Attachment A1

The Status and Plans of Four Parties

<JT-60>

After the shutdown of JT-60U in 2008, the activities and the structure of JT-60 team were substantially shifted towards modification to the superconducting device, JT-60SA, while the team also continued physics studies and plasma evaluations for JT-60SA, ITER and Demo based on existing JT-60U data. Objective of JT-60SA programme being promoted as one of joint programmes under Broader Approach (BA) agreement between Japan and EU and also as a domestic core programme of tokamak development in Japan, is to contribute to ITER Project and also to technical preparations for the decision of DEMO construction, with enhanced performance in plasma duration, plasma shaping control, heat exhaust and particle control, stability control, and heating & current drive capabilities. The first plasma in the end of FY2015 is planned. Procurements of JT-60SA components are shared by Japan and EU. Existing infra-structure and equipments for JT-60U, such as heating & current drive systems, cooling facilities, power supplies, diagnostics etc., will be utilized as many as possible.

Seven procurement arrangements for Japanese contributions were launched between the Implementing Agencies, JAEA and F4E, for the supply of PF (poloidal field) magnet conductor, PF coil manufacture buildings, PF coil manufacturing, vacuum vessel, buildings for vacuum vessel sector-assembly, materials for in-vessel components, and Divertor Components by 2009. The coil manufacture building and PF conductor manufacturing building were completed in Naka Fusion Institute in March 2009, and the superconductor manufacturing machines were installed in the building in September 2009. Two conductors using the copper dummy cables were manufactured in Naka Institute as a trial, and these met the specifications and successfully passed the helium leak test. In March 2010, the first two EF-H (equilibrium field at the high field side) superconducting conductors of 450 m were manufactured successfully. Manufacturing of the first 20 degree sector for the vacuum vessel is going on as scheduled. As for the European contributions to the procurements, the procurement arrangements for Quench Protection Circuit, High Temperature Superconducting (HTS) current leads and Cryostat-base were signed in FY2009.

Removing components in the JT-60 assembly hall, such as the high-voltage bushing of the Negative Neutral Beam Injector (N-NBI), support structures for maintaining the Negative Ion Source and the high-voltage table (HVT) of N-NBI, and the shielding wall (15m width, 16m height, 0.35m thickness) between the JT-60 machine hall and the JT-60 assembly hall were started from November in 2009, and was completed in March 2010. Disassembling of components in the JT-60 machine hall started in this April and will be completed in 2012.

The development of gyrotron with a new improved mode converter was started aiming at 1MW, 100s which is the long-pulse requirement of JT-60SA ECRF system. It was confirmed that the RF diffraction loss was remarkably decreased and the cooling water temperature for the DC break, which had limited the pulse length, was saturated at about half (ΔT~30°C) of that before the improvement. And the pulse length was extended up to 17 s (1 MW) in December 2009. Further conditioning is continued for extending pulse length further. Using another gyrotron, a new operation technique of Active-Anode-Voltage-Control was developed and the pulse length at 1.5 MW was extended from 1 s (achieved in 2007 as a world record) to 4 s. A new linear motion (LM) antenna was designed which has an advantage of feeding cooling water reliably to the linearly driven mirrors, compared to the conventional antenna having rotatable mirrors. The mock-up antenna showed satisfactorily wide steering range (~80°) and sufficiently small beam radius (~100 mm) at EC resonance layer for JT-60SA.

Hydrogen negative ion beams of 490keV, 3A and 510 keV, 1A have been successfully produced in the JT-60 negative ion source with three acceleration stages. These successful productions, the first acceleration of the H- ions up to 500 keV at high-current of > 1 A, have been achieved by overcoming the most critical issue, i.e., a poor voltage holding of the large-sized negative ion sources with the grids of ~2 m^2, which were required for JT-60SA and ITER. To improve voltage holding capability, the breakdown voltages for the large grids was examined for the first time, and it was found that a required vacuum insulation distance for the large grids was 6-7 times longer than that for small-area grids (0.02 m^2). Based on the result, the gap lengths between the grids were tuned in the JT-60 negative ion source. The modification of the ion source realized significant stabilization of voltage holding and also reducing the conditioning time.

