The Steady State Tokamak

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Over the decade of the 90's the prospects for a tokamak based fusion power plant improved dramatically, based largely on the discovery and development of a set of techniques collectively known as advanced tokamak operation. These techniques improve the physics performance of tokamaks while removing the constraint of pulsed operation inherent when some of the plasma current is inductively driven by a transformer. These techniques avoid any inductive current, optimize the bootstrap fraction, require control over plasma profiles, current and pressure most particularly, and develop transport barriers for improved confinement. Wall stabilization may be required for the full realization of this mode of operation. In what follows, steady state operation is assumed to be synergistic with this advanced tokamak construct. All other steady state concepts are uninteresting in the power plant context.

Power plant conceptual designs have incorporated these features to illustrate the benefits and determine the engineering and economic impacts on the design. In Fig. 1 is seen the impact of power plant designs from the mid-80's (Pulsar) through the early 90's (ARIES-I) to the present (ARIES-RS, the advanced tokamak "reversed shear" design). Both the fusion core volume and cost of electricity have been reduced progressively so that in large sizes the advanced tokamak plants have projected costs below 6¢ per kilowatt-hr. The Pulsar study assumed conventional physics performance, with high duty factor pulsed operation. The ARIES-I operates in the first stability regime but its cost reduction is based on very aggressive technology (e.g., peak conductor magnetic field of 21 T), while the ARIES-RS reductions are based on advanced physics and some advances in magnet technology.

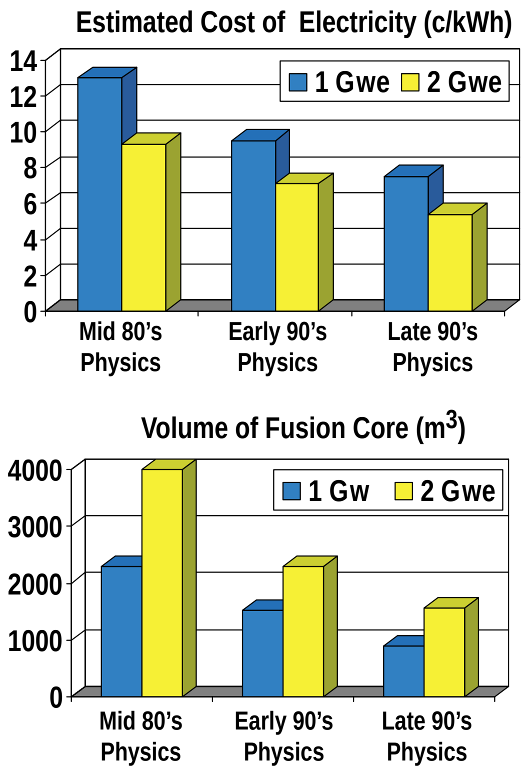

Fig. 1. Our vision of magnetic fusion power systems has improved dramatically in the last decade, and is directly tied to advances in fusion science and technology.

These advanced physics techniques need refinement in order to demonstrate sustainment at the parameter levels assumed in advanced designs, but already there has been remarkable progress at shorter pulse lengths in tokamaks throughout the world – reported in major fusion conferences throughout this decade. A partial summary of critical parameters achieved simultaneously is shown in Fig. 2, from several different DIII-D modes and from JET, compared with the values assumed in several different power plant studies. These parameters include the both the toroidal and normalized beta (β and β_N), confinement enhancement H, bootstrap fraction f_{bs}, density relative to the Greenwald density, and the electron/ion temperature ratio.

