

# 18th IAEA Fusion Energy Conference Summary of Technology and Power Plants

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## 1. Introduction

There were 90 papers presented at the Conference in the category of Technology and Power Plants accounting for about 25% of the total number of contributions. As was the case at the previous meeting, a large number of papers dealt with the ITER-Engineering Design Activity (EDA) and ITER technology R&D. In the author's opinion, the rapid progress made during the ITER EDA extension on the completion of the new ITER-FEAT design and its physics and technology R&D validation stands out as the highlight of the meeting. Steady progress is being made on several other technology fronts as well. The results point towards emerging research trends in the following areas:

- steady-state operation with advanced performance and the increasingly important role of enabling technologies in achieving this goal,
- advanced, high-performance, environmentally attractive materials for the fusion energy goal,
- reactor and near-term applications studies that exploit advances both in the physics and technology fronts for lower cost of electricity and improved safety and environmental features, and
- socioeconomic studies that are helping to promote the attractive features of fusion and its public acceptance.

The remaining sections of this paper are organized along the lines of these major themes; namely, 2) ITER EDA Design, 3) ITER Technology R&D, 4) Progress Towards Advanced Performance and Steady State, 5) Compact Cu Burning Plasma Experiments and Neutron Sources, 6) Advanced Materials Research, 7) Power Plant Design and Economic Forecasts, and 8) Conclusions.

## 2. ITER EDA Design

The focus of the EDA since 1998 has been on ITER-FEAT, a new design to meet the revised (reduced) technical objectives and a cost reduction target of ~50%. The new design will allow ITER to meet the following technical and programmatic objectives:

- the exploration of a range of burning plasmas ( $Q > 10$  reference) with capability to progress to possible modes of steady-state operation at  $Q > 5$ , and
- blanket module testing at neutron wall loading  $> 0.5 \text{ MW/m}^2$  and fluence  $> 0.3 \text{ MWa/m}^2$  over the 20-year project duration.

In the successful re-design effort, considerable emphasis was placed on reducing the machine size to meet the cost reduction target and increasing the plasma shaping capability to compensate for the resultant loss of physics margins (due to the lower plasma volume). The 15 MA inductive

reference configuration ( $R/a=6.2\text{m}/2\text{ m}$ ,  $q_5 = 1.7$ ,  $q_5 = 0.33$ ) that is based on ELMy H mode and conservative ITER Physics Basis 98(y,2) energy confinement scaling, results in a fusion power at the nominal design point ( $H_H = 1$ ,  $\beta_N = 1.77$ ,  $n_e/n_{GW} = 0.8$ ) of 400 MW for 400 s pulse duration [see Fig. 1]. The ELMy H mode was chosen for its reproducibility, robustness, and applicability to long pulse operation, not to mention the existence of expansive multi-tokamak databases on energy confinement scaling and L-H mode transition powers.

At  $H_H = 1$ , the 15-MA inductive scenario will meet the  $Q = 10$  objective over a range in fusion power from 200 MW to 700 MW while remaining within acceptable limits on plasma density  $<$  Greenwald limit,  $\beta_N < 2.5$ , and  $P_{\text{loss}}/P_{\text{LH}} > 1.3$ . A probabilistic analysis gives a high degree of confidence that the primary performance objective of achieving  $Q = 10$  will be met. Simulations using 1.5-D transport codes indicate additional potential performance gains, should they be necessary, from density profile peaking by high field side pellet fueling and ion cyclotron (IC) heating scenarios that favor an increase in the ion heating fraction.

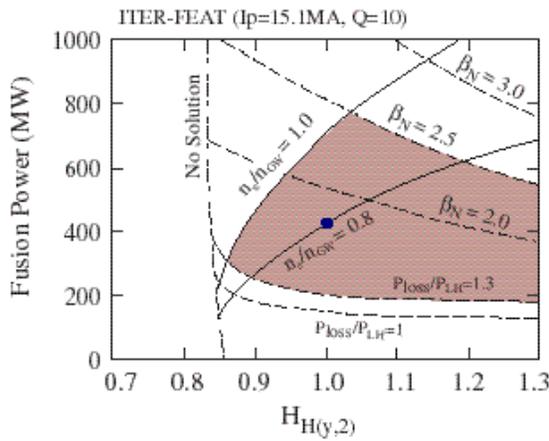


FIG. 1a.  $Q = 10$  domain (shaded) for  $I_p = 15.1\text{ MA}$  ( $q_{95} = 3.0$ ).

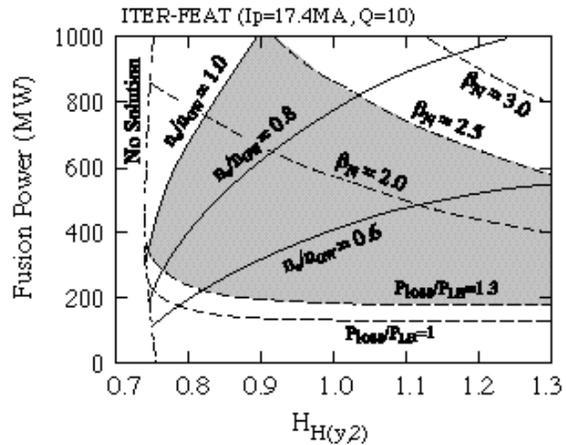


FIG. 1 b.  $Q = 10$  domain (shaded) for  $I_p = 17.4\text{ MA}$  ( $q_{95} = 2.6$ ).

