IFERC Newsletter

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Meeting

7th Technical Coordination Meeting of DEMO Design Activities

The 7th Technical Coordination Meeting (TCM-7) of DEMO Design Activity (DDA) was held at Nagoya University (hosted by Prof. N. Ohno) on 1st-2nd February 2016 with 41 participants (including 16 remote participants); 5 from IFERC-PT, 17 from JA home team and 19 from EU home team. In this meeting, the following topics about progress of DDA in 2015 by EU and JA were covered: 1) plasma design, divertor, first wall (FW), 2) blanket, 3) remote maintenance, 4) pedestal plasma and Edge Localised Modes (ELMs), 5) safety and 6) tritium removal.

1) Plasma design, divertor and FW

Activities in plasma design focus on development of core transport code, investigation of the effect of aspect ratio and vertical stability for higher plasma elongation, disruption study and design of divertor and FW.

- A benchmark test between the TGLF and GENE transport codes showed good agreement.
- The impact of aspect ratio on vertical stability was investigated and the importance of higher plasma elongation for DEMO plasma performance was confirmed. Stability analysis showed that double null configuration and additional conducting shells can enhance elongation.
- The ETA simulation code has been developed for disruption studies and reasonable agreement has been obtained between ETA and DINA for upward Vertical Displacement Events (VDEs) of ITER.
- Divertor design studies explored radiative cooling scenarios and investigated the engineering feasibility of using RAFM pipes. Breeding of sufficient tritium is another key issue. Minima of the divertor target size and tritium breeding ratio (TBR) were evaluated. Making a small divertor can significantly improve TBR.
- Engineering solutions were proposed for the design and integration of the DEMO breeding blanket and FW. This approach is to integrate as much as possible the FW into the blanket box and to adopt a fully decoupled protection (limiter-like solution) only in specific parts where the heat load exceeds the nominal. This strategy is considered to be possible if the heat loads remain within 0.5-0.7MW/m^2 on the majority of the fully integrated (standard) FW. Possible preliminary engineering solutions on the decoupled FW have been presented.

2) Blanket

Thermal structural characteristics of the Water-Cooled Ceramic Breeding (WCCB) Test Blanket Module (TBM) were evaluated by using Finite Element Method (FEM). Assisted by the analysis code, the TBM design was improved and the structural integrity of the modified TBM design was evaluated based on the ASME Boiler and Pressure Vessel Code (BPVC). The TBM box design was improved by adding fillets for in-box Loss-of-Coolant Accident (LOCA). In FW under normal pulsed operation, a fillet radius over 1mm would be necessary for preventing the fatigue damage under 30,000 cycles. The WCCB TBM design has reasonable structural integrity under both normal and in-box LOCA conditions expected in ITER. The design layout of the manifold and cooling pipes in the 4 Breeding Blanket concepts was studied. Design drivers considered are minimization of pressure drops and coolant velocity (in case of helium coolant), the available space in the upper and lower ports and the handling and interface requirements as specified by remote handling. Recent development in ceramic material with mixed orthosilicate and metatitanate composition focusing on tritium absorption/desorption experiments was presented and recent fabrication of Be-Ti rods with the hot extrusion technique was illustrated.

Remote maintenance

To solve one of important engineering issues (the conducting shell to maintain plasma vertical stability) in the banana-shaped all vertical maintenance port scheme, the dependence of plasma vertical stability on conducing shell parameters was investigated with numerical simulations. The double-loop shell best promoted the plasma vertical stability. In order to reduce the risk and to mitigate cooling conditions of the decay heat without an increase of total floor space required for their storage area, arrangement of used segment in the temporary storage area was studied.

Pedestal plasma and ELMs

The pedestal structure has been estimated with DEMO parameters by EPED and MARG2D, with a goal of assessing the applicability of grassy ELM and other modes to DEMO. To predict the pedestal top pressure, scans of several plasma parameters have been performed for DEMO. Triangularity and poloidal beta have a large effect on the pedestal height.

An assessment of the heat impact on divertor plates due to ELMs assuming a burnthrough of the detached divertor plasma shows that a tolerable ELM size is below 1% in terms of $\Delta W/W$ for any assumption, requiring a strong mitigation of natural type I ELMs, in excess of that needed for ITER (factor of 20-100) (Note: W is the plasma stored energy and ΔW is the plasma energy released at an ELM). Mitigation techniques are being assessed, with a special emphasis on how they may impact confinement and what the technical boundary conditions are. In parallel, ELM-free regimes, such as the QH-mode, which may come without penalty in confinement, are assessed as well.

Safety

Safety analysis has focused on large-scale accidents called “upper bounding sequences” to understand their potential hazards and the resulting load conditions to the confinement barriers against radioactive materials.

No melting of in-vessel components by the total LOCA is confirmed. In the case of ex-vessel LOCA, use of the Pressure Suppression System (PSS) for the Reactor Cooling Water System vault is effective for mitigating the accident. In the case of in-vessel LOCA, a concern of over-pressurization of the Vacuum Vessel (VV) was found for a large-scale break of coolant pipes. In order to feed the analysis results back to the DEMO design, parametric scan of the analysis is on-going changing the position of steam relief port and the size of break area, etc.

Tritium removal

An existing facility can be adopted for a tritium removal plant (water detritiation system) of the water-cooled DEMO, even if we assume a conservative tritium permeation rate (6 g-T/day), as long as we can control the tritium concentration of primary cooling water as much as that of CANDU reactor (1 TBq/kg).

As other information, activities in the field of fusion technology at IAEA were presented and JA reported that joint special design team for fusion DEMO has been launched since June 2015.

(IFERC PT, Haruhiko Takase)