<KSTAR>

After the successful first plasma generation in the middle of 2008, several upgrades had been made in the power supply, heating, wall-conditioning, and diagnostic systems of KSTAR device. The cool-down for the 2nd operation campaign was then started from September in 2009 and plasma experiments carried out from October to November. During this 2nd campaign, the machine was operated with the toroidal magnetic field of up to 3T. Circular plasmas with current of 300 kA and pulse length of 2 seconds have been achieved with limited capacity of PF magnet power supplies. To test the operational limit, the toroidal field coils were operated up to 36 kA as the designed operation current was 35 kA, corresponding to the toroidal field of 3.5 Tesla at the axis. The second harmonic pre-ionization with 110 GHz, 250 kW gyrotron at 2 T has been studied. Various parameters such as injection angle, position and pressure have been scanned to optimize the pre-ionization. The ICRF Wall Conditioning (ICWC) was routinely applied during the shot to shot interval. The effect of ICWC has been quantitatively assessed by dedicated diagnostic systems. For the study of in-vessel dust characterization, duct collectors have been installed and coupons have been installed to study the campaign-integrated deposition characteristics.

There are several engineering and physics issues studied during the 2nd campaign. One of main emphases was put on the understanding of magnetics since the jacket of the cable-in-conduit cable (CICC) of the superconducting magnets are made of incoloy 908, which is a slightly magnetic material with the permeability of about 10. A numerical model including the effect of magnetic materials has been developed and the result has been verified by measuring the effect of residual magnetic field by using electron beam and the Hall probe array system. From the accurate measurement of the stray field, it was also found that the KSTAR PF-coil systems are quite up-down symmetric with a negligible installation error. This in turn indicated that the downward shift of plasma column observed during the start-up phase is not from the static field, but from the dynamic field, probably coming from the up-down asymmetric eddy current induced in KSTAR cryostat system. Initial experiments have been carried out of ohmic start-up, MHD phenomena which include sawtooth, locked-mode, disruption etc, and plasma heating by ECH and ICRF in circular ohmic plasmas. Theoretical studies have been also carried out for the 1st and 2nd harmonic ECH pre-ionization mechanisms, and the toroidal current observed during the ECH pre-ionization with a finite vertical stray field.

After finishing the 2nd operation campaign, a significant upgrade of the KSTAR device is being made. To achieve D-shape, diverted plasmas, all of the plasma facing components (PFCs) including divertor will be installed inside the vacuum vessel. The PFCs will be covered with actively-cooled graphite tile until 2012, and then upgraded to be covered with carbon-fiber-composite (CFC) for long-pulse operation. The sixteen-segmented in-vessel control coils (IVCCs) will be installed prior to the installation of the PFCs. The IVCCs will be externally connected to form two sets of circular coils for vertical and radial position control, and then the IVCCs will be additionally connected to form twelve picture-framed coils for RWM/FEC control later. The first NBI system (NBI-1), designed to deliver 8 MW D0 neutral beams into the KSTAR plasmas with three ion sources, is now under fabrication and will be commissioned for the 1 MW beam power with the first ion source during the 3rd campaign. It is now scheduled that the cool-down for the 3rd campaign will start from July, so plasma experiments expected from September.