Inductive operation at 17 MA provides greater physics margins and substantially widens the operating space. At 17 MA, the design also allows for higher fusion gain and the possibility of ignition. With nominal helium pumping, i.e.,  $\dot{n}_e^*/E = 5$ ,  $Q = 50$  is attained but the ignition condition is terminated within about 40 s because of the accumulation of helium. Ignition can be maintained for confinement times 5% to 10% higher than predicted ( $H_H = 1.05 - 1.1$ ) or by increasing the pumping of helium (to  $\dot{n}_e^*/E = 4$ ) through stronger core fueling (i.e. high field side pellet injection) and edge pumping.

The revised design also allows for the possibility of steady state operation at  $Q > 5$ . These reduced current (9 MA to 12 MA) scenarios require up to 100 MW of current drive power and are more challenging from the physics perspective requiring highly shaped plasmas ( $q_5 = 1.9$ ,  $q_5 = 0.4$ ) with weakly monotonic or negative shear profiles and  $H_H = 1.2 - 1.5$ ,  $\beta_N = 3 - 3.5$ . Provision is made in the design to enhance the prospects for high-, long-pulse operation by including saddle coils for resistive wall mode stabilization and sufficient electron cyclotron current drive (ECCD) power to stabilize neoclassical tearing modes.

In summary, the redesign effort undertaken during the past two years has produced a new design that meets the revised technical objectives and, while detailed costing is yet to be performed, a simple re-scaling exercise, based on the cost analysis of the 1998 ITER design indicates an overall reduction to about 56% of the estimated capital costs of the 1998 design. One objective of the technology R&D program to be discussed below is to validate manufacturing processes and design simplifications that would lead to further cost reductions.

### **3. ITER Technology R&D**

Substantial progress has also been made in the past two years in the technology R&D program to validate the key aspects of the design including the manufacture, performance and operation of several critical subsystems. Recent test results from the so-called seven Large Projects, in particular, confirm the readiness of the project to proceed to the construction phase.

#### **3.1 Magnet System**

Two of the Large Projects involve development and testing of prototype magnets for the ITER-FEAT all superconducting magnet system. The new design incorporates the following features:

- 18 high-field (11.8 T) Nb<sub>3</sub>Sn wedged toroidal field (TF) coils,
- a 13.5-T Nb<sub>3</sub>Sn vertically segmented Central Solenoid (CS),
- six NbTi poloidal field (PF) coils, and
- three sets of NbTi correction coils to correct field errors and stabilize resistive wall modes.

While there are many similarities in the design with the 1998 ITER, the new design differs principally in the adoption of a segmented, free standing central solenoid for increased plasma shaping capability and the resultant change to a wedged design for the TF coil inner legs to sustain the centering forces.

The CS Model Coil is a 100-t assembly with dimensions that approximate those of the individual elements of the ITER-FEAT CS stack design. It consists of nested inner and outer modules made from cable in conduit conductor (CICC) utilizing high current density Nb<sub>3</sub>Sn conductor strand ( $J_c = 550\text{-}650 \text{ A/mm}^2 @ 12 \text{ T}$ ) in a 51 mm × 51 mm cross section Incoloy 908 jacket. In the recently completed testing phase at the Naka-JAERI test facility, the coil assembly exceeded all test goals. These include a maximum field of 13 T, a ramp-up rate of 1.2 T/s, a fast discharge ramp-down rate of – 1.5 T/s, a 2.3 K current sharing temperature margin and 10,000 cycle fatigue test of the insert coil. The completion of the 650-MJ CS Model Coil and its successful testing a milestone in the development of fusion technology provides sufficient reassurance of the readiness to construct the ITER-FEAT CS and confirms the adequacy of the conductor to meet the ITER CS operation conditions.

A 31-t reduced-scale TF Model Coil has also been completed demonstrating the manufacturing techniques of the ITER TF coil. The model coil will be tested first on its own and later in conjunction with the LCT coil in the TOSKA facility where a field of 9.7 T at 80 kA will be achieved. By comparison, the peak field and the operating current are 11.8 T and 68 kA in ITER-FEAT. The model coil uses a cable similar to the full size TF coil cable and the cross section of

the TF model coil is smaller but comparable in size to that of the ITER TF coil. A test of a single layer TF insert coil will be performed in the CS Model Coil test facility at JAERI at up to 13 T.

In addition, large forged and cast pieces (about 30 t and 20 t, respectively) have been produced in the TF coil case development program. Investigation of the properties of the forging has revealed yield stress values exceeding the 1,000-MPa requirement.

### **3.2 Vacuum Vessel and Blanket/First Wall**

The ITER-FEAT vacuum vessel (VV) design has many similarities to the previous design including the familiar SS 316L(N)-IG double-walled structure with shielding and cooling between shells. The in-vessel back-plate has been eliminated to allow the largest possible plasma volume within the reduced overall size of the tokamak. The 4.5-t blanket modules are now directly attached to the vacuum vessel with flexible titanium supports.

