# SUMMARY

## ITER EDA and Technology

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## 1.0 INTRODUCTION

The year 1998 was the culmination of the six-year Engineering Design Activities (EDA) of the International Thermonuclear Experimental Reactor (ITER) Project. The EDA results in design and validating technology R&D, plus the associated effort in voluntary physics research, is a significant achievement and major milestone in the history of magnetic fusion energy development. Consequently, the ITER EDA was a major theme at this Conference, contributing almost 40 papers.

Another major theme at the Conference was the significant progress towards the ability to demonstrate steady-state operation with all the associated implications on technology. Of course, ITER has made major contributions to this basic theme as well.

A key issue for the development of fusion energy is how to provide sources of high energy neutrons for materials and component testing, as well as a test of the physics required for producing substantial amounts of fusion power at high-energy gain. Several options were presented at the Conference, recognizing again that this is also to some degree part of the ITER EDA Mission. Looking in the longer term, papers were presented on various embodiments of power plants with several innovative design concepts. Continued progress in fusion nuclear technology and materials was also reported in several papers.

With these themes in mind, this summary is presented with the following sections:

- ITER EDA Design and Technology R&D
- Progress Towards Steady-State
- Fusion Energy and Neutron Sources
- Nuclear Technology and Materials
- Advanced Design Concepts

In total, about 90 ITER EDA and technology papers were presented, accounting for about 25% of the papers at the Conference.

## 2.0 ITER EDA DESIGN AND TECHNOLOGY R&D

### 2.1 ITER OVERVIEW

The ITER Project has been conducted under the auspices of the IAEA according to the terms of a four-party agreement among the European Atomic Energy Community (EU), the Government of Japan (JA), the Government of the Russian Federation (RF), and the Government of the United States (US), referred to herein as the Parties. “The overall programmatic objective of ITER is to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes. ITER would accomplish this by demonstrating controlled ignition and extended burn of deuterium-tritium plasmas, with steady state as an ultimate goal, by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of the high-heat flux and nuclear components required to utilize fusion energy for practical purposes.”

Fusion energy programs throughout the world have benefited from a remarkable degree of openness and global cooperation which has brought with it dramatic progress in scientific understanding and performance achievement. The ITER project arose from the recognition by the leading programs of the comparable positions reached in existing experiments and of the benefit to be derived from undertaking the next step jointly. Collaboration on ITER provides significant savings through sharing of costs, and more importantly, the opportunity to pool experience and expertise gained over recent decades, and to draw from the scientific and technological expertise of all the world's leading fusion experiments and programs in an integrated and focused venture.

The original detailed technical objectives to achieve the overall programmatic objective of ITER were adopted by the Parties in 1992. ITER will have two roughly ten-year phases of operation, the Basic Performance Phase and an Enhanced Performance Phase. The first phase will address the issues of controlled ignition, extended burn, steady-state operation, and the testing of blanket modules. ITER's technical objectives require demonstration of controlled ignition and extended burn, in inductive pulses with a flat-top duration of approximately 1000 s and an average neutron wall loading of about 1 $MW/m^2$. ITER should also aim to demonstrate steady-state operation using non-inductive current drive in reactor relevant conditions. It is assumed that for the first phase there will be an adequate supply of tritium from external sources. The second phase would emphasize improving overall performance and carrying out a higher fluence component and materials testing program. Tritium breeding might be implemented for this phase. ITER must also be designed to demonstrate the safety and environmental acceptability of fusion as an energy source.

The original Engineering Design Activities (EDA) of ITER were completed by the Parties in July 1998 after 6 years' activities. During this period, the Parties agreed to produce a detailed, complete and fully integrated engineering design of ITER and all technical data necessary for decisions on the construction of ITER. The results of the EDA are available to the Parties to use either through international collaboration or within their domestic programs. The deliverables given at the end of the EDA met the original plan. The ITER design, supported by technology R&D, is at an advanced stage of maturity and contains sufficient technical information for the construction decision.

The ITER project has so far proved to be an unprecedented and successful model of international cooperation in science and technology in which all participants benefit not only from the technical results but also from the experience of different approaches to project organization and management. It has proved to be an effective and efficient vehicle for the fusion engineering needed to realize any concepts of commercial magnetic fusion reactors. Bringing ITER to full realization through joint construction and operation will continue this process.

## 2.2 ITER DESIGN

The cross-section of the ITER Tokamak is shown in Fig. 1 and the main parameters are summarized in Table I. The design was defined after careful study of the balance between physics requirements for plasma confinement, control and stability based on ITER Physics Basis and Physics Rules, and engineering constraints such as heat loads, electromagnetic and mechanical characteristics, neutron shielding and maintainability to ensure safe and reliable operation within reasonable cost.

TABLE I: MAIN PARAMETERS AND DIMENSIONS OF ITER

| | |
|---|---|
| Total fusion power | 1.5 GW |
| Neutron wall loading | 1 $MW/m^2$ |
| Plasma inductive burn time | $^{3}$ 1000 s. |
| Plasma major radius | 8.1 m |
| Plasma minor radius | 2.8 m |
| Plasma current (Ip) | 21 MA |
| Toroidal field @ 8.1 m radius | 5.7 T |
| Maximum toroidal field at coil | 12.5 T |
| Auxiliary heating power | 100 MW |

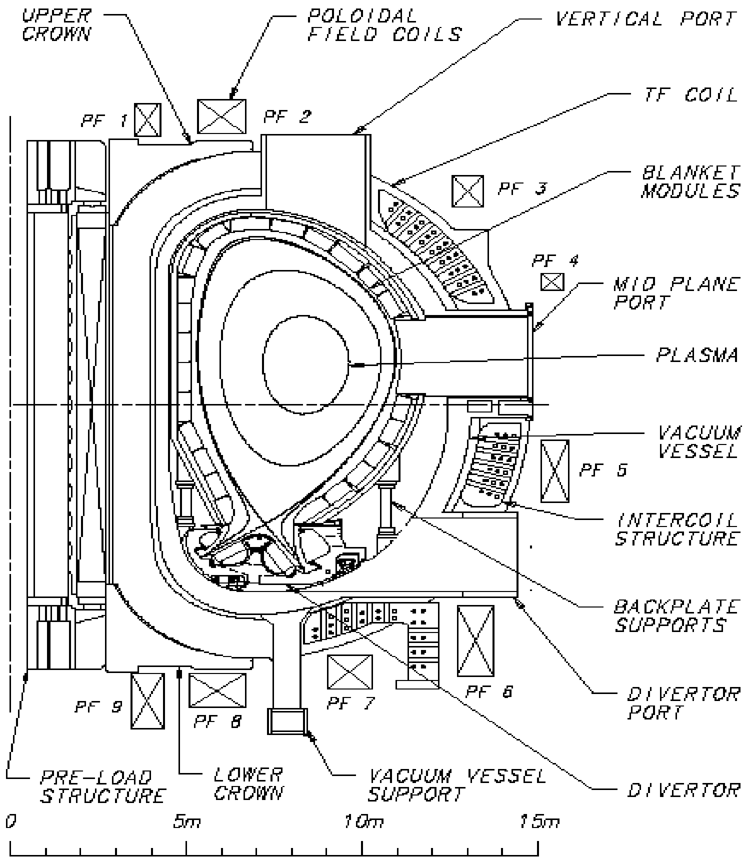
Plasma performance of ITER is assessed based on the most recent experimental results and modeling. The three issues that most directly determine the plasma performance are:

- Energy confinement, edge parameters and capacity to reach and sustain H mode;
- ß (ratio of plasma pressure to magnetic field pressure) and particle density; and
- Impurity dilution, radiation losses, helium exhaust and divertor power handling.

Each of these issues has been studied thoroughly in a collaborative framework of voluntary ITER physics activities, coordinated through Expert Groups, which draws on the full range of physics expertise throughout the Parties' Fusion programs

ITER performance is summarized in Fig. 2 (a) and (b) which plot fusion power for a 21 MA discharge as a function of the H-mode enhancement factor, $H_H$, which characterizes the global energy confinement time in relation to its reference extrapolated value based on ELMy H-mode confinement. The plots take into account critical parameters concerning power loss ($P_{loss}$) **across the separatrix normalized by** L-H power thresholds ($P_{LH}$), particle density (n) normalized by Greenwald density (**$n_{GW}$)** and normalized beta ($\beta_N$) and indicate the domain where the three conditions, $P_{loss}/P_{LH} > 1$, $n/n_{GW} < 1.5$, $\beta_N < 2.5$ are satisfied either in ignited condition (Fig. 2 (a)) or in driven mode with heating power Paux = 100 MW (Fig. 2 (b)). In the case of ignition the available range of operational parameters around their normal values is commensurate with the possible uncertainties in extrapolation of confinement time. In driven modes, the feasible region is extended to cover a larger range of uncertainties.

**Fig. 1 Cross-section of the ITER tokamak**



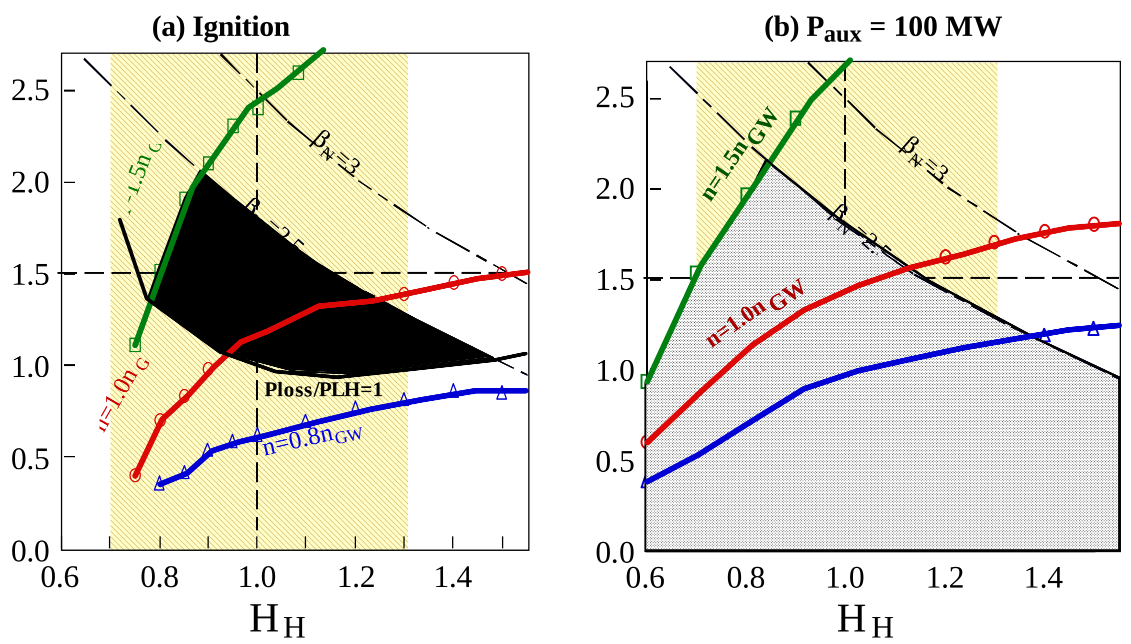



**FIG. 2 Fusion Power Domains @ 21 MA**
**($P_{loss}/P_{LH}$ > 1, $n/n_{GW}$ < 1.5, $\beta_N$ < 2.5)**

The design incorporates all of the provisions needed for the reliable operation and control of ignited/or high Q driven-burn DT plasmas with fusion powers in the 1-1.5 GW range and fusion burn durations ≥ 1000 s. The nominal plasma parameters are chosen such that with “reference” physics basis assumptions about attainable energy confinement, attainable plasma density, adequate divertor target heat load, and projected plasma impurity content, sustained D-T burn with power ≥ 1 GW is possible. Auxiliary heating and/or current drive powers of up to 100 MW are provided for the initiation of ignited burn and for the sustainment of high-Q (≥ 10) driven burn. The in-vessel plasma-facing surfaces and nuclear shielding modules are designed for steady-state power handling capabilities. The Poloidal Field coil system is sized and configured such that static and dynamic plasma equilibrium control at plasma currents of up to 24 MA is possible, and supplies sufficient inductive current drive to enable nominal 21-MA, 1600-s duration pulses (including a 1000-s fusion burn) to be produced. Somewhat shorter duration (500-s burn) inductively-sustained pulses at 24 MA are possible. Extension of the controlled burn duration up to ~ 6000 s in a reduced-current driven-burn mode is also feasible. A true steady-state plasma operation with current driven by non-inductive methods at 1 GW range of fusion power in a reverse shear configuration can also be accessible.

