# Summary of the 21st Toki Conference on

# Integration of Fusion Science and Technology for Steady State Operation

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**Abstract:** *This summary is based on 240 papers presented at the 21st Toki conference held on November 28th to December 1st 2011. It registered a record level of attendance and dealt with a wide range of fusion science and technology aspects for steady state operation which constitutes the greatest challenge on the way to the production of fusion energy on an industrial scale. Prominent aspects reported include the physics scenarios for SSO (Steady State Operation) which has to integrate in a coherent way challenging technologies. These include the technology of materials for structural and first wall components, superconductors for magnetic confinement, plasma heating systems. A considerable effort is being made to integrate the knowledge acquired in these fields into large experiments. The construction and operation of a new generation of fusion machines based on superconducting magnets has made good progress and long pulses operation demonstrate the progress made on SSO in an integrated way. They now reach the GJ level in extracted energy at plasma parameters significantly more relevant than previously. However, the ITER goal of 500 GJ is still far away and in view of the distance to travel, the level of activity in this area of research is expected to grow considerably in the years to come. The projection of present day knowledge to a DEMO fusion reactor is now being addressed by newly formed research groups.*

**Keywords:** Fusion energy, magnetic confinement, steady state operation, tokamaks, helical systems, ITER, heating, current drive, stability, plasma wall interactions, materials, superconductors, divertors, limiters.

## 1. Introduction

The 21st International Toki Conference (ITC 21) was held between 28 November and 1st December 2011 in Toki Japan. Its scientific organisation had chosen a general scheme addressing the "Integration of Fusion Science and Technology for Steady State Operation" which is now considered to be an essential issue in the critical path for fusion development. This is indeed a major stated objective for a new generation of fusion machines using superconducting technology which gives the inherent potential for full steady state operation (SSO). The conference attracted a record number of participants (284) coming from 14 countries. All SSO issues were addressed, the presentations dealing with topics ranging very broadly from confinement physics to enabling technologies, from experiments to theory (including modelling and simulation) and from DEMO concepts to cross cutting aspects between fusion and other fields.

***Table 1:*** Comparison of the energy extracted in present experiments (highest values) to the goals for ITER and DEMO. $E/S_{pl}$ is the energy normalized to the plasma surface.

| Device | Tore Supra Tokamak | LHD Heliotron | ITER | DEMO |
|---|---|---|---|---|
| Energy Goal | 10MWx1000s =10 GJ | 3MWx3600s = 10.8 GJ | 500MWx1000s = 500 GJ | ∞ |
| Energy Achieved | 1. GJ | 1.6 GJ | | |
| $E/S_{pl}$ ($MJ/m^2$) | 10 | 17 | 700 | ∞ |

The challenge of SSO is illustrated on Table 1 which compares the energy extracted in present experiments from LHD (T. Mutoh PL-1) and Tore Supra (G. Hoang I-6) to the ITER full performance objective. In absolute term, a progress of about a factor 300 is required. However, it is more meaningful to compare the extracted energy normalized to the plasma surface ($E/S_{pl}$). The comparison is more favourable in these terms but still a factor of about 40 is needed. Therefore there is a long way to go! This becomes even clearer when inspection of the plasma parameters obtained during these experiments (in particular the normalized plasma pressure beta) reveals that they are far away from the reactor regime. Nevertheless, the values obtained are improving all the time and this was apparent in this conference. The reason for limitation in pulse duration is not generally due to a single cause and when a particular cause is removed, soon appears another one or a combination of causes. The exact reason is not always clear but the most often stated causes are:

(1) Limitation on the specification of one of the machine systems (field, pumping, heating, current drive, plasma control)
(2) Overheating of a first wall component (insufficient active cooling or improper accounting of the power distribution on first wall components) often creating hot spots
(3) Impurity influx due to (2) or to a particular plasma wall interaction i. e. from heating systems
(4) Radiation collapse due to an 'UFO' (unidentified flying object) often coming from the detachment of a fragment of a deposited layer on a first wall component (flakes).
(5) Spontaneous change of the plasma state. This may be due to coupling of slow growing instabilities to causes (2) to (4) identified above or to the interplay between the equilibrium slowly varying on a resistive time scale and plasma instabilities.

Cause (1) has now been eliminated from the machines of the modern generation but causes (2) to (5) are still impeding further progress in performance. They imply both physics and technology and, more importantly, their integration in a single device for a well controlled plasma state. It is the intention of this summary to highlight issues and progress in understanding and in eliminating in practice the limitations.

## 2. Overall progress in SSO experiments

A first striking observation is the successful advances or completion of the new devices based on superconducting technology. The torus of W7X is now closed and ST1 is nearing completion. A major breakthrough is that EAST (X. Gong PL-3) and KSTAR (H. Park PL-4) are now in full scientific exploitation thus demonstrating for the first time the operation and control of tokamaks operated with superconducting coils producing both the toroidal and poloidal fields. EAST and KSTAR have both produced H modes. EAST has developed hundred-second-long divertor plasmas and 8s H modes (X. Gong PL-3); the performance is presently limited by overheating of the graphite of some components.

LHD, the large helical device in Toki, which is intrinsically a steady state device, has continued to lead the field of long pulse operation reaching one hour operation and 1.6 GJ of extracted power (fig. 1). The triple product $n_e\tau_E T_i$ has now reached $3.5x10^{18}m^{-3}$.s.keV (T. Mutoh PL-1). The unplanned plasma termination can be produced by flakes or by sparks in the IC antenna. A burst of EC power has been used to burn off the flakes thus recovering normal operation. The situation was markedly improved using a dipole antenna which launches a higher k spectrum. Similarly, Tore Supra (fig. 2) has reproduced its 1GJ shots but has done so at higher density and higher plasma current. The improved plasma parameters were achieved with the new LHCD system connecting a new set of 3.7 GHz klystrons to an ITER relevant PAM launcher which is actively cooled.