<U.S.>

DIII-D Highlights and Plans

The 2009/10 research program consisted of 16 weeks of physics operations in 2009 and 17 weeks in 2010. Research in support of ITER included experiments simulating the effect of Test Blanket Module (TBM) magnetic field perturbations, disruption mitigation, performance extrapolation and scenario development (startup, rampdown, and helium operation), ELM control and ELM-free operation, and hydrogenic (tritium) retention. An International Team of ITER scientists used a scaled mockup of a single ITER TBM pair inserted in a DIII-D port to assess the effects of TBM ripple on H-mode threshold, plasma performance, ELM suppression, mode locking and fast ion transport: the effect on plasma performance is relatively small with the largest effect being ~20% reduction in plasma rotation. Progress on disruption mitigation included demonstration of active runaway electron control (position and current), diagnosis of their energy distribution, and evaluation of mitigation new mitigation techniques using large shattered D_2 pellets, resonant magnetic perturbations (RMP) and large shell pellets. Shattered pellet injection (SPI) has achieved the highest local electron density for achieving collisional suppression of runaway electrons. Joint experiments in JET and DIII-D extended ρ* transport scaling to the hybrid plasma regime, while an H-mode pedestal similarity experiment showed very little, if any, ρ* scaling of pedestal width, consistent with predictions from the EPED1 code. A broad suite of new multi-field, multi-scale-length fluctuation diagnostics has provided detailed measurements which are being compared with gyrokinetic transport simulations and models for fast-ion stability and transport. An experimental campaign using high purity (>95%) helium plasmas revealed that the L-H power threshold is between 30-50% greater than that for deuterium plasmas. Electron cyclotron pre-ionization for low voltage startup was demonstrated in both deuterium and helium plasmas. ELM-free QH-mode plasma operation was sustained with zero net NBI torque using an $n=3$ non-resonant magnetic field perturbation to maintain the required edge rotational shear. Hydrogenic retention experiments showed that wall uptake occurs during startup/rampup and is very low during H-mode, a promising result for ITER. DIII-D also tested an oxygen bake as a method for removing tritium-containing carbon co-deposits. DIII-D is now beginning a one year shutdown to modify one neutral beam line to allow for variable off-axis injection; install additional n=3 magnetic perturbation coils on the centerpost; and complete a number of diagnostic and other system improvements. These upgrades will increase capabilities for current profile control experiments, energy and momentum transport studies in the core and pedestal regions, research on ELM-control physics, and exploration of 3D field effects on plasma stability and transport. Improved physics understanding and validation of predictive simulations remain a major emphasis of the DIII-D research program.

Alcator C-Mod

Alcator C-Mod has completed nearly 15 out of our planned 18 weeks of research operation in FY2010 (October 2009 - September 2010). We have significantly extended the I-mode regime to high power and plasma performance. I-mode yields strong edge ion and electron temperature barriers, excellent energy confinement ($H_{ITER\text{-}98}$ up to 1.2), and low collisionality. The I-mode regime has no need for ELMs to maintain density and impurity control. Experiments to simulate ITER-like plasma evolution during startup and rampdown have been carried out on C-Mod using the ITER shape and magnetic field, with comparable safety factor, normalized pressure and energy confinement scaling. During ramp-up, with early divert times and transitions to H-mode, significant loop-voltage savings are realized, as predicted for ITER from TSC simulations. Detailed studies of ICRF-induced flow drive on C-Mod reveal that the efficiency depends strongly on He^3 concentration in the $D(He^3)$ mode conversion regime, with driven core toroidal rotation up to 110 km/s (M~0.3). Experimental and theoretical studies of intrinsic rotation show that central toroidal rotation, observed in the absence of external momentum input, scales with edge temperature gradient, and the relationship to fluctuation-induced residual stress is under investigation. For line average electron densities above $1x10^{20}m^{-3}$ LHCD efficiency drops off more rapidly than expected theoretically, and mechanisms of anomalous absorption in the edge plasma are under investigation. A new, advanced Lower Hybrid launcher, aimed at low-loss and high power density (~100 MW/m^2) is currently being installed, and first results are expected this summer. Lower Hybrid waves have been used to produce a seed population of non-thermal electrons (E>100 keV), which can be accelerated during the thermal quench (TQ) phase of disruptions up to ~20 MeV. Modeling using the 3-D NIMROD code shows that, in these conditions, when massive gas puffing is applied for disruption mitigation, the strong MHD activity which grows during the TQ causes a nearly complete stochasticization of the magnetic field, in turn causing loss of the runaway electrons during the TQ. As part of the joint research milestone to characterize power flows in the scrape-off-layer, we have installed several new advanced diagnostics, including IR imaging, probes, and fast thermocouples. Experiments in this area are being closely coordinated with the DIII-D and NSTX facilities, as well as with the US theory/modeling community.

