stellarator.

References

[1.] L. Bromberg,. et al., "Options for Magnetic Field Shaping in Stellarators Using High Temperature Superconducting Monoliths", 23rd Symposium on Fusion Engineering, San Diego, Calif., 2009.

[2.] F. Najmabadi, et al. and the ARIES Team, "Exploration of Compact Stellarator as Power Plants: Initial results from ARIES-CS Study,".

[3.] X.R. Wang, S. Malang, A.R. Raffray and the ARIES Team, "Initial Maintenance Assessment for ARIES-CS Power Plant," 20th IEEE/NPSS Symposium on Fusion Engineering, San Diego, California, Oct. 2003

[4.] L. Waganer and J. Peipert-Jr, "ARIES-CS Maintenance System Definition and Analysis", ARIES-CS Maintenance System Final Report.

[5.] M.S. Tillack, S. Malang, L. Waganer, X. R. Wang, D.K. Sze et al., and the ARIES Team, "Configuration and Engineering Design of the ARIES-RS Tokamak Power Plant", *Fusion Eng. & Des.*,38(1997) 87-113.

[6.] L. Waganer, "ARIES-At Maintenance System Definition and Analysis", ARIES-AT Maintenance System Final Report.

[7.] International Tokamak Reactor (INTOR), Phase One and Two Final Reports, International Tokamak Reactor Workshop organized by the International Atomic Energy Agency, Vienna, from 1985 to 1987.

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