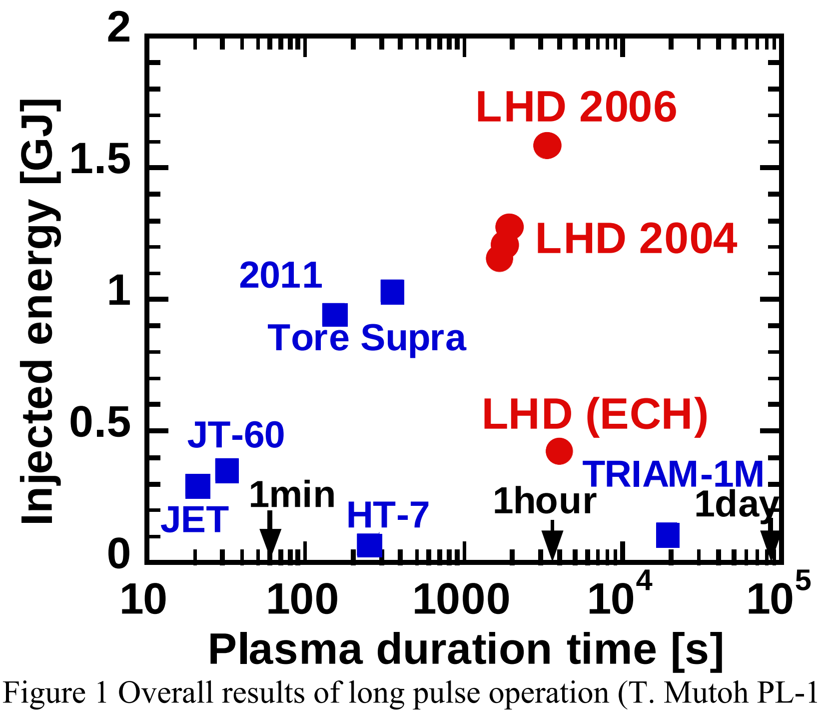



Figure 1 Overall results of long pulse operation (T. Mutoh PL-1)

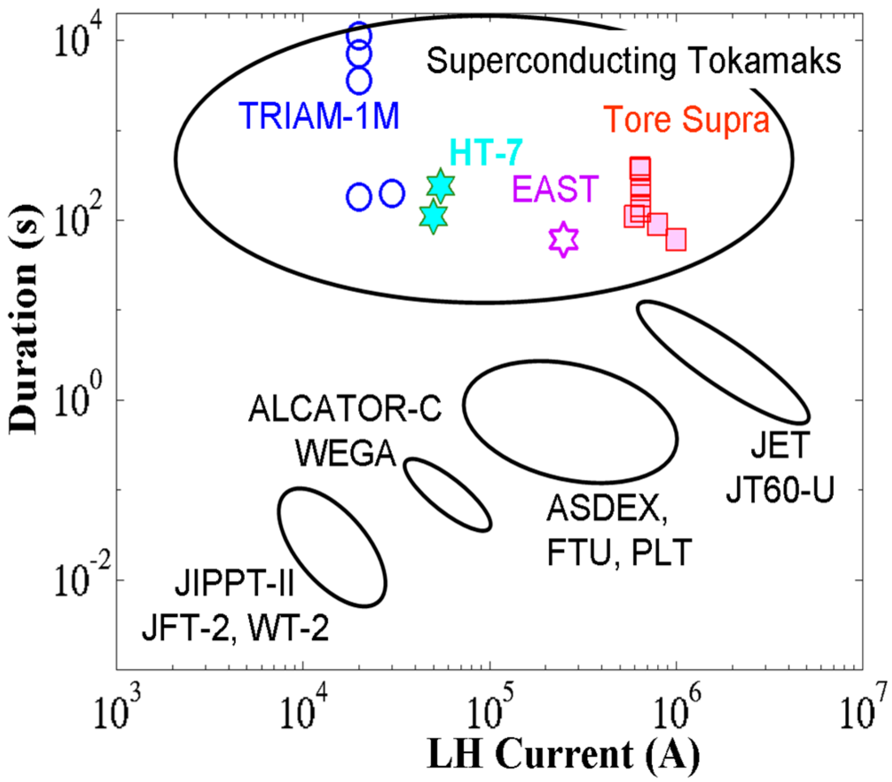



Figure 2: Overall results of long pulse operation of tokamaks with LHCD (T. Hoang I-6) (Reproduced from T. Hoang, Plasma and Fusion Research to be published)

### 3. Theory, modelling, computing.

On the theoretical and modelling side, the present challenges are two folds on one hand the new phase in fusion research requires to integrate in a coherent package the complex interplay of present knowledge on equilibrium, instability and transport and, on the other hand, first principle descriptions of the key physics phenomena which are still not available. For instance, first principle predictions for ITER of the H-mode power threshold and of RMP ELM mitigations are still missing. This conference benefitted from 48 input papers on theory and modelling. It thus provided a good measure of the progress made and of the one which remains to be made.

C. Chang (I-17), focussing on extreme multiscale numerical computation and gave a comprehensive illustration of a coherent package integrating a wide range of physics phenomena involving very different scales.

A full-f gyrokinetic codes XGC1 and a kinetic guiding centre code XGC0 have been developed leading to:

- Whole-volume kinetic first-principles simulation of turbulence in realistic diverted magnetic geometry
- Observation of a globally stiff Ti profile in a flux-driven system
- Plasma momentum generation in the edge and its inward pinch

- Pedestal growth
- RMP penetration, consistently with pedestal transport/evolution.

The results already obtained are impressive and account for a wealth of observations. The key question which remains is the extent of the predictive capability of the codes; for instance is it possible to demonstrate the switch to H-mode and to predict accurately its threshold for ITER?

H. Sugama (I-16) dealt with kinetic simulations of neoclassical and anomalous transport processes in helical systems. Using a gyrokinetic theory and simulation of ITG turbulence, he showed that the neoclassical optimization of magnetic field configurations enhances the generation of zonal flows and hence the regulation of turbulent transport. This result is a rather direct consequence of the radial electric field generated by the ambipolarity condition in helical systems.