A major milestone was achieved in 1998 with the fabrication of the 15 m high full-scale (1998 ITER) sector model and full-scale mid plane port extension within the target tolerance of  $\pm 5$  mm. On-site welding of the VV sector assembly was successfully demonstrated and an analytical model for weld shrinkage was developed and validated for the new ITER design. The major accomplishment reported at the Conference was the successful welding of the port extension to the VV sector using remote welding tools. This sophisticated system features a robot arm on a rail mounted vehicle giving a six degree-of-freedom motion capability. In addition several new welding, cutting and inspection technologies are under investigation to improve the efficiency of VV assembly and maintenance operations.

The blanket module design for ITER-FEAT has been simplified with an eye toward reduced manufacturing costs. This simpler design now incorporates a separable first wall of multiple faceted flat panels mounted to the shield. Significant progress has been made in all manufacturing and testing tasks of the blanket and first wall (FW) project. In particular, the feasibility of manufacturing the FW and limiter with a CuAl25 alloy heat sink has been fully demonstrated and one step hot isostatic pressing (HIP) process parameters for SS/SS, Cu/SS and Be/Cu have been determined. Two Be tile protected mock-ups have withstood heat fluxes of 0.7 MW/m<sup>2</sup> for 13,000 cycles and limiter mock-ups with brazed Be have achieved 6.5 MW/m<sup>2</sup> for 1,000 cycles. Two full-scale 4.5-t blanket shield modules have been manufactured, one featuring a solid HIP process and the other using Powder Hip technology as a potential lower cost blanket manufacturing option.

### **3.3 Divertor**

The ITER-FEAT divertor design consists of 54 cassettes onto which the high heat flux (HHF) components are mounted. The new design is smaller than the 1998 ITER design but retains similar features and functions. The plasma facing components are made of carbon fiber composite (CFC) and tungsten. At present, CFC is the material of choice for the highly loaded divertor strike points. Tungsten is utilized for the upper vertical targets (VT) and for the private region liner and dome high heat flux (HHF) components to minimize sputtering and tritium co-deposition. The B2-Eirene edge plasma code has been used extensively to guide the design and

ensure its functionality. Simulations indicate that a V shaped geometry at the strike point tends to lock recycling neutrals in the corner regions thereby promoting partial plasma detachment. The resultant increase in charge exchange and radiation losses reduces the peak heat flux to the vertical targets by as much as 30%. Large openings in the inner and outer support of the dome were also dictated by modeling to promote a flow of hydrogenic neutrals from the inner strike zone to the outer strike zone in order to minimize asymmetries in the VT peak power loads. The ITER-FEAT divertor design geometry should ensure that the peak power load at the outer strike zone is kept below  $10 \text{ MW/m}^2$  for all but a few ITER operating scenarios.

The development of HHF component solutions that meet the varied needs has kept pace with the evolving design. Several armor to heat sink joining technologies (CFC to Cu and W to Cu) have been developed and many small and full-scale mockups of the VT HHF components have been fabricated and successfully tested at parameters that exceed the ITER goals. In particular, the JA Home Team has developed a large-scale monoblock type CFC/Cu mock-up that has been tested at  $20 \text{ MW/m}^2$  for 1,000 cycles. The EU Home Team has manufactured a medium-scale VT that has withstood 2,000 cycles at  $20 \text{ MW/m}^2$  on its CFC section and 1,000 cycles at  $15 \text{ MW/m}^2$  on its W macro-brush clad section.

### **3.4 Remote Handling and Maintenance**

Large-scale remote handling and maintenance programs support and complement the divertor cassette and blanket module development projects. Two dedicated divertor remote handling and maintenance platforms are now operational—the Divertor Test Platform which addresses full-scale simulations of all handling operations inside the VV and the Divertor Refurbishment Platform which allows simulation of critical hot cell operations. These facilities are motivated by the requirement that the divertor cassettes be removed and refurbished several times during the machine life. Another key factor is the replacement of the plasma facing components in hot cells thereby allowing the reuse of 10-t of the 12-t cassette mass and dramatically minimizing the amount of activated waste generated. The ITER-FEAT cassettes are smaller than their 25-t ITER 1998 counterparts and the space between the cassette bottom and vacuum vessel has been reduced necessitating a new scheme (the cantilever multifunction mover, or CMM) for all radial transport operations.

The basic feasibility of in-vessel divertor maintenance operations was demonstrated in 1998 and more extensive simulations carried out since then have confirmed the early results. In particular, the accuracy of the various movers has been proven sufficient to allow cassette handling and positioning in accordance with the requirements. Moreover, realistic cassette misalignments can be accommodated. The original attachment scheme (based on shear keys) for the PFCs has been extensively studied and, to overcome its limitations due to tight mechanical tolerances, a new multi-link concept is under investigation. With the successful operation of these two facilities, the feasibility of divertor handling and refurbishment has been confirmed.