NSTX

NSTX has completed approximately 3 weeks of operation in its FY10 run campaign after finishing the FY09 campaign in July 2009. During the FY09 campaign, there was great progress in integrating elements of plasma control and wall conditioning to produce long-pulse, wall stabilized plasmas. In these experiments, lithium coating of the plasma facing components was used to reduce the density early in the discharge and to minimize ELMs. High-poloidal beta, κ~2.6 discharges produced a high-fraction of non-inductive current (65%) that was sustained for 800 to 900 ms, corresponding to 2.5 to 3 current redistribution times. Discharges with normalized betas of up to 6 %-m-T/MA and toroidal betas up to 25% were sustained for three energy confinement times. Both high-beta scenarios could enable a CTF to achieve high neutron wall loading to fulfill its mission more effectively. Research pertaining to the Joint Research Target focused on comparing deuterium retention in Ohmic and auxiliary-heated discharges with and without lithium, indicating between 87-94% retention immediately after a shot in all these cases. The deuterium was released on time scales ranging from seconds to weeks or longer. Surface sample analysis indicated that the increasing retention with increasing lithium coverage was due to changes in surface chemistry, and that the prompt recovery even with lithium coatings applied indicated that the deuterium was only loosely bound to the lithium coated graphite. The application of 3D magnetic fields to the plasma was used to trigger ELMs in a controlled fashion in order to reduce impurity accumulation while still maintaining high core confinement and beta in ELM-free discharges. Extensive conditioning of the divertor plates prior to Co-Axial Helicity Injection experiments, along with field-nulling coils in the upper divertor to delay or suppress arcs, led to increases in CHI-generated current to nearly 200 kA at the time of handoff from CHI to induction. This translated into a flux savings of nearly 180 kA of current, which is 25% of the current flattop of 700 kA in a typical long-pulse scenario in NSTX. Recent research on Resistive Wall Modes have predicted the importance of kinetic effects in determining the plasma rotation required to stabilize the mode, and showed that the mode stability is not a simple function of rotation. Experiments indeed showed that higher fast ion content was stabilizing, consistent with the theoretical predictions. “Snowflake” divertor configurations were developed and exhibited a reduction in peak heat flux to the divertor while sustaining high confinement and beta in the plasma core. Experiments to understand the L-H transition better were performed in support of ITER high priorities. It was found that helium plasmas had threshold powers approximately 20 to 30 % greater than those for comparable deuterium plasmas, and application of the 3D fields prior to the transition caused a significant increase in threshold power. Dependences with plasma current and plasma triangularity were also found, and these could be understood within the framework of the neoclassical behavior of thermal ions, and resulting E_r wells due to their losses. A Liquid Lithium Divertor for increased particle control and production of discharges at lower collisionality, was installed between the FY09 and FY10 runs, and a focus of the FY10 campaign, which has just started, will be to assess the LLD and related plasma performance. Further, a BES diagnostic was installed during this period, which, along with the high-k scattering diagnostic, will give extended turbulence k-range coverage. This will enable more comprehensive investigations of mechanisms governing energy, particle and momentum transport.

<EFDA-JET>

The last 12 months on JET have been a period of intense operations (Campaigns C26-C27 from 12 January 2009 to 23rd October 2009; 118 S/T days in two shift operation) with very good machine performance, especially the reliability of the NB injection systems at high power. Other new hardware systems tested and exploited included a high frequency pellet injector, a disruption mitigation valve (DMV), two TAE antenna arrays and several enhanced diagnostic systems. During an intervention (8 April 2009 to 21 June 2009), systems for plasma control and Resonant Magnetic Perturbations were upgraded, respectively, for higher resilience against, and extended control capabilities for, ELMs. On 26 October 2009, the Shutdown started for the installation of enhancements of high scientific value and strategic importance (ITER-like combination of first wall materials (ILW with tungsten divertor and beryllium wall), NB Power Upgrade (30MW long pulse rather than 20MW short pulse to facilitate scenario development at high current, high β and high density); upgraded and new diagnostics; and a series of machine refurbishments). In addition, significant Fusion Technology Tasks were carried out.

The scientific programme contributed to the design of the ITER ICRH system, demonstrating ELM tolerance for all ICRH systems (ITER-like antenna ILA), External Conjugate-T, 3dB couplers), operation at high voltage on antenna strap (42kV), high power density (6.2MW/m^2) and novel arc detection systems with the ILA, input for modelling (TOPICA), verifying the suitability of fundamental hydrogen heating in hydrogen plasmas and second harmonic ^{3}He in hydrogen plasmas for the non-active phase of ITER, and assessing Ion Cyclotron Wall Cleaning under ITER-relevant conditions. Simulations of the ITER baseline scenario from start to finish (D_2 and ^{4}He) contributed to design modifications to the ITER coil set and divertor. The direct effect of fast ions on the sawtooth stability was demonstrated, indicating encouraging control methods for ITER. Detrimental TF ripple effects in H-mode at low edge ν^* were quantified, showing adequate confinement in ITER requires <0.5%, preferably 0.2%-0.3%.