F. Waelbroeck (O-19) addressed the problem of the plasma response to magnetic perturbations which is so essential in the ELM mitigation problem. He gave general considerations showing that fusion grade plasmas are resilient to mode penetration. The very disparate time scales involved, $10^4$ in size and $10^5$ in time, are at the limit of nonlinear initial value codes so he used the singular layer approach which shows that the field is screened at the resonance but amplified away from it. A particle pump out is produced which needs, for stabilisation, to act on the pedestal width as well as on gradients. It scales unfavourably with $\rho^*$.

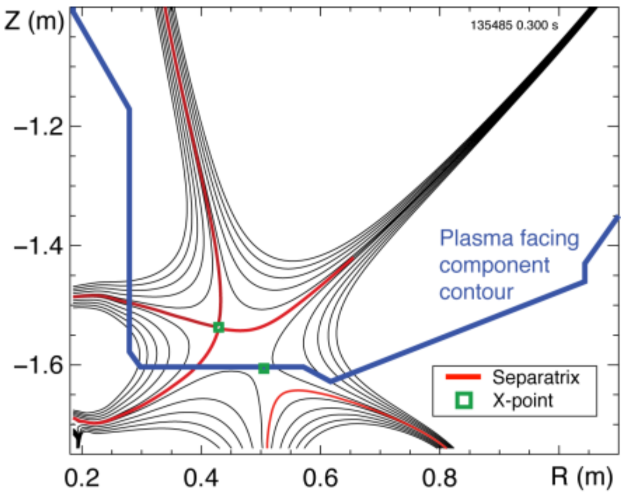
Finally, V. Decik (I-15) discussed the revolution in computing ahead of us by developing codes and hardware which minimizes the memory access time. The use of GPU (graphical processing units) looks very promising for a number of applications.

## 4. Shaping, conditioning and divertor aspects.

Wall conditioning and the thermal loading on first wall components especially the divertor are essential issues for CW operation of fusion machines. This section gathers the related aspects presented at the conference. Interestingly, the effects of plasma and divertor geometries have been revisited and novel ideas have emerged for optimising plasma properties and reducing the thermal load. We first deal with them before moving on to conditioning.

A. Pochelon (I-2) used the great flexibility of plasma shaping on TCV to provide physics insight on its effect on plasma properties. Plasma elongation was varied from 0.9 to 2.8 and triangularity $\delta$ from -0.7 to +1 (positive values correspond to the ITER case). Core confinement and edge pedestal in H mode exhibit strong variations with $\delta$: core confinement degrades with increasing triangularity whilst, in H-mode, the divertor thermal loading due to ELMs increases. The behaviour in the core is reminiscent of the effects of trapped electron modes but is still a puzzle in view of the limited penetration of triangularity in the core. GK simulations can explain such behaviour by showing that the turbulent eddies generated in the edge extend deep in the core. The behaviour of the pedestal is compatible with ideal MHD stability.

Spherical tokamaks (M. Bell I-1, H. Meyer I-5) as well as TCV (A. Pochelon I-2) have devoted considerable attention to reducing the thermal loading on the divertor. In addition to ELM mitigation and radiation increase in the divertor region, this can be achieved by optimising the divertor geometry for a larger wetted area. H. Meyer (I-5) described a "super x" divertor geometry which is being conceived for installation in the upgrade of Mast. It provides a large flux expansion and a long connection length thus reducing the heat load by a factor 4. Particular attention was given to the effect of error fields and ripples which could pose a problem.



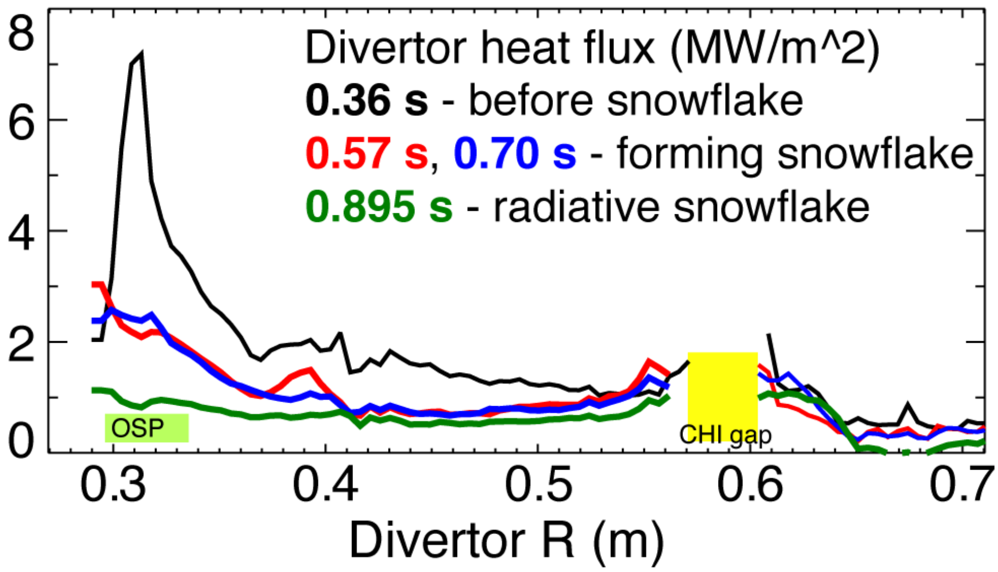




Figure 3: Top plot: Snowflake divertor configuration used in NSTX (M. Bell, I-1). Bottom plot: Measured heat flux before and during the snowflake configuration.

(Original data published in R. Raman et al., Nucl. Fusion 51 (2011) 094011 and S. Gerhardt et al., Nucl. Fusion 51 (2011) 073031)

In a comprehensive review of NSTX results, M. Bell (I-1) also dealt with plasma shaping, mode control and divertor geometry. He emphasized the importance of lithium coated PFC's in increasing confinement, widening the edge pedestal and progressively reducing ELMs. NSTX is now undergoing a major upgrade for operating at higher plasma current and with a reduced collisionality. It is planned to resume operation in 2014 with a $2^{nd}$ beam box and a new centre stack. Strategies are being developed to mitigate the very high divertor heat fluxes in NSTX-U, including using a "snowflake" (higher multipole magnetic field null) divertor configuration. As shown in Figure 3, this has produced a significant reduction of the divertor heat flux in NSTX. Similar configurations have been studied in TCV and MAST.

