US Collaboration on the W7-X Stellarator

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ABSTRACT

The new American-German collaboration on the W7-X stellarator in Greifswald, Germany, is in its first year as an ICC project. Los Alamos, Princeton, and Oak Ridge have organized an effort centered on applications of 3D magnetic fields to improve the performance and design of toroidal confinement devices. In particular, presently we have three focus areas: providing copper trim coils to be mounted externally on the W7-X cryostat, developing a scraper element for the divertor, and working control issues under the theme of 3D Diverted Plasmas. The trim coils must be designed and built on a fast schedule, to meet assembly timelines. The LANL effort involves theoretical modeling of whether the stellarator bootstrap current depends on the radial electric field, and an experimental investigation (IR imaging) of the heat loading and wall interactions in the W7-X 3-dimensional divertor. W7-X is a modular niobium-titanium superconducting stellarator of the helias type, with 5-fold symmetry, and the outer vessel is 16 meters in diameter, not including port extensions. It is presently 4/5 assembled, with the last module being moved into place this year. There are more than 250 ports (being insulated and welded into place now), although only ~150 are available for diagnostics...i.e., 30 per sector. Completion of construction is scheduled for late 2014. Other opportunities for increased collaboration scope abound.

We report on the establishment of a new collaboration which began in FY11, between the US and German fusion research programs, under the auspices of the recently recompeted OFES Innovative Confinement Concepts program, formally known as **SC Program Announcement LAB-10-286 “Research in Innovative Approaches to Fusion Energy Sciences”**. This US/German collaboration is entitled “Control of 3D Diverted Plasmas: Partnership with W7-X”, and is the outcome of a set of discussions and negotiations amongst partners at the Princeton Plasma Physics Laboratory (PPPL), Oak Ridge National Laboratory (ORNL), Los Alamos National Laboratory (LANL), and the Max Planck Institute for Plasma Physics at Greifswald (IPP-Greifswald), as well as follow-on discussions with OFES to resize it within available resources.

The W7-X stellarator is a large (R = 5.5 m, B ~ 3 T) superconducting device (see Fig. 1) based on an optimized quasi-omnigenous configuration[1]. The second adiabatic invariant, J*(ψ) isocontours for trapped particles are aligned with flux surfaces, to minimize neoclassical transport.

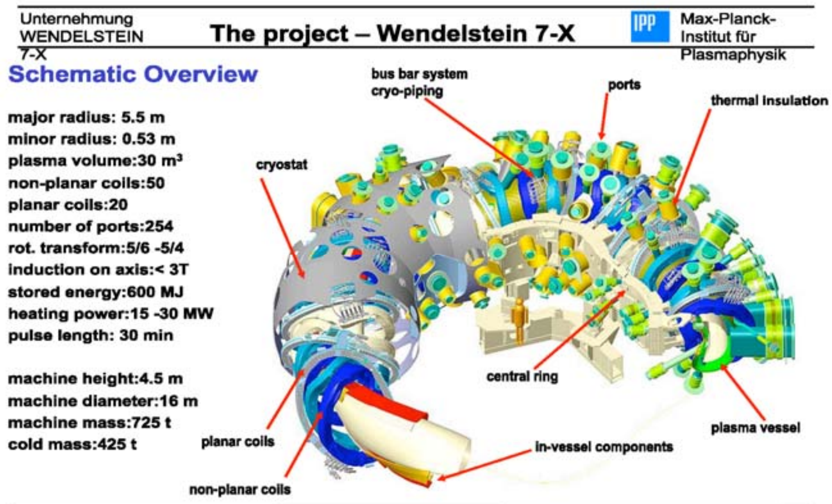


Figure 1: A schematic of the superconducting W7-X stellarator, presently under construction in Greifswald, Germany, and a table of design parameters.

The first three years of operation will aim to optimize operating scenarios for realization of the W7-X goal of steady-state operation of high-density (n ~10^{20} m^{-3}), high temperature (multi-keV) high-beta (~5%), stellarator plasmas. Thereafter, the inertial divertor structures will be replaced by a water-cooled system which will enable W7-X to increase its pulse length to its design value of 30 minutes at peak steady-state heat fluxes of 10 MW/m^2. Since the W7-X magnetic configuration has strongly reduced equilibrium currents, the finite-beta flux surfaces shift very little with respect to the vacuum configuration, and the MHD equilibrium should rapidly approach that of a true steady state plasma. Large driven plasma currents are neither required nor expected in W7-X, though modest ECH current drive will be used to tune the magnetic configuration. This obviates the need to generate and manage large populations of supra-thermal electrons, and so avoids the most frequent cause of failure of plasma-facing components on Tore Supra, the world's most developed superconducting tokamak with actively cooled in-vessel components. The maximum plasma density that can be attained in stellarators is determined not by instabilities, as in tokamaks, but by the balance between input power and radiation losses. Multi-device studies[2] have established density-limit scalings which predict that W7-X easily achieve reactor- like densities > 10^{20} m^{-3}.

The W7-X facility is the product of major investments in fusion research by the German government and EURATOM, and of decades of IPP experience in stellarator physics and engineering. A research team of ~200 scientists is envisioned for W7-X, half of whom will come from research institutions other than IPP. So far, IPP has concentrated its own efforts on the enormous job of constructing the W7-X device and its largest subsystems (e.g., heating). Work is less advanced in several other areas that are critical to timely achievement performance goals of W7-X:

- Effective particle exhaust in a 3D divertor subject to reactor-level heat fluxes.
- Integrated control of plasma and divertor performance in steady-state operation.
- Efficient analysis of the 3D, finite-beta W7-X magnetic configuration.
- Steady-state pellet fueling to maintain high densities without excessive gas build-up (now an out-year effort).