The Blanket Test Platform was completed in 1998 for the purpose of demonstrating the remote handling of full-scale ITER blanket modules. The system consists of vehicle manipulators working on a rail transporter in a toroidal ring structure. Remote handling tests have been performed successfully using the remotized in-vessel transporter system for the installation and

removal of a 4.5-t blanket module within the specified tolerances. A new suppression control scheme has recently been developed that reduces from 2-g to nearly 0-g end-effector acceleration caused by dead load transfer during installation and removal of blanket modules. The current R&D focus is on removal of mechanically attached blanket modules using a sensor based (force and distance) feedback control scheme to deal with the extremely tight tolerances ( $< \pm 0.25$  mm) between the module and the attachment keys and pin on the back-plate. Insertion tests have been carried out successfully with a misalignment of 3 mm in position and 0.5 degree in orientation.

### 3.5 Heating and Current Drive

ITER-FEAT will be provided initially with 73 MW of heating and current drive power consisting of  $2 \times 16.5$  MW units of neutral beam injection (NBI), and 20 MW each of electron cyclotron (EC) and ion cyclotron (IC) power in similar size units. For comparison, the  $Q = 10$  reference scenario requires 40 MW of auxiliary power at the design point and the L to H transition power is estimated to be 48 MW.

The goal of the ITER NBI program is to develop a long-pulse, 1-MeV negative ion based system for both heating and off-axis and on-axis non-inductive current drive. Significant reliability and efficiency gains have been made recently in the development of high confinement, high current ion sources and their operational characteristics to dramatically reduce stripping losses in the acceleration grids. Efficiency improvements made on the JT-60U system over the past two years have resulted in an increase in the deuterium acceleration efficiency from 55% to 74% and a reduction of the ground and accel grid heat loads by a factor of two. As a result, the overall performance of the system has been increased to 5.2 MW @350 kV for 0.77 s and the maximum pulse length is 2 s at 4.0 MW and 360 keV. In addition high ionization fraction ( $> 40\%$ ) plasma neutralizers are under development that have the potential to increase the ion neutralization efficiency for the ITER NBI system to the 60% to 80% range.

The 170 GHz ITER EC system is being developed for on- and off-axis non inductive current drive and control of neoclassical tearing modes—toroidal and poloidal beam steering capability is provided ( $\pm 12.5^\circ$  and  $\pm 5^\circ$ , respectively). The focus of the development programs in the EU, RF, Japan and the U.S., for ITER and other uses, is on gyrotrons capable of producing 1-MW/CW radiation in the millimeter wavelength range. The various systems under development are approaching this goal by incorporating innovations such as low loss (loss tangent  $< 10^{-5}$ ) high thermal conductivity synthetic diamond windows that allow Gaussian beam outputs at over 1 MW, the incorporation of depressed collectors for efficiency gains ( $> 50\%$ ) and reduced collector power dissipation and operation at very high order modes to reduce thermal cavity loads.

IC heating in the 35 to 65-MHz range is being implemented on ITER for heating via proven heating scenarios ( $2 T_e$ ,  $3He$ ,  $D$ ) and on-axis fast wave current drive at 55 MHz. This versatile system is based on the in-port resonant double loop (RDL) antenna technology in use on the Tore Supra tokamak and under development for the long-pulse (300 s) KSTAR tokamak. The IC launcher consists of an array of  $4 \times 2$  radiating elements fed by eight coaxial transmission lines each carrying a nominal RF power of 2.5 MW. The system has been designed for low electric

field at the plasma side of the current strap (for high power density) and to be highly insensitive to load variations (as created by ELMs for example).

### 3.6 Fueling and Pumping

As in the ITER 1998 design, the torus exhaust-pumping scheme for ITER-FEAT employs high-speed, fast-regenerating cryopumps. The unit size and pumping speeds are the same in both designs. Because of the improved divertor conductance of ITER-FEAT and the reduction in the nominal pulse length from 1200 s to 450 s, the number of pumps is reduced from 12 to 6. For both machines an additional four pumps are needed to allow on-line regeneration for pulse lengths greater than nominal. A 50% scale model pump has been developed and the first results indicate that the pumping speed significantly exceeds the requirement of  $1 \text{ liter cm}^{-2} \text{ s}^{-1}$  for the principal gas components (pure protium, deuterium and mixtures containing ITER relevant impurities including helium).

Core plasma fueling in ITER-FEAT will be accomplished by fast pellet injection. Provision is being made for high field side launch as a means to enhance density profile peaking and increase fueling efficiency. In a process called isotopic tailoring, tritium pellets are used to fuel the core while strong deuterium gas puffing maintains the deuterium rich scrape off layer plasma that comes in contact with the first wall. The parameters of the pellet injection system are presently estimated to be characteristic size = 3 to 7 mm, speed = 300 to 1000 m/s, frequency = 2 to 50 Hz, and an injection reliability of 99% for 1000-s cycles. The goal of the R&D program is to develop the technology for a steady-state, low-inventory tritium pellet injection system that meets these parameters. A tritium compatible prototype injector based on a gas gun driver for pellet acceleration and a screw extruder based hydrogen feed system was recently completed and is under test. In the initial tests, protium pellets were formed and accelerated to 0.4 to 0.5 km/s at 1 to 2 Hz for 1500-s continuous steady-state injection.