Emphasis on divertor geometry is also a central piece of the future plans on LHD (T. Mutoh PL-1) with the construction of a closed helical divertor together with active pumping which is expected to improve the control of density at the edge and to increase the recycling in the divertor region thus enhancing radiation and avoiding peaked thermal loading.

I. Tolstihina (O-5) pointed out the importance of the isotopic effect on charge exchange in slow collisions of Li, Be, C and W ions with H, D and T which have resonant channels. This effect could be large and should be taken into account in simulations. Experimental confirmation of the calculations appears required.

H. Takeda (O-6) described divertor simulation studies using the Tsukuba GAMMA 10 end-mirror cell. A heat flux of 0.8 $MW/m^2$ and a particle flux of 6.5 $10^{22}$ particle.$m^{-2}$ were observed. There is scope for a significant increase of the particle flux with additional plasma heating in the end mirror region.

## 5. Elm mitigation.

T. Evans I-3 gave an overview of the suppression of Edge Localised Modes (ELMs) in the DIII-D tokamak with its flexible system producing 3D magnetic field perturbations. Without any mitigation methods, large ELMs would be expected in ITER which would impede the longevity of the divertor. The DIII-D set-up can produce either a narrow stochastic layer (small islands) with a top/down antisymetric field or a wide edge stochastic layer (large overlapping islands) with a symmetric configuration. The ELM suppression in this later case appears to be compatible with ITER low collisionality conditions. The ELM suppressed regime is quite stationary in contrast with the ELM-free regime where the density rises continuously. The ELM suppression which occurs at a well defined q value at the edge is correlated with a reduction of the current and particle gradients. It is very encouraging that ELM suppression has been obtained with all ITER reduced parameters except $\rho^*$ which is not accessible. The ITER RMP coil design (figure 4) is capable of exceeding the DIII-D suppression criterion by more than 50% under all anticipated ITER scenarios

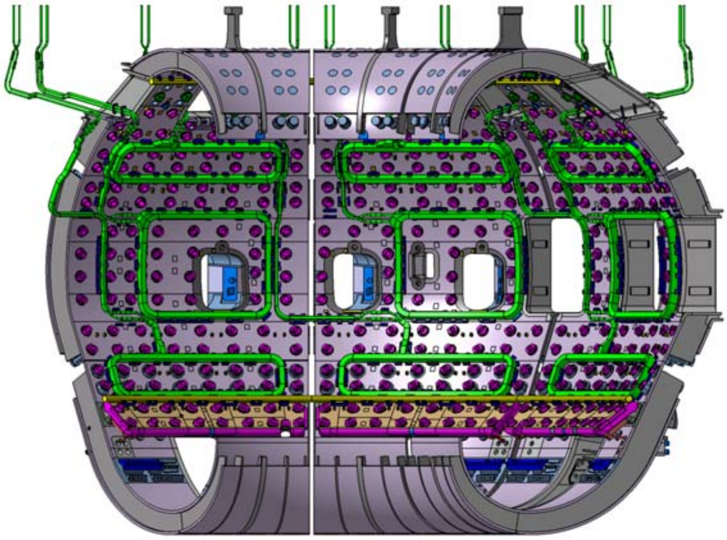


Figure 4: ITER internal coil design capable of producing a broad range of magnetic perturbations at the edge (n=1, 2, 3 and 4)

## 6. Diagnostics, instrumentation and control.

Over the last decades, there has been a continuous revolution in the ability to diagnose fusion plasmas. It was fed both by a growing insight of basic plasma processes and by the exponentially increasing power of electronic data processing. Real time data processing for feedback control has now reached such a power that it is now an essential tool of all plasma operations. It is indispensable for SSO work. A good example of this is the real time surveillance of the Tore Supra PFCs with infrared cameras for power and temperature management. This conference has provided a rich input on the evolution of advanced diagnostics; in particular T. Estrada (I-8) reviewed microwave reflectometry worldwide. Progress was achieved both on the conventional set-up with very fast systems for measuring density and turbulence and on Doppler systems which give new information on plasma flows. Density profiles are now measured with excellent temporal and spatial resolution and the dynamics of turbulence becomes accessible. Describing the ITER reflectometric system, she stressed the challenges for future devices where relativistic effects modify strongly the resonances and cut-off frequencies thus preventing access to the plasma core from the low field side (LFS). HFS launch has to be used. ITER will also use reflectometry for position control during SSO. The principle was demonstrated in AUG.

H. Park (PL-4) gave the latest developments of the KSTAR 2D ECE imaging systems (ECEI). Sawteeth, a new core instability and ELM growth and stabilisation can be visualized in striking details. A second ECEI system located at a different toroidal position is being commissioned for providing 3D reconstructions of MHD instabilities.

A. Melnikov (O-3) reviewed plasma electric potential measurements in 4 devices (T-10, TJ-II, CHS and LHD. Heavy Ion Beam Probing was used to provide a good spatial (1cm) and temporal (1µs) resolution of the plasma electric potential and density. All four machines show striking similarities: when ne increases, the potential grows more negative and is associated with better confinement. Conversely, the potential goes up with ECRH. Neoclassical predictions are compatible with the observed behaviour.

Finally, I. Yonegawa (I-9) from the ITER international organisation described the instrumentation and control for ITER long pulse operation. He stressed the challenge of integrating by a central system many systems procured 'in-kind' with their own local controllers. A plant control handbook defines precisely the standard which is mandatory for all systems. The architecture including a thousand computers has been defined. Many technologies have been selected and a number of pilot projects have been initiated for validating the selected technologies.