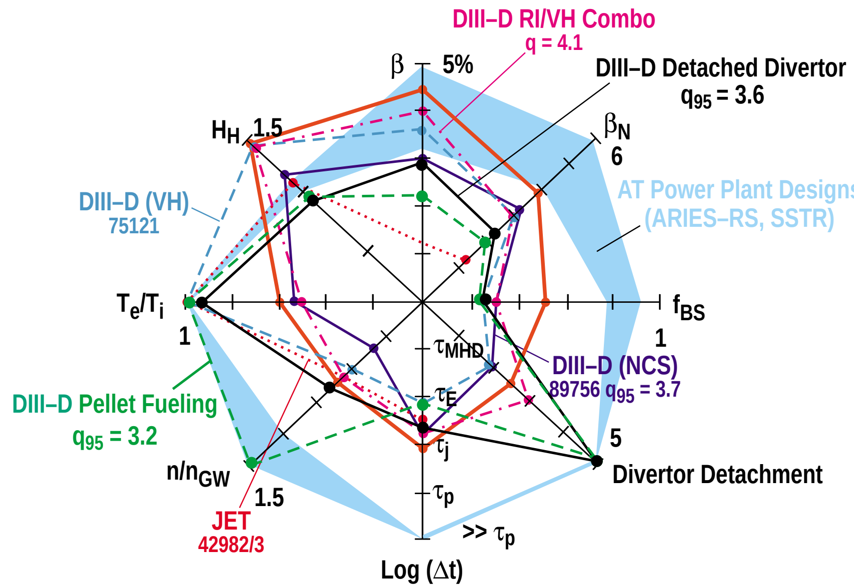

Fig. 2. Tokamak performance envelope.

While there are successes in confinement and normalized beta in these modes, they have been achieved at lower density (normalized to Greenwald), lower electron/ion temperature ratio, and shorter duration's than are required. Perhaps more critically, the performance is achieved in short pulses, and as one attempts to sustain these advanced modes for longer times (relative to the energy confinement time) the performance degrades. This can be seen in Fig. 3 showing DIII-D data for the product of β_NH as the pulse length increases. Perhaps the single most important area requiring advancement is the MHD stability of these modes in their final (steady, controlled) profiles. The challenges in reaching the final negative central shear (NCS) profiles are illustrated in Fig. 4.

For operation at high bootstrap fraction (proportional to β_p) and high β, the stability level as characterized by β_N must be elevated to ~5–6, above conventional Troyon limits. [The product $\beta\beta_p$ is proportional to β_N^2 through a shaping factor.] At fixed field, plasma

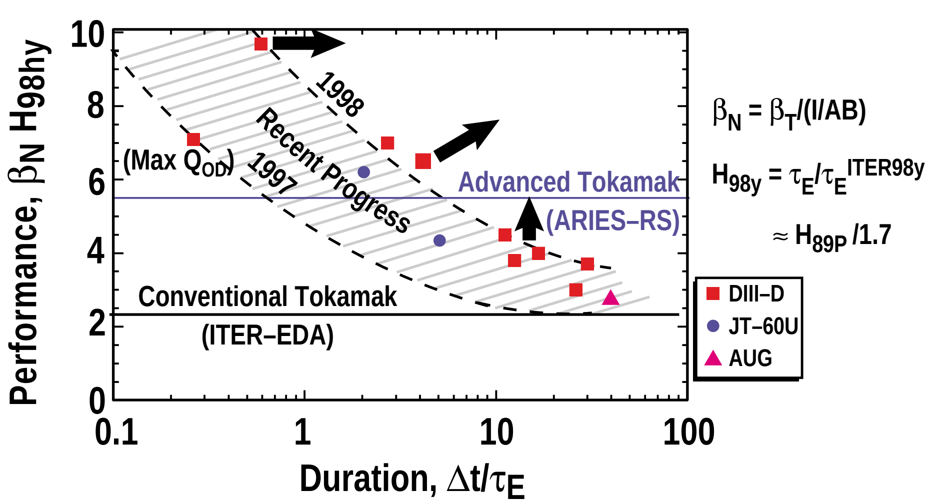

Fig. 3. The DIII–D 1998 research has advanced and extended tokamak performance and duration.