A wide range of experiments were carried out to qualify the ITER baseline and advanced scenarios in deuterium, helium and hydrogen and provide an improved basis for extrapolation to ITER. Stationary ELMy H-modes were developed up to 4.5MA at 3.4T, showing H_{98}~0.9 for input powers up to 2.2 times the L-H threshold power. Plasma performance was found to be independent of heating mix (NB/ICRF) and temperature profiles were similar, despite a strong reduction (5-10) of toroidal rotation and dominant electron heating with ICRH.

Qualification of the ITER Advanced Scenario showed all ITER normalised conditions relevant for steady state operation (β_N=2.7, H_{98}=1.2-1.3, T_e=T_i, f_{bs}~0.4, f_{GW}~0.7, but not ρ^*) could be achieved. Furthermore, Hybrid scenario were matched to AUG/DIII-D and extended to H_{98}~1.4 and ρ^*=0.005 at high triangularity and a toroidal field of 2.4T. Detailed analysis suggests a link between rotational shear and reduced ion stiffness.

A dedicated helium campaign, including current ramp-up and ramp-down scenarios for the start-up of ITER, concluded that helium operation will provide a robust test of the experimental space available in ITER. The H-mode power threshold is similar to that in deuterium, thus providing the possibility for H-mode studies in ITER in helium.

ELM mitigation techniques (RMP, kicks, pellets) were shown to increase the ELM frequency by a factor of 3-5, but not yet to the level required for ITER. Complete ELM suppression was not observed. Detailed IR observations show a modest reduction of 30-40% in ELM associated peak heat loads on the outer divertor. ELMs are triggered only when the pellet is significantly large (and fast) to penetrate to the top of the pedestal.

An important part of the programme was devoted to preparatory experiments for the ILW, including the completion of the characterisation of the carbon wall. hydrogen retention in carbon plasma facing components, carbon source strength and carbon migration towards the divertor were determined. Experiments using extrinsic impurity seeding (neon or nitrogen) show a reduction of the inter-ELM power loading to ~1MWm^{-2} for only ~10% loss in energy confinement. Disruption studies using the DMV concentrated on reducing transient heat loads (50% heat load reduction during thermal quench) and techniques for the suppression run-away electron production.

The Experimental Campaigns of 2009 were characterised by a strong involvement from the European Associations (294 people from 22 countries; 50 ppy total; 42 working days per person on average) which was complemented by strong International (non-EU) collaborations (with the US, the Russian Federation, Japan and Korea; 34 scientists; 3ppy or 6 percent of staffing in 2009). 39 ITPA ITER high priority coordinated experiments were covered, requiring 60 percent of overall run-time. An Agreement for Cooperation between EURATOM and Brazil was signed on 27 November 2009; a visiting researcher from Brazil is already evaluating at JET the TAE system.

The forward plan for JET (“Reference Scenario”) covers the exploitation of the ILW up to 2014, followed by DT experiments up to the end of 2015. Two Task Forces have been established for the Experimental Campaigns of 2011 and a General Planning Meeting on 1-5 March 2010 started the process of experiment elaboration which will culminate in a Second planning meeting on 15-19 November 2010. From late April/early May 2011, the experimental programme will begin to focus on the characterisation of the ILW, together with exploration of ITER operating scenarios with the ILW and physics issues essential to the efficient exploitation of the ILW and ITER. Critical issues will be the minimisation of T-retention, material erosion and migration, mixed material effects, melt-layer behavior, impurity control, and the development of ITER scenarios fully compatible with a Be/W material mix. A major challenge will be to accommodate ~40MW of heating power with the ITER-like combination of first wall and divertor materials.

For the longer-term JET programme (“Alternate Scenario”), two feasibility studies, launched in 2009, have quantified the resources needed for an ECRH system (~10MW) for heating and current profile control (collaboration with Russian Federation) and a system of Resonant Magnetic Perturbation (RMP) coils for ELM control (collaboration with US). These systems would allow the full exploitation of the ITER-like Wall and provide JET with a significantly increased capability for preparing ITER operations. This Alternative Scenario requires a significant increase in the level of contributions by international partners who would then become involved more actively in the decision-making, including the joint revision of the JET plan of exploitation.

Since 2000, 152 JET FT tasks have been launched (allocated resources ~23M€ (~2.6M€ in 2009)), concentrating on tritium in tokamaks, tritium process and waste management, plasma facing components, engineering, and neutronics and safety.