The essential engineering features include:

- An integrated structural arrangement in which super conducting magnet coils (20 cased toroidal field coils, 9 poloidal field coils and a monolithic central solenoid) and vacuum vessel are linked to provide an overall assembly which simplifies the equilibration of electromagnetic loads in all conditions, relying largely on the robustness of strong TF coil cases (Fig. 2); and

- Modular in-vessel components (blanket modules on back-plate and divertor cassettes shown in Fig. 3) designed to be readily and safely maintainable by a practical combination of remote handling and hands-on techniques.

- The tokamak is contained in a cryostat vessel, situated in an underground pit, inside a building of about 50 m height. Peripheral equipment such as fueling and pumping, heat transfer, auxiliary heating and remote handling are arranged in galleries around the main pit. The main services required for ITER such as the electrical power, cooling water, fuel treatment, information flow, assembly and maintenance facilities, waste treatment, etc. are distributed in ancillary buildings and other structures throughout a site about 60 hectares overall.

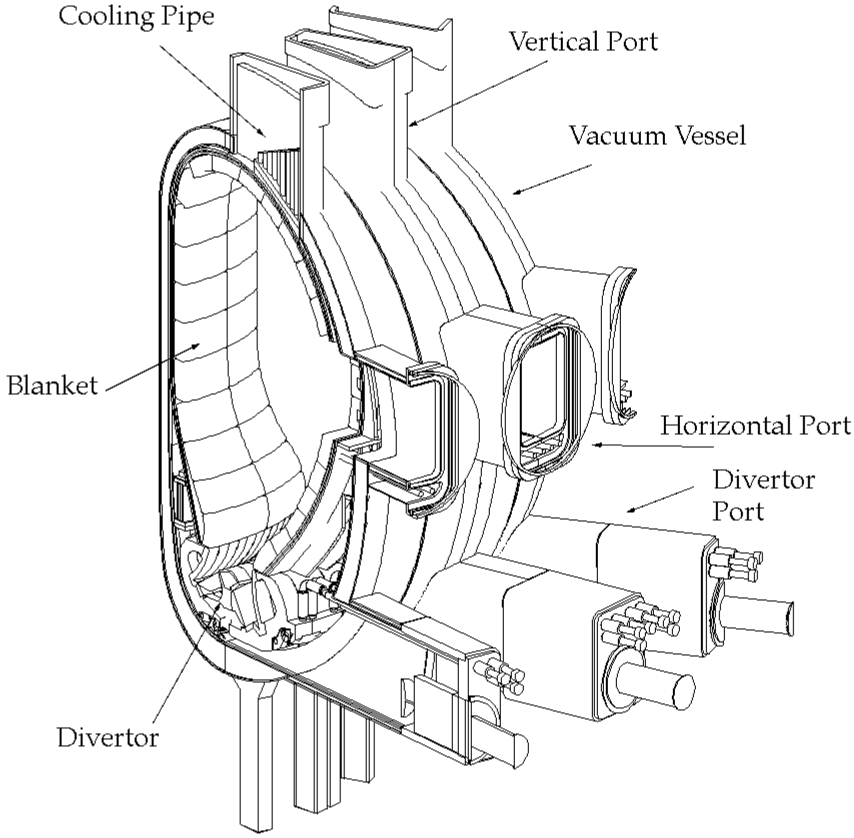



**FIG. 3. Isometric view of vacuum vessel, blanket and divertor.**

In order to ensure ITER would be site-able by any of the Parties, it was recognized that a design was needed that would be robust to variations in safety approach and criteria. For this purpose, the ITER safety design guideline was developed with all Home Teams and has been implemented in the ITER design. This includes radioactive dose and release design guidelines established in accordance with internationally accepted conservative criteria and the principle of As Low As Reasonably Achievable (ALARA), and the well-established nuclear design concepts of Defense in Depth and Multiple Lines of Defense.

A comprehensive safety assessment of the ITER design has been completed which clearly showed that a high level of safety is integrated into the ITER design. Radioactive effluents and emissions during normal operation are well within ITER design release limits established in accordance with internationally accepted criteria and the principles of ALARA. A comprehensive analysis of reference sequences has been performed using the best safety analysis computer programs available worldwide with conservative assumptions. Radioactive releases are well within the ITER design release limits conservatively established.

In addition to these studies, ultimate safety margins have been studied in order to demonstrate the intrinsic positive safety characteristics of magnetic fusion. The fusion reaction is self-limiting bounded by the ß-limit of the plasma. Under any failure conditions of the vacuum vessel or the in-vessel components, the fusion reactions are physically impossible. The radioactive inventory is moderate and the ultimate performance of confinement barriers that needs to be assured in accidents will be about one order of magnitude reduction for tritium and mobilizable metallic dust for ITER, whereas six to seven orders of magnitude reduction is required for iodine and rare gas in fission power reactors. Furthermore, radioactive decay heat densities are moderate. Therefore, structural melting of the plasma vessel is physically impossible and fast acting emergency cooling systems are not required.

Due to increasing financial constraints, the Parties are seeking cost reduction at the expense of assured performance. A Special Work Group of the Parties' representatives developed new technical requirements for possible changes to the original technical objectives with a view to establishing option(s) of minimum cost still satisfying the overall program objectives of the ITER EDA Agreement. The developed technical guidelines are as follows:

Plasma Performance

- Extended burn in inductively driven plasmas at Q>10 for a range of scenarios;
- Aim at demonstrating steady-state through current drive at Q>5; and
- Controlled ignition not precluded.

Engineering Performance and Testing

- Demonstrate availability and integration of essential fusion technologies;
- Test components for a future reactor; and
- Test tritium breeding module concepts.

In order to select major parameters and design features of a reduced cost ITER by the end of 1998, an intense joint work of Joint Central Team and Home Teams is under progress. The existing EDA technical output of design choices, generic technologies and large R&D results are generally directly applied to a reduced cost ITER. Therefore, a reduced cost ITER will be able to be well developed in a relatively short period and the detail design report will be available by July 2000 when the joint assessment of the ITER construction and operation by the Parties is planned.

