IFERC Newsletter

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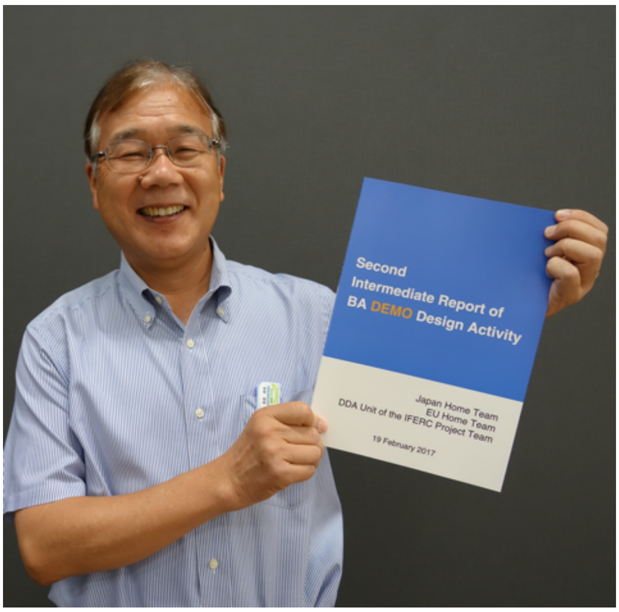
DEMO Design Activity

Second Intermediate Report of BA DEMO Design Activity (DDA)

The 2nd Intermediate report (2nd-IR) of BA DDA was published in Feb. 2017, and summarized the progress of common research topics in DDA after the 1st Intermediate report. The 2nd-IR emphasizes the design integration and impact on DEMO system design, as well as progress in subjects and component design. Two new chapters have been added: "Plant design, balance of plant and tritium processing" and "Key findings and Recommendations for future work".

The DDA is a joint design activity that is supported by domestic DEMO design activities of Europe (EU) and Japan (JA). In EU, DEMO design and associated R&D were implemented by the Power Plant Physics and Technology (PPPT) department under the EUROfusion Consortium. In JA, the Joint Special Design Team for DEMO was organized in order to conduct the DEMO development programme in a nation-wide manner with an enlarged participation of industry.

The goals of the DDA were achieved in May 2017, as originally planned. However, taking advantage of the extension of the BA, work will continue up to December 2019 to further improve and complete the DDA, and a final report will be issued thereafter.



2nd intermediate report and N. Asakura (DDA-Leader)

At this stage, the main design requirements are : (i) to protect the divertor from excessive heat loads, which continue to represent one of the greatest challenges in fusion technology; (ii) to be tritium self-sufficient (TBR >1); and (iii) to convert thermal power into electricity. Integrating these constraints and assumptions into 0-D systems codes has led to similarities of the design configurations identified by EU and JA (Table 1).

Table 1: main design parameters of EU and JA DEMOs

Parameters	EU(DEMO1)	JA (2014)
Major/minor radius, R_p(m)/a(m)	9.1/ 2.9	8.5/ 2.4
Aspect ratio, A	3.1	3.5
Vertical elongation, κ_{95}	1.59	1.65
Plasma current, I_p (MA)	19.6	12.3
Fusion power, P_{fus} (GW)	2	1.42
Net electricity, P_{el} (MW)	500	~200
Magnetic field, B_T (T)/ B_{max} (T)	5.7/12.3	5.94/12.1
Beta normalised β_N	2.6	3.4
Confinement enhancement, HH_{y2}	1.1	1.3
Bootstrap current fraction, f_{bs}	0.35	0.61
Normalised density, n_e/n_{GW}	1.2	1.2
P_{sep}/R (MW/m)	17	< 34
Neutron wall load (MW/m^2)	1	1
Operation	Pulsed/ 2 hrs	Steady-state

Key findings and/or achievements in each area can be summarized as follows:

DEMO design and systems codes

JA has focused on a steady-state DEMO reactor concept, which provides sufficient operational flexibility, including pulsed operation in the initial phase. Vertical stability of elongated plasmas has been studied in depth, as well as the stabilization of elongated plasmas higher than JA DEMO 2014, i.e. κ_{95} > 1.65, to allow higher plasma current and higher plasma density at a normalized density n_e/n^{GW} ~1.2. EU has continued a concurrent approach considering two different DEMO concepts, based on pulsed (DEMO1) and steady-state operation (DEMO2). Additionally, the investigation of a broad range of design options including a double-null magnetic configuration and a flexi-DEMO concept, that allows both pulsed and steady-state operations, has started.