## 7. Heating systems.

There have been no drastic changes in the performance of the 4 usual heating and current drive systems but steady progress is noted. The methods have their advantage and drawbacks and it is not envisaged to only rely on a single method. IC and LHCD are presently well suited for achieving steady state operation but need to improve on coupling and on minimizing edge effects in particular on plasma purity. EC and NBI do not have a coupling problem but are still significantly far from achieving the required long pulse operation on plasmas. The progress on gyrotrons is illustrated on figure 5.

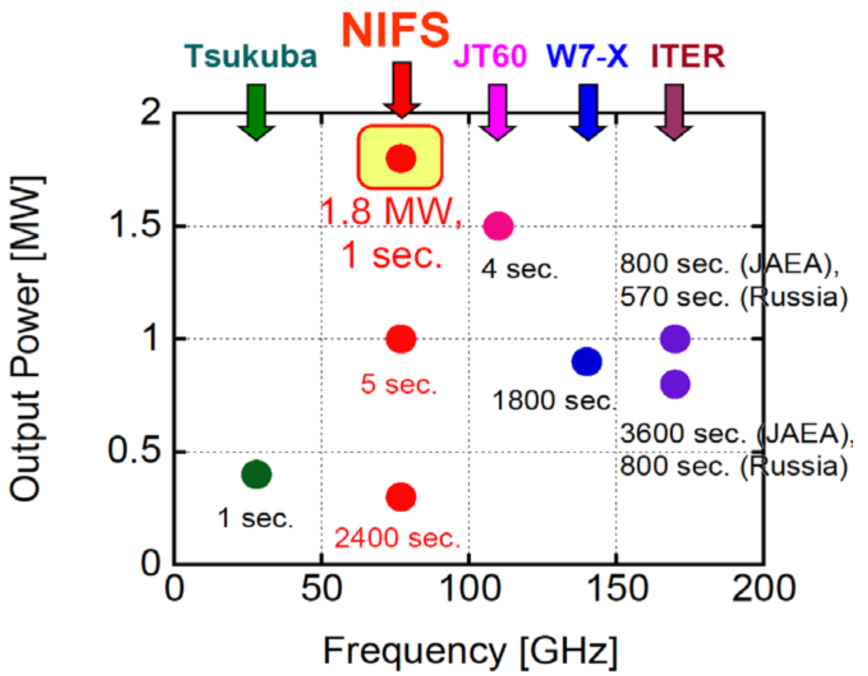


Figure 5: Status of long pulse operation of gyrotrons for magnetic fusion (T. Mutoh PL-1)

T. Mutoh (PL-1) reported that the LHD EC system has been successfully upgraded to 4MW and a very high electron temperature of 25keV has been achieved with 3.7 MW of coupled power albeit at a rather low density of $2.10^{18}$ $m^{-3}$. The upgraded NBI system (16 MW of N-NBI at 180 keV, 13 MW of P-NBI at 40-60 keV) combined with reduced recycling obtained with ICH wall conditioning, allowed an ion temperature of 7 keV to be achieved. New IC antennas were installed to provide a high parallel wave number. In combination with EC, it allowed higher plasma densities to be sustained but plasma was terminated by a sudden influx of iron when a spark on the antenna occurred.

T. Hoang (I-6), on behalf of an international task force, gave an overview on the progress of LHCD for long pulse tokamak operation. The overall results on experiments are summarized on figure 2. ITER scenarios including LHCD show that longer burn durations would be achieved by using it for saving volt seconds during the current ramp-up phase and driving current in the plasma periphery (r/a ~ 0.7) during the burn phase thus tailoring the non inductive drive in advanced tokamak modes. This forms the scientific basis for the design of a 20 MW LHCD system for ITER using 48 klystrons of 500 kW of the type prototyped by the KSTAR group. Prototypes of CW windows and mock-ups of a 5 GHz mode converter have also been manufactured by the Tore Supra team.

S. Shiraiwa (I-4) focussed on the understanding of the decrease of LHCD efficiency at high density which is observed for instance on Alcator-C-mod. Modelling using an advanced full wave code indicates parasitic absorption at the edge. This effect is reduced when single pass absorption in the plasma core is large. It should not happen in ITER conditions where the single pass absorption is predicted to be large.

A. Isayama (O-7) described the plans for the ECH system for JT-60SA. The system will have the multiple functions to heat, stabilize instabilities, plasma initiation and wall cleaning. The goal is to install 7 MW/100s in 9 units. At present, 1 MW for 31s and 1.5 MW for 4s have been achieved. Dual frequency gyrotrons at 110 GHz and 138 GHz are being developed for providing the possibility of core heating at 2.3T. The launcher and the power supplies are also progressing. For the W7X EC system, M. Hirsch (O-9) indicated the strategy to cope with the microwave stray radiation which poses a serious risk of damages of components during long pulse operation.

M. Wada (O-8) described the development of SSO ion sources for ITER NBI systems. Sources with the required current density have been demonstrated but reduction of the tungsten evaporation and of the caesium consumption need to be achieved.

Following a different track from the usual RF systems, H. Idei (I-7) described the physics of electron Bernstein wave heating and current drive (EBWH and EBCD). He demonstrated that the waves are well adapted to the low field of the new spherical tokamak in Kyushu, QUEST, using the TRIAM 8.2 GHz system. A CW phased array antenna was developed and non inductive plasma current start-up and sustainment experiments were performed. About 15 kA for 100 kW were achieved (start-up and sustained). The duration reached ~ 37 s but was limited by impurity contamination. 8.56 GHz 250kW klystrons are being developed for the next phase.

## 8. Superconductors.

M. Noe and P. Komarek (PL-6) provided a highly documented history of superconductivity applied to fusion magnets (figure 6) and then moved on to discussed the future. This history is marked with very successful completions of several superconducting machines starting in 1978 with T7 (figure 6). For instance, Tore Supra, since 1988, and LHD, since 1998, are performing very reliably since their commissioning. More recently EAST and KSTAR operate with both TF and PF superconducting coils. KSTAR was the first project to make the economy of performing separate tests on each magnet. This ensemble of results led to an industrial capacity to construct complex magnets with sizes above 10m and with fields of up to 10 T at temperatures from 1.8 to 4.5 K. Since 2005, high temperature superconductors (HTS) are commercially available (YBCO). YBCO coated conductors arranged in twisted ribbon stacks seem to be the most promising candidate for manufacturing large coils at high fields with this technology. It has excellent mechanical properties and could operate at about 70K. HTS offer a prospect for simplification of the entire cryo system leading to reduced construction and operation costs.