## 2.3 ITER TECHNOLOGY R&D

The overall philosophy for ITER design has been to use established approaches and to validate their application to ITER through detailed analysis and by making and testing large/full scale models and prototypes of the critical systems. Major technical challenges in ITER are as follows:

- Unprecedented size of the super conducting magnet and structures;
- High neutron flux and high heat flux at the first wall/shield blanket;
- Extremely high heat flux in the divertor;
- Remote handling for maintenance/intervention of an activated tokamak structure;
- The first fusion machine with large radioactive inventory; and
- Unique equipment for fusion reactors such as fueling, pumping, heating/current drive system, diagnostics, etc.

ITER is being supported by extensive technology R&D to validate key aspects of design, including development and qualification of the applicable technologies and development and verification of industrial level manufacturing techniques with related quality assurance. Technology R&D for ITER is now focused on seven large projects each devoted to one of the key aspects of the design.

Two of the Projects are directed towards developing superconducting magnet technology to a level that will allow the various ITER magnets to be built with confidence. The Central Solenoid (CS) Model Coil Project and the Toroidal Field (TF) Model Coil Project are intended to drive the development of the ITER full-scale conductor, including the manufacturing of strand, cable, conduit and termination, and the conductor R&D in relation to AC losses, stability and joint performance. These Model Coil projects also integrate the supporting R&D programs on coil manufacturing technologies, including electrical insulation, winding processes (wind, react, and transfer) and quality assurance. In each case the Home Teams concerned are collaborating to produce relevant scale model coils and associated mechanical structures. The total planned production of 29 t of $Nb_3Sn$ stand, from seven different suppliers throughout the four Parties, has been produced and qualified.

The CS model coil will be the world's largest, pulsed superconducting magnet at a field of 13 T (2T/s), a mass of 100 t and stored energy of 640 MJ. The cabling and jacketing technology and winding techniques have been established and these activities have been completed. The heat treatment to react the superconducting alloy without degrading the mechanical properties of the Incoloy jacket, has been successfully achieved. The outer module of the CS model coil has been completed in Japan and the inner module has been completed in the U.S.

For the TF model coil (mass of 35 t, stored energy of 61 MJ), forging and machining of the radial plates are complete. Cabling and jacketing work is also complete. Winding, reaction treatment and transfer of the reacted conductor on the radial plates have been also successfully demonstrated. All this work was performed in the EU.

Dedicated coil test facilities, for the CS Model Coil in Japan and for the TF Model Coil in the EU, have been completed and stand ready to install the model coils for test programs aimed at gaining broad experience in their operational flexibility and to understand their performance margins. A 1 km jacketing has been completed in the RF which confirmed the fabrication feasibility of the full size both in the length and the cross section.

Three Projects focus on key in-vessel components, including development and demonstration of necessary fabrication technologies and initial testing for performance and assembly/integration into the Tokamak system.

In the Vacuum Vessel Sector Project, the main objective is to produce a full scale sector of the ITER vacuum vessel, to establish the tolerances, and to undertake initial testing of mechanical and hydraulic performance. The key technologies have been established and, in relation to manufacturing techniques, two full-scale vacuum vessel segments (half sectors) have been completed in industry, using a range of welding techniques, within the required tolerances. They were welded to each other at the Japan Atomic Energy Research Institute (JAERI) to simulate the field joint at the ITER site.

The Blanket Module Project is aimed at producing and testing full scale modules of primary wall elements, and full scale, partial prototypes of coolant manifolds and backplate, and at demonstrating prototype integration in a model sector. The key technology has successfully developed, tested and qualified a range of crucial material interfaces such as Be/Cu and Cu/Stainless Steel, bonded using advanced techniques in the four Parties. A full scale model, without the attached components, has been completed in Japan. The shield-modules are attached to the backplate by mechanical means based on flexible connections to the backplate and interlocking, insulated keys between adjacent modules. These components were also developed. Full-scale modules with attached components are under fabrication in the EU and will be tested to confirm that they meet the anticipated loads, the electrical insulating and the remote handling requirements together with the necessary accuracy of positioning. Also, experiments at the Fusion Neutronics Source Facility at JAERI has confirmed the shielding performance of bulk ITER shielding blankets and the effect of steaming between blanket modules.

The Divertor Cassette Project aims to demonstrate that a divertor can be built with tolerances and to withstand the very high thermal and mechanical loads imposed on it during normal operation and during transients. To this end, a full-scale prototype of a half-cassette is being built by the four Parties and subjected to high heat flux and mechanical tests in the US. The key technologies of the high heat flux components of the divertor have been successfully demonstrated in the four Parties, using W-alloy and CFC as plasma facing materials bonded to copper cooled by high velocity water using both hypervaportron and swirl-tube technologies. Components have been successfully tested to high heat fluxes up to 25 $MW/m^2$ for 1000 cycles.

The last two of the Large Projects focus on ensuring the availability of appropriate remote handling technologies which allow intervention in contaminated and activated conditions in reasonable time scales. The Blanket Module Remote Handling Project is aimed at demonstrating that the ITER Blanket modules can be replaced remotely. This involves proof of principle and related tests of remote handling transport scenarios including opening and closing of the vacuum vessel and of the use of a transport vehicle on monorail inside the vacuum vessel for the installation and removal of blanket modules. The procedures have already been successfully demonstrated at about one fourth scale so as to reduce the risk/cost for the development of full-scale equipment. Work is now in progress on a full scale demonstration. The fabrication of the full scale equipment/tools, such as rail-mounted vehicle/manipulator system, and cooling pipe welding/cutting/inspection tools has been completed in Japan.

In the Divertor Remote Handling Development, the main objective is to demonstrate that the ITER divertor cassettes can be removed remotely from the vacuum vessel and remotely refurbished in a Hot Cell. This involves the design and manufacture of full scale prototype remote handling equipment and tools, and their testing in a Divertor Test Platform (to simulate a portion of the divertor area of the Tokamak) and a Divertor Refurbishment Platform to simulate the refurbishment facility. Construction of the necessary equipment and facilities has been completed mainly in EU and integrated tests started.

While not an ITER R&D project, a noteworthy achievement in remote handling technology was the demonstration in JET of a fully remote exchange of the divertor. The complete Mark IIA divertor (144 modules) was replaced with the Gas Box Divertor (192 modules).

Heating and current drive technology is critical to ITER's mission and performance goals. Four types of systems are being developed: neutral beams, electron cyclotron, ion cyclotron, and lower hybrid. The overall goal for the neutral beam system is to deliver CW beams at 1 MeV (based on negative-ion sources) with about 17 MW injected per port. Negative-ion based systems have been used on JT-60U at currents of 14 A ($D^-$) and voltages of 300 kV. Operation of negative-ion sources for long periods (1000 s) at current densities of 200 $A/m^2$ have been achieved in Europe. Also, ceramic insulators with voltage holding capability of 1 MV are being developed at CEA Cadarache. The RF is developing a plasma neutralizer with a goal of yields of about 80% neutrals.