In recent years, physics and engineering modules in systems codes have been improved in EU and JA, and sensitivity analyses were carried out to determine the impact of key uncertainties on the physics assumptions on the overall DEMO performance.

DEMO physics basis

The study of the plasma start-up and shutdown scenarios is a common issue in both JA and EU. JA started the calculation, considering the temporal changes in plasma elongation, poloidal betas and internal inductance by MECS simulation. The studies included flux saving scenario by plasma heating during the start-up, and the fusion output control simulation by pellet fueling. The EU design has an elongation of κ_{95} = 1.59, and the approach is also motivated by maximizing the elongation while maintaining vertical stability and acceptable PF and CS coil power requirements. Several ramp-up scenarios were simulated by METIS.

The location of the peak heat load at the first wall ($q_{\rm wall}$) was determined by both EU and JA, with developing the analysis models. The max. $q_{\rm wall}$ is ~ 0.6 $\rm MWm^{-2}$ for both plasma designs. Fast particle heat loads due to Toroidal Field (TF) ripple were also investigated with orbits following Monte-Carlo (MC) codes, suggesting that the highest fast particle heat loads will not exceed 0.1 $\rm MWm^{-2}$.

Divertor and Power Exhaust

JA and EU studies have covered several physics and engineering designs, leading to a baseline concept based on a single-null divertor with plasma detachment. Remote radiative cooling with impurity seeding is a common concept for JA and EU DEMOs. In JA, about 40-45% of the heating power is assumed to be radiated in the main plasma ($P_{\rm rad}^{\rm main}$), and the divertor design has been investigated with a power handling capability of $P_{\rm sep}/R$ = 25-33 $\rm MWm^{-1}$, where an additional 40-45% is radiated in the SOL and divertor. EU concept requires higher $P_{\rm rad}^{\rm main}$ of more than 60% to maintain $P_{\rm sep}/R$ = 17 $\rm MWm^{-1}$, which is more or less similar to that used by ITER. JA and EU approaches to the power exhaust scenario will provide important case-studies for the future decision of the DEMO divertor design. Both concepts will require highly radiative plasmas with a total radiation fraction of around 80-85% and a peak divertor heat load of 5-10 $\rm MWm^{-2}$.

The target must use water cooling with Cu-alloy pipes, at least, near the strike-point to handle the peak heat load of 10 $\rm MWm^{-2}$. ITER-like technology is currently considered for the DEMO targets, i.e. W-monoblock, Cu-alloy (CuCrZr) and water-cooling, and the impact of neutron-induced damage on degradation of materials properties is being investigated.

In-vessel and tritium breeding blanket (BB) design

Four BB concepts are currently being considered in the EU: the helium-cooled pebble bed (HCPB) and helium-cooled lithium lead (HCLL), both of which are planned to be tested as part of the current ITER EU-TBM Programme, and the water-cooled lithium lead (WCLL) and dual coolant lithium lead (DCLL) concepts. JA has continued the investigation of the water-cooled ceramic breeder (WCCB) BB. The interaction between the BB design activities and the R&D support activities in the fields of functional materials (breeder materials and neutron multipliers) and tritium technologies has been progressively reinforced during the BA.

Assessment of the Tritium Breeding Ratio (TBR) by the optimization of BB is one of the most important common issues. The EU DEMO design of 2014 of HCPB (1.04), HCLL (1.07), DCLL (1.04), WCLL (1.13) will be improved to HCPB (1.20), HCLL (1.17), DCLL (1.26) by design and arrangement optimization. In JA, when the interior design of BB-modules is optimized in accordance with the poloidal neutron flux distribution, TBR is assessed to be 1.16 including a coverage loss of ports and ribs of BB modules. In EU, the back supporting structure for the different type of BBs, their manifolds to distribute/collect the coolant to/from the BB-modules, and the attachment design were investigated.

Remote Maintenance (RM)

Efforts have been devoted to formulate common design requirements for RM. The work has focused on the design of the remote handling equipment for in-vessel components (IVCs), hot cell areas, and the overall reactor layout. The EU approach to vertical maintenance Multi-Module Segment (MMS) was eventually adopted by JA for a DEMO design with $R_{\rm p}$ larger than 8 m. EU has concentrated on the engineering design of a blanket MMS transporter aiming to confirm the feasibility of performing IVC manipulations of massive components. The layout of the pipes in the vertical port and the IVC handling strategies have also been analyzed. JA has focused mainly on a comparative study of the multi-segment and a horseshoe-like scheme.