At the Durham European reference laboratory, D.P. Hampshire (O-17) measured the properties of LTS and HTS samples versus strain in a broad range of magnetic field values. He gave the physics insight for the mechanisms at work either within the grain itself of the superconducting material or at its boundary.

T. Hemni (O-17) described the method to measure directly the internal strain of $Nb_3Sn$ cable-in-conduit by neutron diffraction. The initial tests have been successful and systematic measurements which have been unfortunately delayed by the recent huge earthquake are now imminent.

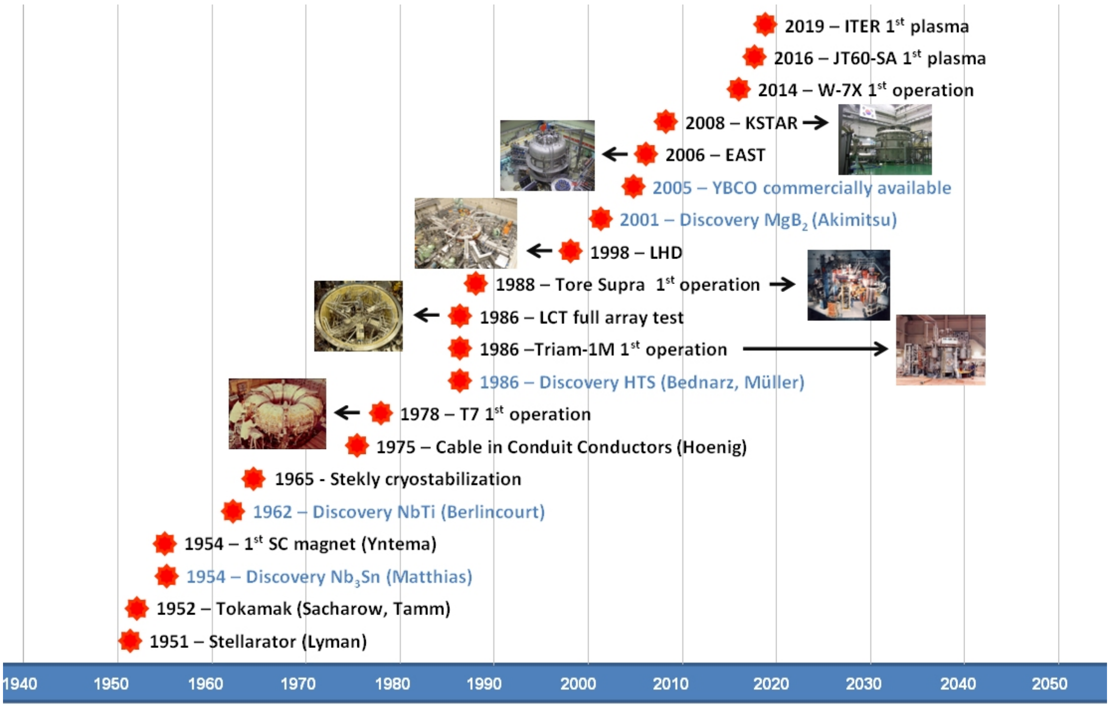



Figure 6: summary of the history of superconductivity applied to magnetic fusion (M. Noe and P. Komarek PL-6).

## 9. Materials.

S. Zinkle (PL-2) reviewed the status of material required for fusion reactors. Whilst the materials required by ITER are available, there is a large gap to be filled before getting all those necessary for a fusion reactor. Research on structural materials which are submitted to 14 MeV neutron irradiation produced by the fusion reactions suffer from the lack of an adequate neutron source such as the one foreseen in the IFMIF programme. It is thus now forced to rely, on one hand, on an indirect experimental approach (triple beam, weaker neutrons from fast reactors) and, on the other hand, on thermodynamic modelling and on numerical simulations of a highly multiscale nature. The challenges focus on moderating the effect of irradiation at around the 25 dpa level, on the temperature increase of the ductile/brittle transition and on avoiding helium bubble formation in the material (figures 7). Three types of materials are studied (figure 8) – reduced activation ferritic/martensitic steels possibly doped with nano size oxide particles (ODS) - refractory vanadium alloys and composite SiC/SiC ceramics. The ODS have improved strength and the potential to withstand the expected helium production within the core of the material but its properties require to be validated; for instance, the tritium trapping rate is not known. It is also stressed that these advanced steels depends very sensitively on the exact composition of the steel and on its heat treatment. However, progresses on thermodynamical calculations indicate pathways for significant improvements with small changes on the detailed process.

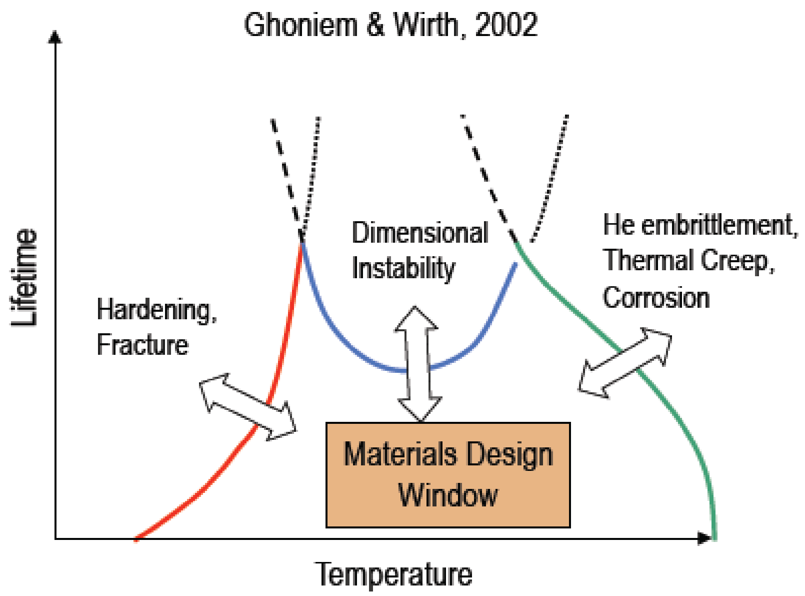



Figure 7: Operating temperature window for materials under 14 MeV neutron irradiation (S. Zinkle PL-2) (Reproduced from S.J. Zinkle and N.M. Ghoniem, J. Nucl. Mater. vol. 417 (2011) pp. 2-8)