The electron cyclotron system relies heavily on the development of gyrotron and window technology. This has been the primary focus of the ITER EDA R&D program in Japan, EU, RF and U.S. The main goal is the demonstration of a 170 GHz, CW, ~ 50% efficient 1 MW gyrotron, together with dielectric windows for use on the torus and tube. Within the final four years of the EDA, the best gyrotron power performance at 170 GHz is 1 MW output at up to 2 s and 1.75 MJ in 10 s operation. Demonstration of a high efficiency ( ~50%) depressed collector operation at 100, 140 and 170 GHz was also obtained.

One of the major successes of the development program has been the demonstration of a water cooled, single disk, diamond window. The technology to manufacture large diameter diamond disks has been rapidly developed, and the material quality has been improved to the point that 2 MW CW windows are now theoretically feasible. The technique to bond the disks to metal tubes has been developed, prototype bonded disks have been subjected to the tube bakeout cycle, establishing the feasibility for their use on gyrotrons, high power tests of the material at 170 GHz have been made, and a complete window assembly has been fabricated and operated on a tube.

For ion cyclotron systems, one of the important parameters affecting the reliability of operation is the maximum operating voltage, which is dependent on plasma coupling. Present estimates show that, due to the low plasma coupling in ITER, the maximum system voltage exceeds 50 kV. Therefore, specific antenna optimization studies for high voltage operation have been carried out in the R&D program and validation tests have been performed in vacuum on a full scale prototype in the U.S. The prototype tests have demonstrated that RF voltages over 60 kV can be maintained for time intervals limited by overheating only.

One of the important R&D areas related to safety in ITER is beryllium interactions with steam and air. A substantial data base has been developed experiments in the U.S., Russia and Kazakhstan. This data also includes determination of the characteristics of irradiated Be.

In other ITER R&D work, a test facility has been built in the EU for an ITER model cryopump and tests have been done on cryosorption panels. A tritium fuel processing system using electrolytic reactors and palladium diffusers has been developed. One also notes the development at JET of the Active Gas Handling System which demonstrated the operation of a closed-loop tritium reprocessing system.

## 3.0 PROGRESS TOWARDS STEADY-STATE

One of the major quests in fusion energy research is to develop the ability to operate in a steady-state mode. This has important implications for plasma science and technology. This section summarizes the technology progress reported at this Conference and follows naturally from the previous section describing ITER technology R&D, which also plays a key role in essential steady-state technologies such as superconducting magnets, high-heat-flux components, plasma heating and fueling, and the tritium fuel cycle.

A number of plasma devices reported at this Conference are providing important advances in steady-state technologies in their experimental operations and/or R&D to support design and construction activities. For example, high power radiofrequency (RF) systems have been developed for long-pulse operation on Tore Supra. A two-minute discharge was obtained where 80% of the plasma current (800 kA) was sustained with RF waves. RF power densities of ³ 10 $MW/m^2$ have been achieved for ion cyclotron heating (40-80 MHz) while 25 $MW/m^2$ has been achieved for lower hybrid heating (3.7 GHz) for tens of seconds. A new guard limiter has achieved steady-state heat fluxes of 10 $MW/m^2$; it is made of carbon fiber composite tiles brazed on to Cu-Cr-Zr plates.

The Large Helical Device (LHD) construction was completed and operation began in 1998. A key feature of this long-pulse device is the use of superconducting coils. The helical coil system is composed of two interlinked, continuous coils (each weigh 120 tons) which will produce a peak field of 9.2 T with a stored energy of 1.6 GJ. The conductor is NbTi (chosen for its mechanical flexibility) in a pool-boiling configuration. Super-fluid He cooling will be used in a later phase to achieve 9.2 T at the coils. The poloidal coils use a NbTi, forced-flow, cable-in-conduit configuration to reduce AC losses.

The LHD project also has steady-state heating technology for ICH, ECH and neutral beams. The ICH system has demonstrated stand off voltages of 40 kV for 30 minutes. An ECH system (84 GHz) has produced 100 KW, also for 30 minutes. Also, a negative-ion neutral beam system at 180 keV with a 10 sec capability is being developed.

The Korea Superconducting Tokamak Advanced Research (KSTAR) Project is under construction and is a steady-state-capable advanced superconducting tokamak. Major parameters of the tokamak are: major radius 1.8 m, minor radius 0.5 m, toroidal field 3.5 Tesla, and plasma current 2 MA with a strongly shaped plasma cross-section and double-null divertor. The initial pulse length provided by the poloidal magnet system is 20 s, but the pulse length can be increased to 300 s through non-inductive current drive. The plasma heating and current drive system consists of neutral-beam, ion-cyclotron waves, lower hybrid waves, and electron-cyclotron waves for flexible profile control in advanced tokamak operating modes. The project has completed its conceptual design and has moved to the engineering design and construction phase. The target date of the first plasma is 2002.

A toroidal array of 16 toroidal field TF coils produces the 3.5-T toroidal field at the nominal plasma center. The TF and PF conductors are internally-cooled, cable-in-conduit superconductors. The conductor for the TF and five of the PF coils is $Nb_3Sn$, whereas NbTi is the conductor for two PF coils.

The $Nb_3Sn$ strand selected for the TF and PF coils is "HP-III" strand based on ITER superconducting strand specification. The eight inner PF coils form the central solenoid (CS) assembly. The magnet system and the rest of the tokamak systems are housed in a common cylindrical cryostat, which is evacuated prior to cooling down the superconducting coils. In order to verify characterizations of the KSTAR CS coil, two CS model coils have been fabricated and successfully tested.

The heating system on the KSTAR consists of neutral beam injection (NBI) and radiofrequency (RF) systems. The flexibility to provide a range of control functions including current drive and profile control derives from the use of multiple heating technologies: tangential NBI (energy of <120 keV, 8 MW), ion-cyclotron waves (frequency range of 20-60 MHz, 6 MW), and lower-hybrid waves (frequency of 3.7 GHz, 1.5 MW). The system can be upgraded to 21.5 MW with the addition of a 6 MW NBI or, if necessary, up to 27.5 MW by adding additional NBI or RF units and rearranging other ancillary hardware.