Elements of the new collaboration presently include:

1. Pressure-Induced Stochastization of Equilibrium Flux Surfaces

Developing a predictive understanding of beta-limiting mechanisms is a critical scientific issue for stellarators. Research on the W7-AS stellarator has indicated that, in at least some regimes, the beta limit is likely caused by the stochastization of an increasingly large volume of field lines with increasing beta. The strong influence of the divertor control coils on the width of the stochastic region[3], and the expectation that the edge stochastic region will in turn affect the divertor performance, imply that there will be a close coupling between research in this area and that on divertors. We will remove the stellarator symmetry constraint from PPPL's PIES code, and will begin work on developing an equilibrium reconstruction code for W7-X, based on the PIES and SIESTA codes.

2. Trim Coil System

The trim coil system (coils and power supplies) fills a mission- and time-critical need in the W7-X project. PPPL is leading this task, with strong involvement by ORNL. IPP has supplied coil and power supply technical specifications, magnetic load cases, space envelopes, and locations of all leads and coolant interfaces. The U.S. team has developed detailed designs and manufacturing drawings for the coils, performed structural and thermal hydraulic analyses, completed preliminary and final design reviews, and initiated procurement. The first two coils will be delivered to IPP in June, 2012. In conjunction with the control coil design activities, the U.S.-IPP team has updated the power supply technical specification. The U.S. team has begun design work on the power supplies, with the goal of manufacturing and delivering them to IPP by April, 2014, so as to be available to support W7-X commissioning activities. The budget for the coil project is \$2.9M, and the preliminary estimate for the power supplies is \$2.7M.

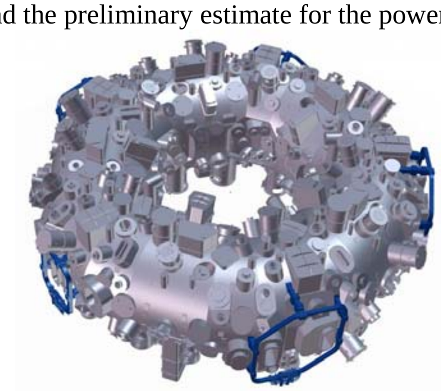
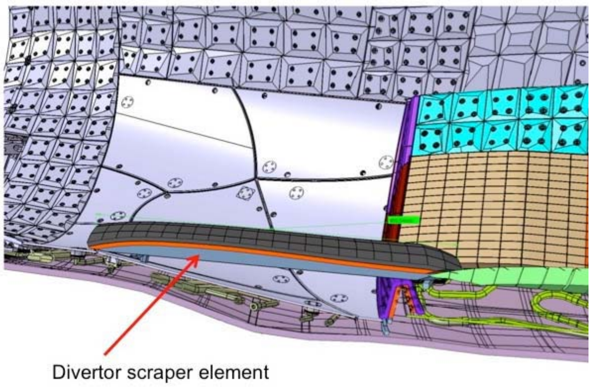


Figure 2: PPPL/ORNL are designing and building five normal-conducting trim coils , which will allow individual adjustment of the divertor heat patterns in the five modules of W7-X. Shown is a CAD drawing of the W7-X outer vacuum vessel, with attached trim coils (in blue).

3. Divertor Science and Technology

The W7-X configuration includes a magnetic island divertor with a 3D structure, which leads to local heat flux concentrations on divertor elements greater than 10 MW/m2, comparable to those expected in a magnetic fusion reactor. The structures must be shaped and actively cooled in these conditions, and the required technology solutions are of broad importance for fusion energy science, not only W7-X. We will perform the design of a new internal structure, the "scraper element," which will expand the high-heat-flux (HHF) coverage area of the W7-X first wall and is expected to use ITER-relevant monoblock technology. This recent addition to the W7-X design will increase the safe steady-state operating space of the machine and allow greater flexibility for control optimization.



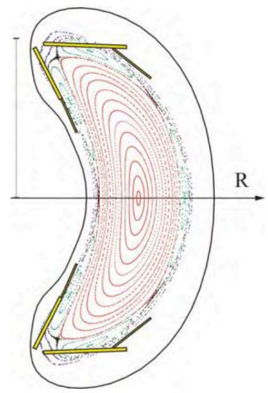


Figure 3: Divertor scraper elements will protect the throat (narrowest part) of the divertor against heat transients during evolution of the plasma beta. The magnetic configuration near the island divertor plates is shown on the right.

4. Plasma Control

W7-X is the first optimized stellarator that will operate at high power for long pulse. As a result of long pulse operation, it will also be the first optimized stellarator that will need to manage the resultant heat and power fluxes in true steady state. The design of W7-X incorporates an island divertor that will distribute the heat on water cooled carbon divertor tiles. In some scenarios, the bootstrap current may affect the interaction between the divertor tiles and the heat and particle fluxes, causing undesirable concentrations of the heat leading to a potential over-temperature of the tiles. The predictive models that will be used to develop control scenarios and design controllers[4] require calculations of the bootstrap current (LANL theory effort) and divertor response. The proposed control activities consist of 1) developing simplified models of the physics which determines the transient behavior of the plasma including; the bootstrap current, the response of the plasma to the control actuators, and the effect of the plasma motion on heat flux distribution on the divertor, and 2) developing controllers to maintain the heat flux within desired parameters. To measure the heat fluxes, full IR coverage of the divertor is planned. LANL expertise in visible and IR imaging systems[5, 6] will be aimed at imaging the interior of W7-X in multiple wavelength bands (visible and IR simultaneously). We will utilize the U.S. IR system on Alcator C-Mod as an initial test platform, and provide high resolution IR “science” cameras to the W7-X endoscopes. Real-time image analysis algorithms to quickly & reliably respond to the heating will be developed for integration into the W7-X plasma control system.

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