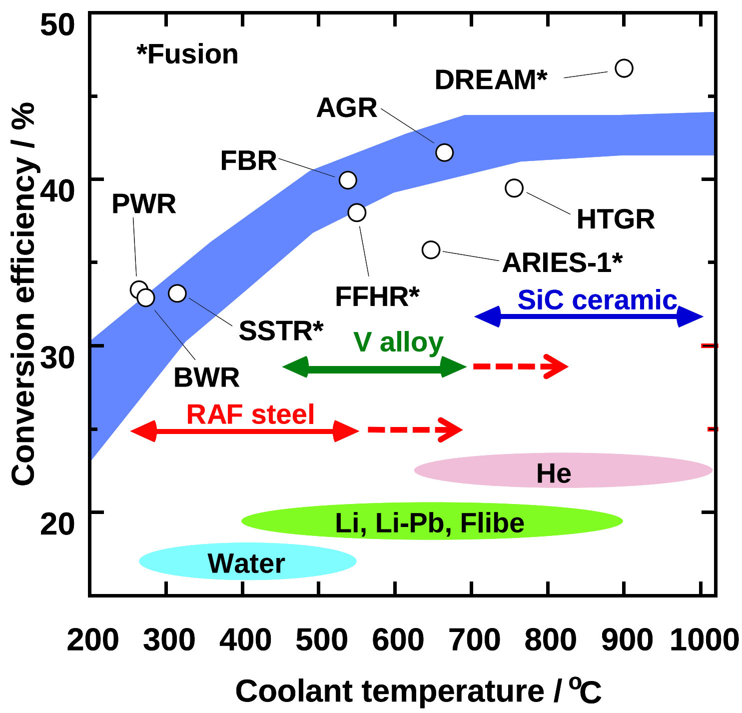



Figure 8: Heat/electricity conversion efficiency versus the coolant temperature allowed by materials (N. Yanagi I-13)

## 10. Blankets, tritium, DEMO and the reactor

The conference attracted a wealth of studies concerning DEMO, the fusion reactors and related technologies. General characteristics of a few types of DEMOs under consideration are listed in Table 2. A number of general observations are striking:

- Several competent groups have recently been formed for focussing on fusion power plant (FPP) R&D.
- Quite different types of fusion plants are considered. They range from very compact steady state tokamaks to large helical systems. Pulsed tokamaks with energy storage are also considered.
- Authors generally agree on the critical issues (K. Yamazaki O-15, S. Zinkle PL-2) and on the major challenges to overcome. For instance, the thermal loading on divertors (R. Wolf O-16) and the choice of long lasting materials under neutron irradiation are quoted by all.
- The success of ITER and of long pulse operation on existing experiments appear clearly in the critical path to the next step.
- The strategies to DEMO and beyond can greatly vary from a group to another in particular on how to bridge the gap between ITER and FPP. IFMIF, CTF and progressive DEMO steps are variable items under discussion among the various groups
- There would be great benefits to enhance the collaboration among the various groups in order to expedite the large amount of R&D work necessary before the construction of a fusion DEMO

G. Federici (PL-5) summarized the activities of the newly formed EFDA power plant physics and technology department. There is a new emphasis in the EU to redirect a predominantly physics research to a reactor relevant technology and physics development. This includes DEMO design activities done in the frame work of the European/Japan broader approach activities. Two different approaches to DEMO are considered: an inductive cyclic concept with modest power density (a fairly straight scale-up of ITER) and a more compact one running in steady state. A key constraint is the divertor power load. Conventional radiating divertors and advanced super X or snow flakes concepts are both envisaged. Putting an initial emphasis on fusion systems design codes, he stressed that a DEMO programme would strongly benefit from a focussed material irradiation programme.

F. Najmabadi (I-12) compared pulsed, with massive thermal energy storage, and SS tokamaks (70-150MW CD power; 60-70% of bootstrap). He finds SST superior for providing a lower cost of electricity and stresses that plastic analysis gives larger engineering margins for SST. A credible energy storage concept should first be developed when considering a pulsed machine.

H. L. Kim (I-18) considers, on behalf of the NFRI DEMO team, cross-cutting ideas for fusion DEMO plant(s) with current and generation IV nuclear power plants (NPP).

DEMO is envisaged in 2 steps: a K–DEMO-1 using the technologies of current NPP with a construction starting in 2022 to validate the technical feasibility and a K-DEMO-2 (construction in 2030) to demonstrate the economic feasibility. In both cases, the selection of materials has to be done at an early stage. The power, irradiation and availability would evolve in 2 stages of power: 60 MWe, 4 dpa and 600 MWe, 200 dpa. Many R&D facilities (High neutron flux, hydraulic test loops, T removal facility etc.) would be shared with the NPP programme. The commonalities would greatly expedite R&D and mitigate risks. The success of ITER and long pulse operation of KSTAR are in the critical path.

N. Yanagi, T. Goto et al (I-13) described the design and R&D for an LHD type heliotron DEMO with tritium sufficiency (Figure 9). For 3 GWth, the machine is large ($R_{plasma}$=14.4m, a=2.54m, Bp=5.1T, about 4xLHD) but draws great benefit for having no CD, no disruptions and a moderate neutron wall load (1.5 MW/m$^2$). The 1$^{st}$ superconducting option is a CIC conductor with $Nb_3Al$. The operating temperature of the blanket is enhanced by avoiding thermal creep with ODS RAFM. The strike point on the divertor is swept using helical divertor coils.