The objective of the JT-60 Super Upgrade design and related R&D is to establish an integrated physics and technology basis for a steady-state tokamak fusion reactor and for a reduced size of ITER. The toroidal field of 6.25 T at 4.8 MA is supplied by 18 superconducting toroidal field magnets, and the plasma current up to 10 MA with a current flat top of 200 s is generated inductively by 10 superconducting poloidal coil systems. Although a DD operation is mainly assumed in JT-60SU, the machine has been designed to be able to perform DT operation at a certain level by adding extra radiation shielding.

A $Nb_3Al$ conductor is adopted in JT-60SU. $Nb_3Al$, as a high field superconducting wire, has a much better critical current against density strain on the conductor than $Nb_3Sn$. It has been demonstrated that 11 km $Nb_3Al$ strand with Jc = 650 A/mm$^2$ can be made. Almost all important engineering techniques for manufacturing $Nb_3Al$ strand for JT-60SU has been established.

A design study in the U.S. has considered a reduced size superconducting tokamak with the goals of studying burn physics either in an inductively driven standard tokamak (ST) mode of operation, or in a quasi-steady state advanced tokamak (AT) mode sustained by non-inductive means. This is achieved by reducing the radiation shield thickness protecting the superconducting magnet and limiting the burn mode of operation to pulse lengths as allowed by the TF coil warming up to the current sharing temperature. "Marrying" the AT and ST modes of operation would allow optimized high gain burn physics studies in an Advanced Tokmak Burning Plasma Experiment (ATBX), sustained by RF and NBI current drive techniques. The proposed device has a major radius of 5.60 m, magnetic fields above 6.0 T, and currents in the range of 12-15 MA with pulse lengths up to 300 sec. Such a device would achieve $Q \gtrsim 10$ with conventional ITER confinement scaling (ST mode) as well as with advanced tokamak (AT) physics rules in the driven mode. An ATBX class device should have significantly lower costs than ITER, perhaps by as much as 50%.

## 4.0 FUSION ENERGY AND NEUTRON SOURCES

The worldwide fusion energy research community is preparing to enter a new era—the production of substantial amounts of fusion energy and neutrons. Several types of devices, including ITER as described in Section 2.0, were described at this Conference and are summarized in this section.

Further work on the Ignitor device was reported. The purpose of the Ignitor experiment (R=1.32m, a=0.47m, $\kappa$=1.8, $I_p \approx$12MA, $\beta$ =1.3%) is to produce deuterium-tritium plasma regimes where ignition can take place. At ignition, all thermal energy losses from the plasma are compensated by the $\alpha$-particle heating produced by D-T reactions. The reference design parameters of the machine could allow it to reach ignition by ohmic heating alone, but in order to expand the range of experiments that can be performed and to gain more control over the radial profiles of the current density and plasma temperature, an ICRH system has been included. In the present design, this system can deliver a similar power (around 20 MW) as the $\alpha$-particle heating under projected ignition conditions.

The first wall in Ignitor covers the entire surface of the vacuum vessel, with the exception of the ports; it basically functions as a fully extended limiter that offers the maximum possible area for spreading the plasma heat load. Different materials have been considered. In the end, molybdenum has been chosen due to several advantages: high radiative cooling, good plasma density control, and low temperature for vacuum conditioning. Plasma facing components are designed for heat peak loads in excess of 1.35 $MW/m^2$, considering a radiated power fraction of 70% of the total power. In the operating scenarios conceived for Ignitor, the plasma column rests either on the entire first wall or on the inboard side of it. This “limiter” solution relies on the unique characteristics of high density plasmas, involving a high radiative power in the edge region (“radiating mantle”) with a resulting uniformity in power load and a reduction of impurity production.

An important need for fusion energy development is the qualification of materials in an appropriate test environment. In an evaluation process based on a series of technical workshops, it was concluded that an accelerator driven D-Li stripping source would be the best choice to fulfill the requirements within a realistic time frame. In response to this need, an international design team with members from Europe, Japan, USA and Russia has developed under the auspices of the IEA during a Conceptual Design Activity Phase (1994-1996), a concept for an accelerator driven D-Li stripping source. This IFMIF reference design is based on conservative linac technology and two parallel operating 125-mA, 40MeV deuteron beams that are focused onto a common liquid lithium target with a beam footprint of 50 mm by 200 mm. The materials testing volume downstream of the Li-target is subdivided into different flux regions: the high flux test region (0.5 liter, 20-55 dpa/full power year), the medium flux test region (6 liter, 1-20 dpa/fpy), and low flux test regions (>100 liter, <1 dpa/fpy). The developed design includes extensive reliability, availability, maintainability as well as safety studies and is conceived for long-term operation with a total annual facility availability of at least 70%.

Concepts were also presented at the Conference for plasma-based neutron sources. One of these is the Gas Dynamic Trap (GDT) from the RF. The GDT is a mirror machine with a high mirror ratio (>10) where fast atoms of tritium are injected into a deuterium plasma at the turning points at the mirror configuration where most of the energetic ions are located. Studies indicate that neutron fluxes of 3 to 5 $MW/m^2$ are possible (depends on achievable ion energies) with relatively low annual tritium consumption rates (several hundred grams/year).

Another plasma-based neutron source concept that is receiving increased attention is the spherical tokamak (ST). Papers from the UK, US and China explored a range of applications, including DT experiments, volume neutron sources including transmutation of radiative wastes, and power reactors (see Section 6.0). As a neutron source, the ST is envisioned to have a major radius of 0.8 to 1.4m, a plasma current of 10 to 20 MA, fusion power levels of 100 to 200 MW, and neutron wall loadings of 1.0 to 2.5 $MW/m^2$. A technological issue is the center post (in ST’s there is little room for any neutron shielding) and design concepts employing both single-turn, dispersion-strengthened copper and flowing liquid metal (Li and LiPb) are suggested.

## 5.0 NUCLEAR TECHNOLOGY AND MATERIALS

Design and development work on ceramic breeder DEMO blankets is underway in Japan. The primary concept is a water-cooled concept (15 MPa water pressure with an outlet temperature of 320_C) utilizing small pebbles of $Li_2O$ and Be as a neutron multiplier in a layer configuration. The structural material is a reduced-activation ferritic steel alloy (F82H). The alternative concept uses helium gas as a coolant (8.5 MPa, 480_C). The R&D program includes development of fabrication technology (e.g., hot osostatic pressing for bonding), in-pile tests of the tritium release of lithium ceramics, chemical compatibility tests between Be, $Li_2O$ and F82H, and thermomechanical tests of first wall panels.