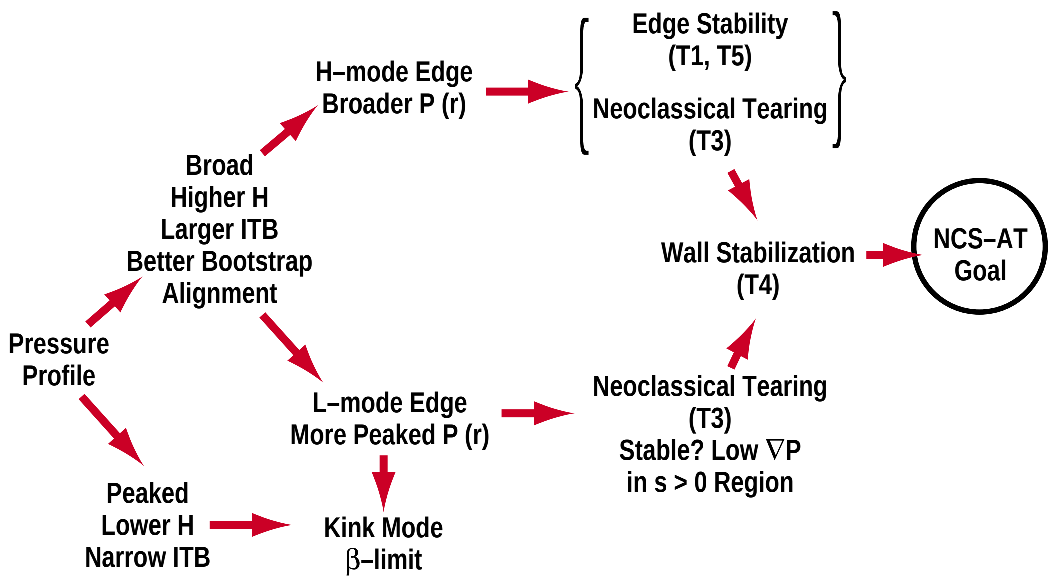

Fig. 4. The path to NCS-Advanced Tokamak goal leads through many stability issues.

shape, and size one has $I_p^2 \beta_p \sim \beta$, so one can think of advanced mode operation as raising β_p [by operating at lower than normal plasma current I_p for the particular machine], then enhancing β by attaining high β_N through wall stabilization and profile shaping. To stabilize modes with a conducting shell and feedback control will introduce further complexity into the device, as illustrated in Fig. 5 which shows the design of the shell for the Tokamak Physics Experiment (TPX). This particular design allowed stable operation up to values of $\beta_N = 4.8$.

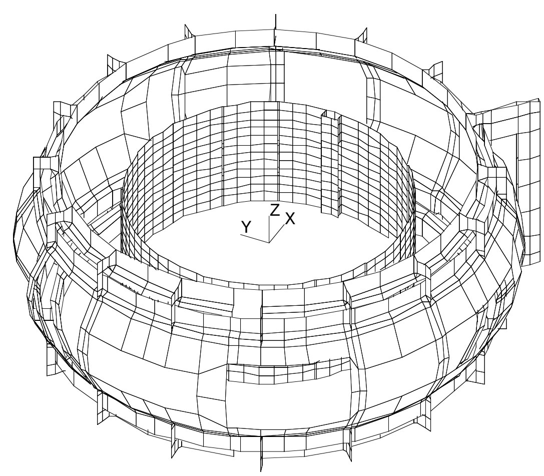

Fig. 5. Passive stabilizer "cage" for MHD control ⇒ TPX.

The MHD operating space where one can tradeoff β for β_p at a given stability level (β_N) is illustrated in Fig. 6, where S is a shaping factor and ε is the inverse aspect ratio. Several recent power plant design operating points are superimposed on this space. It is important to note the range of beta values assumed in these plant designs, from a low of 1.9% for ARIES-I to a high of 5% for ARIES-RS. The SSTR design from Japan is also at low beta. But since there must be similar values of βB^2 (proportional to fusion power density) to make economic sense, the magnetic fields are significantly different. The on-axis fields for ARIES-RS, SSTR, and ARIES-I, respectively are at 8, 9, and 11 T, but more importantly have peak conductor fields of 16, 16.5, and 21 T. The EDA design of ITER, for comparison, has axial and peak fields of 5.7 and 12.5 T using Nb_3Sn conductor, and this is a state-of-the-art design. A practical limit for this conductor is about 13 T, using conventional cooling and maintaining appropriate temperature margins. Higher field values than those of ITER will require either advanced conductors or cooling methods not available today, so high β is clearly essential. Further, to ensure sufficient β_p for very high bootstrap fraction then requires high values of β_N.

