US/Japan Workshop on
MHD Stability Control and Related Confinement of
Toroidal Plasmas, 6-8 February 2006,
JAEE-Naka, Japan
with ITPA meeting of MHD Topical Group, 6-9 February

Venue and Dates
The workshop will be take place in Naka Fusion Site, Japan
Atomic Energy Agency, Naka-city, Ibaraki, Japan from 6 to 8
February 2006. The workshop will be held with ITPA meeting of
MHD Topical Group, 6-9 February 2006.

Scope of the Workshop
The workshop will include the topics of MHD stability and control,
stability related energetic particle, and MHD stability related to
confinement of toroidal plasmas; Tokamak, Helical, RFP and ST.
- MHD stability and control: RWM, NTM, ELMs, Sawtooth
- MHD Stability related to Energetic Particles: Alfven
eigenmodes, Fast particle confinement, Effects of ripple loss
- MHD stability related to confinement: Degradation or healing
of confinement due to MHD, Interaction of MHD and
confinement, Integrated modeling.

Organising committee for US/Japan workshop
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Japan Atomic Energy Research Institute (JAERI) has been merged with Japan
Nuclear Cycle Development Institute, and the new organization of Japan
Atomic Energy Agency (JAEE) was established in October 1, 2005.
Recent schematic view of National Centralized Tokamak
3. Design Progress of the National Centralized Tokamak Facility

Two designs proposed for the National Centralized Tokamak facility (NCT) is assessed with respect to the physics requirements such as break-even class plasma, heating and current drive capability, MHD stability, divertor performance, and plasma controllability. After fruitful discussions with the leading plasma physics researchers in Japan, the machine parameter is based on that with wider operational space in the plasma shape flexibility which is regarded to be of importance for the achievement of high-β plasma. Engineering design of the main components of superconducting TF and PF coils, vacuum vessel, in-vessel facilities, and cryostat is performed to investigate the structure optimization on the view point of manufacturing processes, operation and maintenance feasibility.

3.1 Physics Design

The machine parameters are assessed on the view point of the capability to break-even class plasma, high-β plasma, heat and particle controllability in divertor, flexibility of aspect ratio and plasma shaping, full current drive controllability [3.1-1].

Break-even class plasma of equivalent Q\text{DT}~1 will be achievable with \( I_p=5.5 \) MA, HH\( y_2 \sim 1.4 \), and the neutral beam power \( P_{NB}=13 \) MW. The consistency with a break-even class plasma and a high-β plasma is estimated, i.e., \( Q_{DT}^{eq}=1 \) and \( \beta_N=3.5 \) will be simultaneously achieved at \( I_p/B_T=4.5\)MA/2.3T, \( q_95=3.5 \), HH\( y_2=1.5 \), \( f_{GW}=0.9 \), and NB power of 25MW. In such a condition, collisionless plasma with low normalized Larmor radius is satisfied in the range of \( \rho^*=0.005-0.008 \), \( \nu^*=0.01-0.1 \).

Advanced operation of high-β with full current drive accessibility at \( I_p=3 \) MA, \( \beta_N=4 \) is estimated by the ACCOME analysis on the assumption of HH\( y_2=1.99 \), \( q_95=6.1 \), \( q_{min}=2.0 \), \( f_{GW}=0.5 \), with total NB power of 25MW in the case of negative NB at off-axis. Current profile control by the combination of on-axis and off-axis beams enables such an advanced operation scenario.

The advantage of low aspect ratio in the MHD stability is evaluated by the ERATO-J code analysis [3.1-2]. Figure 1.3.1-1 shows the dependence of the critical \( \beta_N \) as a parameter of the normalized wall location, \( r_w/a \) (\( r_w \): wall location, \( a \): plasma minor radius), by \( n=1 \) and \( n=2 \) toroidal modes in the double null reversed shear plasma with \( \kappa_95=1.8 \), \( \delta_95=0.4 \), \( q_{min}=2.4 \), and parabolic pressure profile. In general, critical \( \beta_N \) is higher in smaller \( r_w/a \) due to the wall stabilization effect. In the figure, the critical \( \beta_N \) is compared in two cases of plasma aspect ratio of \( A=2.5 \) and \( A=3.0 \). As clearly seen in the figure, the critical \( \beta_N \) tends to be higher in the low aspect ratio,

Preliminary analysis by ‘VALEN code’, under the collaboration with Columbia University and PPPL, is conducted to estimate the achievable \( \beta_N \) with the active control of RWM stabilisation in the 3-dimensional geometry of vacuum vessel and the stabilising plates. The analysis indicates that the maximum achievable \( \beta_N \) is about 3.8, and the limitation is brought by the weak coupling of the magnetic flux of the in-vessel coils with the plasma because of the shielding effect by the stabilising plates. The code analysis in the ITER geometry predicts that the coupling could be effectively enhanced if the in-vessel coils are located around the port duct.