It has long been recognized that attainment of the safety and environmental potential of fusion energy requires the successful development of low-activation materials for the first wall blanket and other high heat flux structural components. Only a limited number of materials potentially possess the physical, mechanical and low-activation characteristics required for this application. This Conference included mainly results from Japan and the U.S. where the current structural materials research effort is focused on three candidate materials: advanced ferritic steels, vanadium alloys, and silicon carbide composites. Recent progress has been made in understanding the response of these materials to neutron irradiation.

Advanced ferritic steels have high thermal stress resistance due to low thermal expansion coefficients and higher thermal conductivities than austenitic steels. They exhibit good corrosion resistance in water and liquid metals, and oxidation resistance at ~550_C. Processing and fabrication techniques are at a mature state for these materials because of extensive industrial experience, and because of substantial research conducted to develop these steels for use as cladding and duct materials in liquid metal reactors. Finally, these steels have demonstrated excellent swelling resistance under fission reactor irradiation conditions.

The advanced ferritic steels being considered for fusion are a modified composition of conventional Fe – 9-12% Cr steels. In the low-activation steel Mo is replaced by W or V and Nb by Ta. The major technological challenges facing successful application of these steels include: potential interaction of a ferromagnetic material with the high magnetic field of a fusion plant; the effect of irradiation on fracture behavior; the effect of transmutation generated He on mechanical properties; and relatively low high-temperature creep strength. The U.S. and Japanese effort on advanced ferritic steels is part of an International Energy Agency cooperative program.

Experiments, measurements and calculations of the magnetic field in the Hitachi tokamak HT-2 showed that ferritic steel can be used as structural material in fusion devices, in spite of its ferromagnetism. The HT-2 has a stainless steel SS304 vacuum vessel and F82H plates were added just like a first wall, to simulate a ferritic vacuum vessel. The vacuum quality was not degraded. The magnetic field due to the F82H magnetization was small enough that it did not cause a problem for plasma discharges.

Vanadium alloys containing 4-5% Cr and Ti exhibit physical, thermal, and mechanical properties that are favorable for fusion applications. These alloys are particularly attractive in combination with lithium-cooled blanket designs. Favorable characteristics include low long-term radioactivity, high heat load capacity and resistance to void swelling. The unirradiated tensile properties and limited thermal creep data suggest an upper temperature limit between 650_C and 750_C. The industrial experience and manufacturing capacity of vanadium alloys is very limited in comparison with advanced ferritic steels; however, scale-up from laboratory heats to commercial production of 500 to 1200 kg ingots has been successfully accomplished.

A major focus of the U.S. research and development effort for vanadium alloys is to explore the effects of irradiation on constitutive behavior and fracture properties at low temperatures. A series of experiments has recently been completed at temperatures ranging from ~100_C to ~500_C and doses between 0.1 and 18 dpa. Preliminary results indicate significant changes in mechanical properties at temperatures below about 400_C. At high temperatures (³650_C) the primary concern is the effect of neutron damage and helium transmutation on creep rupture properties. As with advanced ferritic steels, the combined effects of neutron damage and helium generation can cause loss of tensile and creep ductility by growth and coalescence of helium bubbles at grain boundaries. Considerable experience has been obtained on V-4Cr-4Ti irradiated in the range of 420_C to 600_C to neutron doses of 24 to 32 dpa, and with helium generation rates in the range of 0.4 to 4.2 appm/dpa. In this environment, the material shows promising resistance to radiation-induced embrittlement and swelling.

Development of V-Ti-Cr alloys containing Si, Al and Y are underway in Japan. The loss of elongation, which may restrict the lower temperature limit of the design window for vanadium alloys, has a correlation with the concentration of interstitial impurities, especially oxygen. Therefore, purification of the alloy is one of the ways to retain uniform elongation. Since the addition of Al and Y made the oxygen concentration in solution low, modification by the addition of Si, Al and Y may be a promising way for the development of vanadium alloys.

Silicon carbide (SiC) composites are attractive for structural applications in fusion energy systems because of their low-activation and afterheat characteristics, excellent high-temperature properties, corrosion resistance and reasonable dimensional stability. High-strength, continuous fibers with near stoichiometric SiC composition reinforce the composite and impart improved fracture toughness compared to monolithic SiC. Utilization of these new materials for fusion applications will require optimization of their radiation stability, thermal conductivity, gas permeability, chemical compatibility with fusion relevant environments, and joining methodology. Very little is known about the radiation performance of these materials, thus the primary issue is the development of a radiation tolerant SiC fiber and the integration of that fiber into an acceptable composite architecture.

Irradiation experiments on standard, commercially available SiC composites have demonstrated that a significant reduction in strength occurs at radiation doses as low as one dpa. The thermal conductivity of SiC or SiC composites is also significantly reduced by irradiation-induced point defect generation. For exposures to about 26 dpa, the thermal conductivity of the standard composite was reduced by 60% from 7 to 3 W/m-K at 800_C. Recent advances in fiber and composite manufacturing techniques have demonstrated the capability to produce material with much improved thermal conductivity of 35 W/m-K at 1000_C. Newly developed, advanced SiC fibers, as well as composites made from these fibers, are being produced and evaluated for fusion applications.

A joint U.S.-Japan program (Jupiter) is studying the dynamic behavior of fusion reactor materials and their response to variable and complex irradiation conditions. The materials under study include reduced-activation structural materials (ferritic steels, vanadium alloys and SiC composites), other refractory and copper alloys for high-heat-flux applications, and ceramics for insulator needs. The most important variable is the irradiation temperature history. The results also show that radiation-induced electric degradation is not a problem in ceramics up to 3 dpa.

## 6.0 ADVANCED DESIGN CONCEPTS

The papers reported at the Conference on Advanced Design Concepts emphasized the two principal toroidal confinement concepts: tokamak and stellarator/helical reactor configurations. Tokamak reactor concepts included a broad range of low and high aspect designs as well as several design innovations.