The current, very active, research program into advanced modes is perhaps the most important topic for tokamak research today, and the outcome will decide whether the tokamak is to operate continuously and in an interesting parameter range for power plants, or must operate in a pulsed mode. It is a pivotal issue for the tokamak program, and leads to very different physics operating directions and supporting technology. The pulsed mode would bear some resemblance to the ITER EDA mode, and would be a large radius machine. Its physics would be much closer to that already demonstrated in the tokamak program, and result in less complex core systems. For the advanced tokamak modes, one requires strong shaping, internal conducting structures and feedback control for enhanced stability, and multiple current drive systems for profile control.

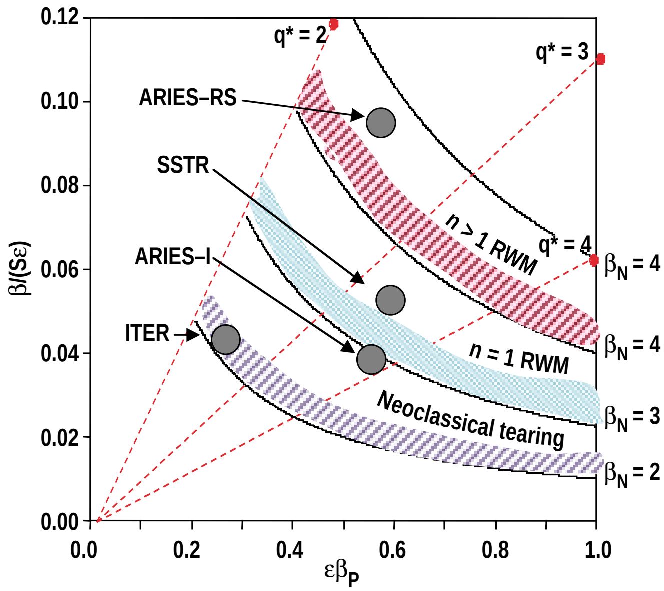

Fig. 6. MHD operating space for tokamaks.

Given this background, what are the opportunities for steady state tokamak research today? First, existing tokamaks can continue their efforts to sustain high performance modes in the correct parameter regimes for times as long as their magnet and current drive systems can operate. Both JET and JT-60U have extensive current drive systems at high power and have attempted to do this. Unfortunately, the combination of sufficiently enhanced confinement at high values of β_N have not been demonstrated in long pulse. For example, the best value of β_N in JET at long pulse with good confinement is less than 3, where values of 5–6 are desired. Similar results are found in JT-60U. One might argue that existing machines do not yet have the necessary control tools, and there is some merit to that. So the program over the next 5 years on these devices should be to establish the maximum levels of performance that can be sustained in long pulses.

The TPX program for steady state advanced tokamaks aimed at extending the necessary performance levels to time scales that wall processes were in equilibrium with the plasma, and that is a next step. In Korea, the superconducting KSTAR device has many of the features of the TPX and should be able to reach that goal in its later phases. A view of KSTAR is shown in Fig. 7, with the schedule for its construction and operation shown in Fig. 8.

A last step in the steady state advanced tokamak program is to show that these performance levels, for very long times, are not degraded when the plasma is heated largely by alpha power. In fact, it is likely that the introduction of that powerful heating system will alter the construct for steady state high performance operation developed using deuterium fuel. In fact, it could open new opportunities. This step however, should probably be combined with the study of burning plasma physics.

To summarize, the viability of the tokamak could well rest on the success of advanced tokamak scenarios — those that create steady state high performance core plasmas. The ARIES-RS study is a good example of how these regimes can improve the projected cost of electricity for fusion power plants. But despite 10 years of extensive work since these modes were first discovered, we have only transiently met key physics metrics for success — and we have no assurance that the required control is compatible with a dominated power input. And, innovative technology is also a key part of lowering projected COE even if these modes provide the needed high performance. So the challenges for this next decade are clear, and the goal of established advanced tokamak performance limits should be the highest priority for the tokamak program.