Other common topics include (i) the divertor transporter engineering designs, (ii) vertical port and transportation to hot cell, pipe-connection technologies and have been improved both in JA and EU. At the same time, radiation conditions after shut-down, before/during RM were analyzed in JA and EU.

Both parties are also developing the concept of the waste storage facilities and waste management strategies. The design studies have highlighted the large volumes of waste generated by the periodic replacement of the IVCs, which has a detrimental impact on the hot cell and waste processing areas, which will have to be several times the size of those for ITER.

Superconducting Magnets

Both EU and JA currently consider Nb_3Sn as the primary candidate material for the superconducting strand of the TF coils. In EU, experimental tests carried out in EDIPO and SULTAN on full scale DEMO conductors showed no degradation with electromagnetic cycles, which constitutes a significant step forward. JA considers cryogenic steels with improved allowable stress to reduce the cross-section of the TF inboard leg.

EU considers 3 different TF coil conductors, with TFC fabrication by single/double layer winding of a strengthened cable-in-conduit type conductor, or pancake-winding. No radial plates (RP) are considered in any of the TFC winding pack concepts. JA also considers a TF design without RP as an alternative to reduce fabrication costs.

An analysis to determine the optimal number of TF coils was carried out by JA and EU, with JA favoring 16 TFCs and the EU presently opting for 18 – but with ongoing studies investigating a 16 TF coil option.

Plant design, Balance of plant and Tritium processing

EU has investigated the primary heat transfer system for WCLL and HCPB, including the intermediate heat storage transfer system, the energy storage system, the power conversion system, and the overall plant layout. JA study on the design of heat transfer system reveals that an integration of several cooling water loops with different temperatures from power source components such as BB, divertor, back supporting structure, is a critical issue in terms of the pumping power. Simplification and integration of the cooling system are necessary for the fusion power plant.

Tritium extraction systems are considered for HCPB, HCLL, WCLL and DCLL breeding blanket in EU, and the most promising candidates are proposed. JA has focused on tritium removal from the primary cooling water for safety, concluding that the tritium removal system of CANDU reactors will be applicable to manage the tritium concentration of the cooling water.

Auxiliaries and other plant systems such as heat storage system for pulse operation and plant power diagram concept considering operation mode, were summarized. Plant layout and plot plan were shown in both EU and JA.

Safety

Functional Failure Modes and Effects Analysis (FFMEA) have been completed for all systems, postulated initiating events have been selected, and the relevant reference accidental sequences have been defined. EU has defined the Structures, Systems and Components (SSCs) important to safety.

JA selected large-scale Loss of Coolant Accidents (LOCA) as upper bounding sequences and has conducted hydraulic and thermal analyses for a water-cooled DEMO blanket concept. Work is in progress to define the confinement strategy for radioactive materials and the accident prevention and mitigation systems. These analyses indicated that the IAEA evacuation-free requirement will be satisfied even large-scale accidents.

A radioactive waste management strategy has also been developed. An important finding from JA assessment based on the residual heat and dose rate of used IVCs showed that all radioactive waste will be low-level waste, and could be disposed in shallow land burial after a cooling-down period shorter than 10 years.

Design requirements and Structural materials R&D

The knowledge on Reduced Activation Ferritic Martensitic (RAFM) steels such as F82H in JA and EUROFER in EU has significantly increased over the past decade. There are still issues at both ends of the design temperature window, e.g., loss of ductility, fracture toughness (DBTT) and creep-fatigue in less irradiated locations (softening). JA and EU cooperated intensively in the development and the compilation of a Material Property Handbook.

Work is ongoing to qualify current grades of steel and to develop, in parallel, new alloys with improved radiation resistance and a larger temperature range. Basic properties and key issues: microstructure and mechanical properties of F82H tungsten-inert-gas (TIG) weld joint was summarized. New developed option: EU sub-project on advanced steels include EUROFER-LT. New released updates: time-independent properties (mainly tensile properties) and time-dependent properties such as creep and fatigue properties were assessed. In JA and EU, design rules have to be further developed and are planned to fill gaps in existing design codes (RCC-MRx, ASME, ITER SDC).

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