### 3.7 Safety

The approach adopted for the ITER-FEAT safety design and safety systems implements the philosophy of deployment of fusion's favorable safety characteristics, passive safety and defense in depth. This proactive philosophy was evident from the results reported at the Conference including:

- Detailed temperature response calculations that indicate that active cooling systems will not be required to remove the small decay heat from the structure in the event of a loss of coolant accident. This would seemingly eliminate the possibility of radioactive release from the vacuum vessel whose temperature is not expected to exceed  $600 \text{ }^\circ\text{C}$  even under worst case conditions.
- The detection of small coolant water leaks in the ITER vacuum system has been shown to be effective for detecting cracks before they propagate, confirming the principal of leak before break. Therefore, it can be concluded that the structural integrity of the ITER vacuum vessel can be assured by monitoring water leakage.
- A passive vacuum vessel pressure suppression system that limits the pressure rise resulting from coolant interaction with hot vessel components following an ingress of coolant event

has been demonstrated on a 1/1600-scale (volume) test facility. An analytical model has been developed that predicts the in-vessel pressure response to within 5%.

- The decay heats have been measured for 32 materials of interest to ITER and found to be predictable. In addition, neutron transport and activation codes are being validated with 14-MeV neutron experiments for shielding and dose rate assessments.
- In the area of in-vessel tritium inventory reduction, techniques are being explored to remove the C/T co-deposited layers that are consistent with ITER operating conditions and characteristics. Recent tests indicate that ozone is a promising candidate for removal of C/T co-deposited layers at 200 K lower than O/air (and therefore consistent with ITER vacuum vessel temperature constraints).

### **3.8 Conclusions—ITER EDA and Technology R&D**

In summary, remarkable progress has been made during the EDA extension in both the design and technology R&D phases of the ITER project. The new ITER design meets the revised technical objectives and the initial cost estimate approaches the 50% reduction target. The technology R&D program, in particular the test results from the seven large projects, has validated the key aspects of the design and continues to be an unprecedented model for international collaboration. The EDA extension agreement requirements have also been satisfied the new design continues to provide an integrated demonstration of the scientific and technological feasibility of fusion. The design incorporates provisions for an extended high-Q ( $> 10$ ), driven-burn with fusion powers in the 200 to 700-MW range and burn duration  $\sim 300$  s. Ignition is not precluded and the possibility exists to achieve true steady-state operation at  $Q > 5$ . The favorable evaluation of ITER 1998's safety and environmental characteristics remain valid in ITER-FEAT and encouraging positive steps have been taken in consideration of potential ITER hosting sites in Canada, Japan, and Europe.

## **4. Progress Towards Advanced Performance and Steady State**

The trend towards steady-state operation and advanced performance was a common theme at the Conference. This was evident from the results presented by the experimental teams, the experimental facilities planned in the near future and from the technologies needed to enable the improved levels of performance. Indeed, it is difficult to imagine steady state operation in improved energy confinement regimes at high  $Q$  without technological innovations in superconducting magnets, PFCs and the heating, current drive and fueling tools required to manipulate the plasma profiles.

### **4.1 Existing and Planned Experimental Facilities**

The large helical device (LHD), a nominal 3.0-T, 3.9-m radius all superconducting stellarator, completed trial operations in March 1998 and has since completed two additional experimental campaigns, the latest having recorded over 17,000 plasma discharges. Of particular interest is the uneventful performance of the 960-MJ magnet system which consists of two NbTi/Cu/Al composite helical coils cooled by pool boiling LHe and six force flow cooled NbTi/Cu CICC PF coils. Other key components of the system include a 5.6-kW, 650-l/h cryogenic plant which has accumulated 13,400 hours of operation, nine flexible superconducting bus lines measuring 497 m

in length and an 850-t capacity cryogenic structure support system. Notable magnet system highlights during the experimental campaign include:

- achievement of 2.91-T central field at a radius of 3.6 m by employing a current grading method in the three conductor blocks of the helical coils,
- stable excitation of the helical coils to 11.45 kA following the occurrence of a normal propagation event,
- flawless excitation of all six poloidal field coils,
- normal insulation values and the absence of resistive voltages in all joints, and
- achievement of cool-down times of 23 days.

Very long discharges (> 70 minutes) sustained by lower hybrid current drive have been achieved on TRIAM-1M, a 0.84-m superconducting tokamak equipped with molybdenum limiters and all-metal plasma facing components for particle recycling and wall pumping studies. An interesting phenomenon has been observed in ultra long discharges (>70 minutes) in which the wall repeats a process of being saturated and refreshed implying recycle coefficients that temporally exceed unity. In high power, high density discharges, wall saturation is observed in about 40 s leading to a loss of density control and discharge termination—discharge termination occurs more rapidly with increasing density.

KSTAR is a medium scale ( $R = 1.8$  m,  $a = 0.5$  m) steady-state advanced tokamak ( $\epsilon_x=2.0$ ,  $\epsilon_y=0.8$ ) under construction in South Korea that features, like ITER, many technological advances including:

- a fully superconducting magnet system (TF and segmented central solenoid using a CICC with advanced Nb<sub>3</sub>Sn superconductor and Incoloy 908 conduit),
- actively cooled in-vessel components,
- long-pulse current-drive and heating systems for non-inductive current drive, current density profile control and control of the plasma pressure profiles, and
- a full complement of plasma diagnostics.