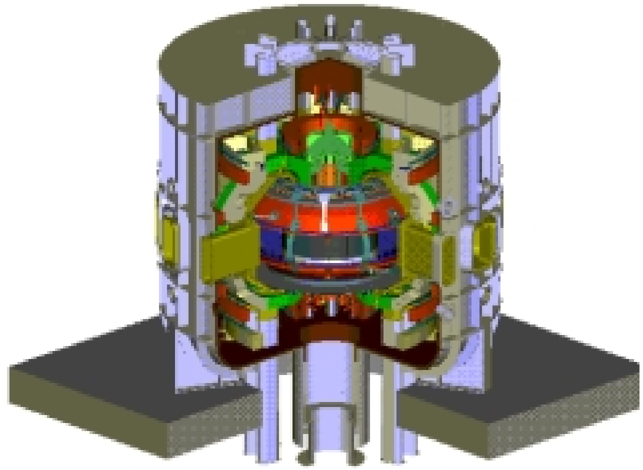

Fig. 7. A view of KSTAR.

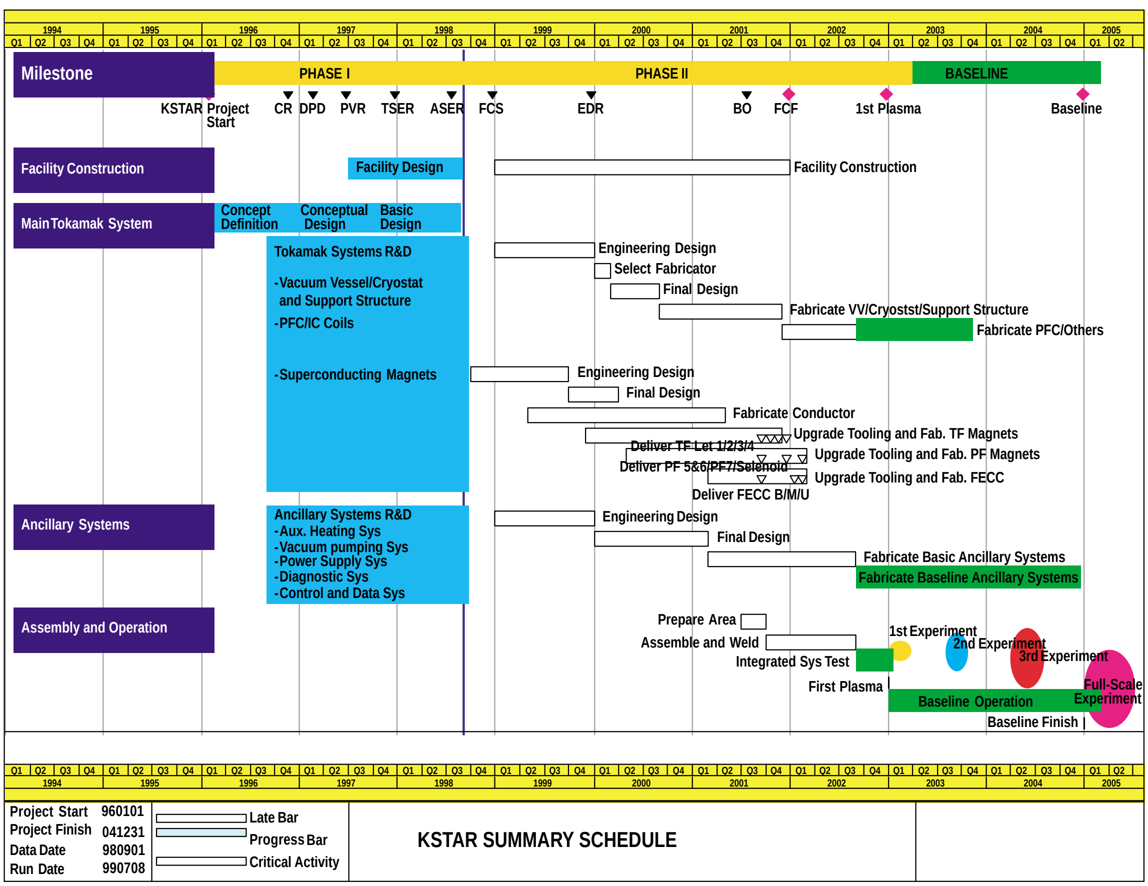

Fig. 8. KSTAR summary schedule.