S. Masuzaki (I-20) addressed the particle and heat control in steady state helical reactors. Particle control and a burning rate of 0.18% are achieved with pellets with sizes from 6.2 to 7.8mm and speeds from 1.2 to 10 km/s which appear feasible within the present technical basis. With regards to power control, it is first noted that the divertor trace is nearly 10 times longer than in tokamaks providing less power flux density. However being non uniform, it is still too large so mitigating methods have to be used such as sweeping of the magnetic axis or impurity seeding (increased radiating fraction) or RMP to increase the radiating region.

***Table 2:*** General characteristics of a few types of DEMOs under consideration

| **Demo types** | **Typical parameters (R, a)** | **Pros** | **Cons** |
|---|---|---|---|
| **Pulsed tokamak** | 9.5 m 3m<br>A ~ 3 | No CCD<br>Moderate power density<br>ITER scaled-up | Energy storage to be defined |
| **Advanced steady state tokamak** | 5.5 m 2.6<br>A ~ 4 to 6 | No thermal cycling<br>Smallest machine | Current Drive (CD) required, prone to instabilities, high power density and erosion |
| **Helical DEMO** | 14m 2.5m | No disruption, no CD, moderate power densities | Large size and complexity |

Japanese research on tritium science and technology for fusion was reviewed by T. Tanabe (I-10). Seven facilities can handle tritium in Japan contributing to studying the tritium cycle, its behaviour in blanket modules with a minimum T breeding ratio of 1.07 and its confinement (permeation and leakage), retention and removal. A main challenge is the large tritium through-put required by the low burning efficiently (~1%) combined with requirements for an extremely low leakage and for an accurate accountancy.

The total decay heat in a 3GW fusion DEMO reactor using F82H steel is estimated by Y. Someya (O-11) to be 46 MW after shutdown, 7.3MW after one week and 2.5 MW after a month. The blankets made in F82H are the largest contributors. Tungsten is responsible for the largest decay heat density. Using SiC/SiC would reduce a lot the blanket contribution to the decay heat.

M. Enoeda (I-11) summarized the Japanese developments for an ITER test blanket module (TBM) and for an advanced breeder for DEMO. The programme based on WCCB concept (water cooled ceramic breeder where breeding is achieved with $Li_2TiO_3$ pebble bed) and using RAFM steel (F82H) leads to installation in ITER in 2020 and requires a start of construction in 2014. The impressive programme includes the design, tests (including tritium recovery measurements) and mock-ups. Japan acts as a port master and is the TBM leader for testing the WCCB TBM. Japan is also a partner for the tests of other concepts (HCCB/HCSB, LiPb based TBM).

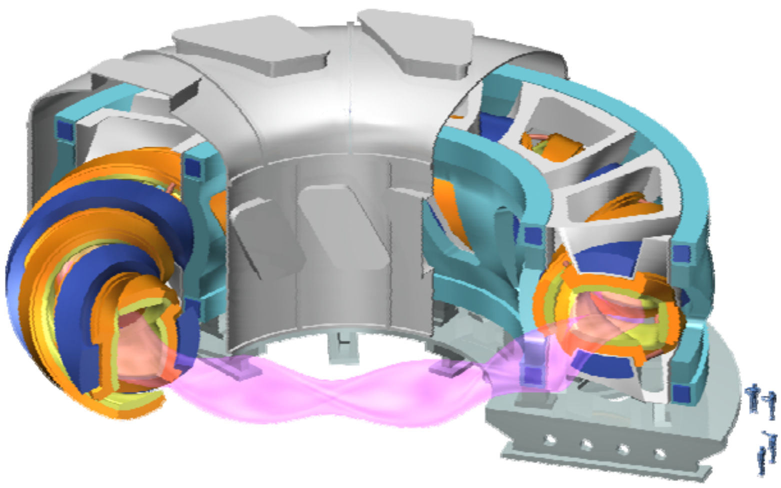


Figure 9: Helical DEMO reactor (N. Yanagi et al I-13)

## 11. Cross cutting aspects

The conference has been a rich forum for synergies between disciplines which, at first glance have little in common. The cross cutting aspects between fission and fusion have been discussed earlier within this article in the reactor section and the common material issues have also been stressed. On a different subject, K. Tanaka (I-19) described the research platform on laser and ion beams and identified the potential contributions to fusion research. For example, the laser produced plumes can be similar to the one in divertors during ELM's and can provide new information on the vapour shielding effect as a function of the material ablated (C, W, Li etc..) as well as on hydrogen co deposition. The considerable stagnation of carbon plumes reveals the importance of molecular formation in absorbing the kinetic energy within the plume.

## 12. Final remarks.

The 2011 international conference held in Toki will be remembered as a very successful venue. This series of conference is a matured major conference which attracts outstanding contributions in a broad domain. Among the numerous areas covered by this summary, we first note that very long pulse operation, which was the main theme of this venue, is clearly a most essential element in the critical path to a successful ITER programme and then to fusion DEMOs. SSO experiments are still in the low GJ domain and there is a long way to achieving ITER and DEMO requirements. The present and the new super conducting machines which are now coming on line, address this issue as a matter of priority. They are well armed for providing answers required not only by ITER but also by the new groups recently constituted for designing the DEMO power plants as the next step in fusion research after ITER. The success both in the attendance and in the quality of the contributions to this conference demonstrates that the SSO issues are being addressed with an increasing level of strength. Despite the good progress achieved to date and the innovations summarised here, this level will need to be further increased in the near future. The scientific community is well prepared to responding to the SSO challenges.

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**Nota Bene:**

The letters and numbers in the parentheses are the IDs of the papers presented at the conference. The program and the abstracts of the papers are available at the following site:

http://www.nifs.ac.jp/itc/itc21/ITC-21_book_of_abstracts.pdf