The mission of the project is to develop a steady-state-capable advanced superconducting tokamak to establish a scientific and technological basis for an attractive fusion reactor. The baseline operation is a 20-s pulse driven by the PF magnet system. Extending the pulse length to 300 s and increasing the plasma performance is the objective of an upgrade to an actively cooled divertor and an increase in heating and current drive power from 16 MW to 26 MW.

A full range of development activities has been undertaken by the project such as a full-scale double-wall vacuum vessel, a new superconducting strand and conductor, a central solenoid model coil, a 300-s ion source and a 6-MW double resonant loop prototype fast wave current drive antenna. Design of the experimental facility is complete and construction is ongoing with completion expected in 2004.

FTU-D is a proposed modification to extend FTU operation to strongly shaped plasmas ( $R/a = 1$  m/0.2 m,  $\epsilon_x = 1.6$ ,  $\epsilon_y = 0.8$ ,  $B = 2.5$  to 5 T) utilizing as much as possible the existing FTU tokamak device and facility infrastructure. It would provide a high-field, high-aspect ratio variant of the advanced tokamak for investigation of operation at high  $\beta_N$ , high bootstrap current fraction (at least 60%) in high energy confinement regimes obtained by current and pressure profile

control. Shaping would be affected by unbalancing currents in the windings of the air core transformer thereby allowing for single and double null highly shaped plasmas within the existing circular vacuum vessel. The primary technology upgrades to affect the modification include upper and lower TZM toroidal limiters with 12-MW/m<sup>2</sup> capability, a new central solenoid and power supply amplifier and feedback system for position control.

## 4.2 Steady-State Technologies

In addition to the many technology contributions discussed above, several other developments were reported, most notably in the area of novel steady-state heat removal technologies:

- A dynamic ergodic divertor is planned for TEXTOR. This major new installation employs a rotating perturbation on the magnetic field at variable frequency (from DC up to 10 kHz) that smears local peaks in the heat flux and promotes differential plasma rotation.
- Initial tokamak experiments with free surface and capillary-based liquid lithium plasma facing components have begun on the CDX-U and T-11M tokamaks. These experiments are taking the first steps to investigate the potential of liquid metal surfaces to solve the divertor and limiter high heat load problem.
- The operating window has been determined for “liquid walls” using the UEDGE edge plasma code. The initial conclusion is that the evaporation rate is too high for devices such as tokamaks but could be acceptable for magnetic fusion energy concepts that feature short connection lengths and high-density operation, such as the FRC.
- The operating window for a falling pebble divertor has been determined. Calculations and low energy neutral beam implantation experiments indicate that a system can be realized that meets ITER divertor heat flux and pumping targets.

## 5. Compact Cu Burning Plasma Experiments and Neutron Sources

The Fusion Ignition Research Experiment (FIRE) is a design study currently underway in the U.S. to assess near term opportunities for advancing the understanding of self-heated fusion plasmas. The emphasis is on understanding the behavior of fusion plasmas dominated by alpha heating ( $Q > 5$ ) that are sustained for durations comparable to the plasma time scales ( $> 20 \tau_E, \sim \tau_{skin}$ ). The nominal design parameters for the experiment ( $R/a = 2 \text{ m}/0.52 \text{ m}$ ,  $q_5 = 1.8$ ,  $q_5 = 0.4$ ,  $B_t = 10 \text{ T}$ ,  $I_p = 6.44 \text{ MA}$ ) have been chosen in accordance with the philosophy to provide access to the alpha-heating-dominated regime using the present advanced tokamak database while maintaining the flexibility to study other AT modes (e.g., reversed shear, pellet enhanced performance) at lower magnetic fields and fusion power for longer durations.

IGNITOR, like FIRE, is a compact ( $R/a = 1.32 \text{ m}/0.47 \text{ m}$ ) high field copper machine; but, it is being provided with higher field (13 T) and current (11MA) capability in accordance with its more ambitious ignition mission. The results of confinement simulations reported at the Conference show that control of sawtoothing and the density profile are critical elements for achieving the ignition goal. The study further concludes that the incorporation of a 10-MW to 20-MW ICRF heating system and a high-speed pellet injector in the project is essential for providing the flexibility to counter unexpected, adverse plasma behavior.

The International Fusion Materials Irradiation Facility (IFMIF) is a proposed accelerator based intense D-Li neutron source for developing and qualifying radiation resistant, low-activation materials under fusion-relevant conditions (neutron spectra and fluences). Last year a review of the Conceptual Design Activity (CDA) of IFMIF was undertaken to identify potential cost reductions without changing its performance mission  $2\text{-MW/m}^2$  neutron fluence in a volume of  $0.5\text{ cm}^3$ . To accomplish this goal, the accelerator upgrade was eliminated and the deployment was staged to reduce capital cost from \$797M to \$488M. The stages of deployment and their respective goals are:

- one accelerator @ 50-mA operation for ITER tritium breeding blanket materials selection, fusion-fission data correlation and generic damage studies, and
- 125-mA ( $1\text{ MW/m}^2$ ) and 250-mA ( $2\text{ MW/m}^2$ ) stages for alloy development at DEMO relevant fluences and irradiation up to 100 to 200 dpa for high-performance materials for fusion reactors.