Controllability of the EC resonance for the NTM suppression is estimated by modified Rutherford equation. The minimum EC power for the stabilization of \( m/n=3/2 \), and \( 2/1 \) mode in the resonance of fundamental EC wave injected with 90 GHz, O-mode into the normal shear plasma with \( q_o=1 \) is 0.51 MW, and 1.1 MW, respectively. Those requirements meet the present EC design.
Simulation analysis for divertor particle and heat flux controllability is performed using with SOLDOR/NEUT2D code. In the ITER-like divertor configuration with long leg length, a partial detachment is well maintained. Parameter surveys in the incline angle of the divertor plate and the distance between the pumping duct and the hit point on the plate are also made in order to optimize the divertor geometry. On the other hand, in the optimized shape configuration with a low aspect ratio and a high triangularity, the shortening of the leg length of the inner divertor and the insufficient cryopanel surface area bring the degradation of the pumping speed and of the particle controllability. Further optimization in both the aspects of the divertor pumping and of the plasma shaping is required.

3.2 Engineering Design

Based on the design with wider operational space, structures of main components and manufacturing processes are reviewed to optimize the space utility and maintenance, especially around the midplane area. Whole assemble of the tokamak is illustrated in Fig. I.3.2-1.

TF and PF coils: In order to ensure the space margin for the extension of the flexibility in the aspect ratio and plasma shape, TF coil is enlarged in the vertical direction. Each TF coil has 114 turns to correspond the maximum B<sub>T</sub>R of 8.11 Tm. The number of turns of PF coils are increased to realize the maximum plasma current of 5.5 MA for 100 s. Support structure of the CS and the divertor coil is unified with that of the TF coil in order to cancel out the mechanical stress by the electromagnetic force as an internal force.

Vacuum vessel: The width of the double-wall is designed as 24 mm with the plate ribs of 40 mm in width on the view point of the reinforcement. By the brief stress analysis with FEM code the interval of the ribs is determined to 300 mm with the welding depth of 24 mm.

Stabilizer plates: In order to compensate the thermal stress during the VV baking, crank support structure is adopted. Support leg is made of SUH660 with the electrical insulation coating at the crank-pin and the joint part of crank support. Strength of such a structure is confirmed by the stress analysis with the temperature difference of 300ºC between the vacuum vessel. Mechanical strength is also confirmed against the electromagnetic force during disruption event including a halo current.

Divertor: Movable louver or sliding shutter in order to adjust the divertor pumping speed during the plasma discharge is designed. It is located in front of the cryopanel under the private dome or outer baffle plates. The effective pumping speed for deuterium gas is estimated as 100% to 10% due to the change of the conductance of the adjustable louver or shutter in 100 m³/s to 1 m³/s within the duration of about 1s. Strength of the structure is confirmed by the stress analysis against the thermal stress and the electromagnetic force.

Cryostat: New design of the spherical cryostat is developed in order to ensure the enough space for maintenance in the joint area with NB injection port. It consists of upper pan, middle vessel, lower pan and support base. Each block is connected by the flanges with the lip seal to maintain the vacuum condition. Stress analysis performed by a 3D-model indicates that each part of the cryostat satisfies the structural strength against the complex load from electromagnetic force and seismic force.

Bending strain of Nb<sub>3</sub>Al CICC:

In order to estimate the effect of bending strain on the critical current (I<sub>c</sub>) of Nb<sub>3</sub>Al cable in conduit coil (CICC) [3.1-3], test facility for loading the tensile and compressive stresses is designed and manufactured. The loading test is performed on Nb<sub>3</sub>Al strand (strand sample) wound around the spring-shape holder, and two Nb<sub>3</sub>Al strands and one Cu wire inserted into a
stainless steel conduit (triplex CIC sample). \( I_c \) of the strand is measured with the strain range from -0.86% to +0.18% at 4.2K in the external magnetic field of 6-11 T. The dependence agrees well with the theoretical prediction by Durham's equation [3.1-4]. Based on those results some relaxation mechanism of bending strain in the conduit will be investigated.

**Shielding material:** The typical performance of the heatproof boron-doped neutron shield resin, developed last year, is examined [3.1-5]. The same level of the neutron shielding characteristic as that of polyethylene is confirmed by the penetration tests of 2.45 MeV DD-neutrons and of the continuous energy neutrons from \(^{252}\)Cf source. The heatproof temperature determined by the deflection load is about 300ºC. The tensile, bending, and compressive tests based on the JIS standard show the enough mechanical strength both at room temperature and at 250ºC. The resin is suitable for NCT to set up around the port section to suppress the streaming neutron and at the neutron shielding material for the plasma diagnostics around the vacuum vessel.

**References**