A Russian paper developed the parametric dependence of major radius, plasma current and fusion power on aspect ratio, magnetic field, elongation, wall loading, plasma beta and safety factor. High aspect ratio was preferred because of reduced plasma current, higher bootstrap fraction and better use of the maximum magnetic field at the coil. A Japanese paper explored a design with A=8, a major radius of 12m, a bootstrap fraction of 40% and a fusion power of 3 GW.

An example of an intermediate aspect ratio (3.4) is the Japanese Compact Reversed Shear Tokamak design. This design is similar to ITER with advanced tokamak modes. The device has R=5.4m, $\kappa$=2.0, Ip=12 MA and Bmax = 12.5 T. The reactor produces ~3 GW of plasma power with a wall loading of 4.5 MW/m$^2$ and operates at a thermal efficiency of 41% using a super-heated steam cycle.

At the other end of the aspect ratio spectrum is the spherical tokamak (ST). The ST as a power reactor has been explored by Europe (led by Culham) and U.S. studies (ARIES Team) with similar parameters. As an example, the 1000-MWe ARIES-ST power plant has an aspect ratio of 1.6, a major radius of 3.2m, a plasma elongation (at 95% flux surface) of 3.4 and triangularity of 0.64. This configuration attains a $\beta$ of 54% (which is 90% of the maximum theoretical $\beta$). While the plasma current is 31 MA, the almost perfect alignment of bootstrap and equilibrium current density profile results in a current-drive power of only 31 MW. The on-axis toroidal field is 2.1 T and the peak field at the TF coil is 7.6 T, which leads to 288 MW of Joule losses in the TF system.

The design of the center-post is probably the most challenging engineering aspect of a spherical tokamak fusion system. Single-turn TF coils are preferred in order to reduce Joule heating through higher packing fraction and reduced shielding requirement (no insulation) even though such a TF system will require high-current, low-voltage supplies with massive busbars. The ARIES-ST TF is a single turn configuration. The TF return legs form a continuous shell with a constant area cross section; i.e., the shell thickness increases proportional to major radius away from the center-post. This shell configuration also allows the TF system to act as the primary vacuum boundary and the need for an additional vacuum vessel is eliminated. The maintenance scheme envisioned for ARIES-ST centers around removal of the fusion core and the center-post as a complete unit from the bottom. Therefore, demountable joints are provided on the outboard at below the mid-plane.

Center-post is cooled with cold water with an inlet and outlet coolant temperatures of 30°C and 75°C, respectively. The maximum conductor temperature is 125°C. The coolant removes the center-post Joule losses (230 MW) and the nuclear heating (160 MW). A 20-cm thick first-wall and shield is included in the inboard. This shield reduces the Joule losses in the center-post because of the reduced nuclear heating and the associated cooling requirement.

The reference ARIES-ST blanket design uses ferritic steels as structural material with helium as coolant and LiPb as both a coolant and a tritium breeder. SiC composite inserts are used in order to achieve a high-coolant outlet temperature and a reasonable power conversion efficiency. The 12-MPa helium coolant exits the blanket at 500°C (set by the maximum operating temperature of ferritic steel) but is superheated by the LiPb coolant which has an exit temperature of 700°C. Through this technique, a 45% thermal cycle efficiency has been obtained.

Several papers at the Conference described innovative design concepts for tokamak reactors. Examples include the following:

- The use of high temperature superconductors in plasma stability control coils (Japan). This results in improved plasma stability control loading to higher elongation and smaller reactors.

- Use of force balanced coils (FBC's) to balance the net centering force and net radial hoop force due to poloidal and toroidal currents (Japan). The FBC's can function as both toroidal and primary coils for ohmic heating, thus simplifying the design of pulsed tokamaks.

- A pebble drop divertor (Japan). A large number of small refractory pebbles are used as divertor surface. These divertor pebbles are dropped in the divertor space to form a curtain at the outer edge of the plasma. A marked feature of the pebble drop divertor system is the use of multi-layer pebbles. The multi-layer pebbles consist of at least three layers to satisfy the requirements for plasma facing materials. A kernel is made from graphite or low-Z refractory ceramic like SiC, BeO, or $Al_2O_3$.

- A liquid lithium divertor (Russia). A lithium capillary-pure system (being tested in the T-11 tokamak) has been proposed which is capable of steady-state heat loads of up to 100 MW/m$^2$.

The use of the tokamak reactors as breeders of fissile fuel and burning of high-level, long-lived radioactive wastes (China and Japan) were also discussed as well as the life cycle omission of $CO_2$ from tokamak reactors in comparison with other energy systems (Japan).

The stellarator as a reactor concept has been explored as an extension of a LHD-type device (compact helical devices in Japan) and the Wendelstein 7-X device (high aspect Helias device in Germany). Owing to inherently current-less plasmas, such reactors have attractive advantages, such as steady operation and no dangerous current disruptions.

The LHD concept is characterized by two advantages: a simplified superconducting continuous-coil system; and an efficient closed helical divertor. Focusing on these advantages, two reactor candidates have been proposed: a Force-Free Helical Reactor (FFHR) with a continuous-coil system; and a Modular Heliotron Reactor (MHR) with an efficient closed divertor. Two possible approaches are investigated: increasing the toroidal field in FFHR and increasing the plasma minor radius in MHR. Both are characterized by a major radius of 10m with aspect ratios of 8.3 and 5.9, respectively. In both cases, the helical coil-to-plasma distance, $\delta L$, for the blanket and shielding is a common constraint. Therefore, it is necessary to introduce innovative concepts of blanket systems and coil configurations in both cases.

The dimensions of a Helias reactor are: major radius 22m, average plasma radius 1.8m, magnetic field on axis 4.75 T, fusion power about 3000 MW. Neoclassical transport sets a lower limit on plasma confinement. Extrapolating empirical scaling laws to a Helias reactor shows that anomalous confinement determines the ignition conditions. The coil system of the Helias reactor is roughly four times as large as the Wendelstein 7-X device and produces about the same field configuration. The maximum field strength of 10 T at the coils is small enough to use NbTi superconductors at 1.8 K. The 'cable-in-conduit' conductor is designed for a nominal current of 37.5 kA and has an aluminum alloy jacket.

The combination of helical windings, planar toroidal field coils and vertical field coils, installed in most of the stellarator devices is of advantage for experiments since it provides a large amount of flexibility. However, helical windings have only a limited potential regarding field optimization with respect to plasma confinement and stability and pose technical difficulties due to the interlinked TF coils. The concept of modular coils as used in the Helias Reactor overcomes these difficulties by needing only one coil system.