The estimated cost of the first phase is \$301M making it financially much more attractive than the previous version. The project has entered a three-year Key Element Technology Phase that will focus development on CW deuterium beam operation, Li target handling and test cell technologies in order to reduce the key technology risk factors.

## **6. Advanced Materials Research**

Nicely complementing the achievements reported above and the reactor and socio-economic studies to be discussed later is the impressive progress made by the materials science program in the pursuit of the structural materials that will be required for the fusion energy mission. The development of radiation resistant, low activity and low decay heat materials is key to realizing the potential safety and environmental advantages of fusion. In this respect, improvements have been made on pre- and post-irradiation properties of the three principal materials under investigation internationally—reduced activation ferritic-martensitic steels (RAF), Sic-SiC ceramic composites and vanadium alloys.

### **6.1 Reduced Activation Ferritic Steels**

Of these three classes of materials, the RAFs have achieved the most technological maturity. Response to neutron irradiation in fission reactors (under the extensive US/Japan JUPITER collaborative research program) of 9Cr-2W RAFs have proven them to be more resistant to irradiation hardening and helium embrittlement than other candidate materials. The radiation induced shift in the ductile to brittle temperature transition (DBTT) of RAFs has been found to be about a factor of four smaller than conventional 9Cr-1Mo steels. Reducing the Cr content below 9% has proven effective in reducing the DBTT shift. Families of this material have now been produced that exhibit modest shifts in DBTT after irradiation and irradiation embrittlement appears to saturate above 10 dpa indicating that this may not be a critical issue in fusion applications. Similarly, the effects of helium transmutation have been assessed in helium implantation experiments. At concentrations up to 580 ppm, no significant enhancement occurs in the DBTT confirming the high resistance of RAFs to helium embrittlement due to the high trapping capacity of its martensitic structure for helium atoms.

Finally, the introduction of oxide dispersion strengthened (ODS) steels promises significant improvements in the high temperature dynamic strength properties of RAFs. In particular, the new ODS steels developed by the Japan Nuclear Cycle Development Institute show dramatic improvement in the unirradiated high temperature creep strength in comparison to the ferritic steel F82H.

## **6.2 Vanadium Alloys**

Significant strides have also been made in the development of the technologies for large-scale manufacturing of high-purity V-4Cr-4Ti alloy by focusing on improving present commercial production processes of vanadium metal, and optimizing alloying, plating, sheeting and wiring techniques. Particular emphasis has been placed on reducing carbon, nitrogen and oxygen impurities, which are known to deteriorate workability, weldability and the radiation resistance of vanadium alloys. Medium (30kg) and large size (166 kg total) high-purity V-4Cr-4Ti ingots designated as NIFS-HEAT-1, 2(A) and 2(B) have recently been produced demonstrating the technology for fabricating large V-4Cr-4Ti alloy pieces with < 100 ppm C, ~ 100 ppm N and 100 to 200 ppm O concentrations.

Hydrogen embrittlement, which is expected to occur when the concentration of hydrogen exceeds 400 to 1000 wppm, has also been addressed by exposing samples for nine months in the neutrals rich (0.1 to 0.2 Pa) environment behind the divertor baffles of the JFT-2M tokamak. Hydrogen bulk concentrations of only 1.3 wppm were measured. This surprisingly low value is attributed to the formation of 200- to 400-nm oxide film diffusion barriers at the surface.

## **6.3 Silicon Carbide Composites**

Impressive gains have also been made recently in improving the baseline and irradiated properties (strength, fracture toughness and thermal conductivity) of SiC/SiC composite materials. These improvements have resulted from modification and optimization of processing techniques for reinforcement fibers, matrix materials and fiber-matrix interfaces that have produced composites of higher purity and crystallinity, lower porosity SiC matrices, and improved fiber-matrix interfaces. Recent results on mechanical properties of SiC/SiC under neutron irradiation are extremely positive—composites with the new Hi-Nicalon Type-S SiC fibers exhibit no mechanical property degradation up to 10 dpa. These improvements have led to significantly widened design windows for the fusion energy application and further gains are expected as new near-stoichiometric and crystalline SiC fibers are integrated into composite structures.

## **7. Power Plant Design and Economic Forecasts**

Power plant studies are exploiting the improved performance and technology themes and this in turn is leading to an improved vision for fusion energy. Several papers presented at the meeting dealt with various aspects of the prospects for fusion energy including systems code studies that identify major technological and physics cost drivers, detailed point designs of tokamak, stellarator and fusion/fission hybrid reactors, and market analyses and socioeconomic studies that

identify the characteristics that can make fusion energy competitive with other energy technologies in future energy markets.

## 7.1 Tokamaks

In addition to the advanced physics and technology themes, the trend towards more compact systems was apparent in both tokamak and stellarator reactor concept designs. Two compact advanced tokamak designs were presented, the ARIES-AT ( $R = 5.4$  m,  $B = 5.6$  T,  $I_p = 13$  MA) and the A-SSTR-2 ( $R = 6.2$  m,  $B = 11$  T,  $I_p = 12$  MA). Both share common technology innovations such as high temperature superconducting (HTS) TF coils, SiC/SiC blankets, and high-temperature (i.e. high thermodynamic efficiency) Brayton energy conversion cycles. They differ in the degree to which advanced physics is stressed relative to the technology. The ARIES-AT philosophy exploits advanced physics with  $n = 5.6$  to 8 and a 90% bootstrap current fraction. High- $n$  operation opens the possibility for the use of low-field, NbTi magnet technology. The A-SSTR-2 has somewhat less aggressive physics goals ( $n < 4$ , bootstrap current fraction = 0.83) but relies more heavily on high field operation (23-T coil peak field on the new Bi-Ag alloy HTS conductor). Both studies have in common the objective of reducing the capital cost of fusion power plants. The A-SSTR-2 targets a normalized (relative to present day coal fired plants without CO<sub>2</sub> controls) cost of electricity (COE) of 1.5 for introduction into the market in 50 years. ARIES-AT projects a COE of 50 m\$/kWh.

A study of the Spherical Tokamak (ST) as a fusion/fission hybrid candidate for nearer-term applications such as transmutation of high level nuclear waste and breeding of fissile fuel is underway in China. These studies conclude that a 100-MW fusion power “driver” operating at a neutron wall loading of 1 MW/m<sup>2</sup> can transmute the high level waste generated by 10 pressurized water reactors, be self sustaining and reduce the biological hazard potential of the entire system by two orders of magnitude in 30 years.

## 7.2 Stellarators

Stellarator reactor design and systems code studies are also addressing more compact (lower aspect ratio,  $A$ , and fewer field period,  $M$ ) variants of traditional stellarator magnetic configurations. Helias is a  $R = 18$  m,  $M = 4$ , Wendelstein 7-X based reactor design that is more compact than the previous W 7-X extrapolation (the  $M = 5$ ,  $R = 22$  m HSR). Its magnetic field structure has been optimized for equilibrium and low transport (neoclassical losses). The magnetic field spectrum has been chosen so as to produce a true minimum-B configuration at the operating plasma pressure thereby providing good confinement of fast alpha particles (2.5% loss fraction). The  $\beta$  limit is estimated at 4.3% and ignition is predicted with empirical scaling—there is no need for confinement enhancements. These attractive physics characteristics lead to a number of technological advantages including the possible use of NbTi magnets and, for a fusion power of 3,000 MW, an average neutron wall loading of only 1.2 MW/m<sup>2</sup> that corresponds to a nine-year first wall projected lifetime. A cost reduction of the reactor core of 20% is expected as a result of its more compact dimensions.

The effect of recent experimental results on the characteristics of LHD-type helical reactors has also been examined through systems studies codes. Plasma projections based on the 2 × higher

energy confinement time, the high  $\beta$  (2.4 %) achieved in inward shifted configurations and the  $1.5 \times$  higher density boundary observed on LHD have been used to evaluate the COE for an  $A = 10$ ,  $M = 14$  design and a more compact  $A = 5$ ,  $M = 8$  design. A 25% reduction in COE is realized in the more compact design. Finally, systems studies codes have been used to provide a first assessment of the  $M = 2, 3$  ultra-low aspect ratio ( $A = 3$  to 4.5) quasi-axisymmetric and quasi-poloidal stellarator-tokamak hybrids that are currently the focus of the U.S. compact stellarator proof-of-principle program. The studies point to possible stellarator reactors with major radii as small as 7 m and neutron wall loading in the 4-MW/m<sup>2</sup> range.

### 7.3 Economic Forecasts

We conclude this section with a brief summary of the results of two studies, one specific to Western Europe and the other global in scope, on the future potential of fusion energy. These socioeconomic studies take into account many variables such as regional energy demand, prevailing economic conditions, the social and environmental costs of energy technologies, regulatory restrictions on CO<sub>2</sub> emissions and the characteristics of fusion (high capital cost, widespread availability, long term supply, attractive environmental and safety features) and its market introduction date, deployment pace and cost reduction factors to project electricity market share in this century. The absence of CO<sub>2</sub> emissions and the low external costs (i.e. the impact on human health and environment) attributable to fusion compensate for its high capital costs (66 m\$/kWh COE assumed in one of the studies for first generation power plants) in the comparison with fossil fuels, nuclear fission and other renewable energy sources. Both studies conclude that under restricted greenhouse gas emission scenarios, fusion can capture a significant electrical power market share by the end of the century—up to 30% globally for a CO<sub>2</sub> concentration constrained at 550 ppm.

## 8. Conclusions

While there were numerous important new technical results reported at the meeting, the following brief list of achievements captures not only the highlights but may also foretell possible positive future trends and prospects for the entire field of research:

- The rapid progress made during the ITER EDA extension on the completion of the new ITER-FEAT design and its physics and technology R&D validation can only be described as remarkable. The success of the joint activities demonstrates the feasibility of the design and the project's prospects for meeting its programmatic objectives.
- Steady-state operation with advanced performance and the recognition of the role of enabling technologies in achieving this goal is an emerging theme in fusion energy development.
- Advanced materials featuring environmental attractiveness and improved performance are becoming realistic for fusion energy applications.
- A broad range of options for tokamak and stellarator fusion reactors are under investigation—concepts and near-term applications are being explored that advance both the physics and technology fronts towards lower cost of electricity and improved environmental attractiveness.
- The attractive features of fusion are being recognized and its potential as a competitive energy source is being quantified by socioeconomic studies.