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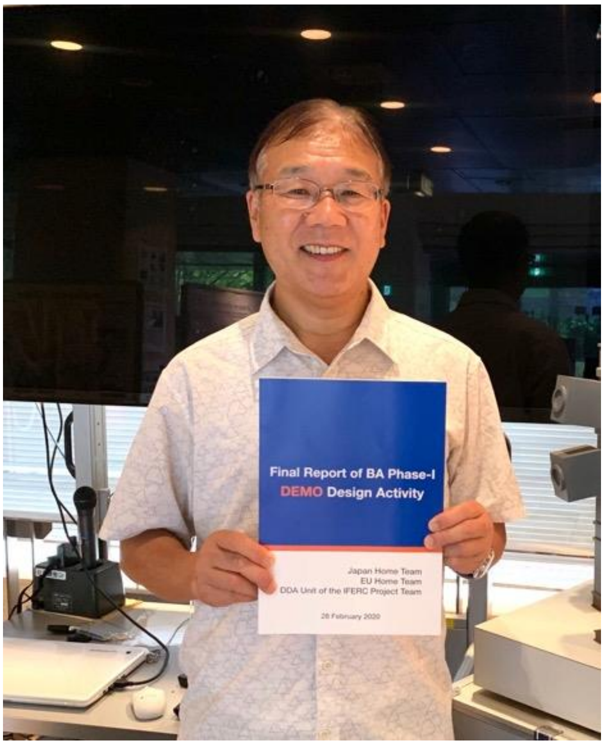
DEMO Design Activity

Final Report of the BA DEMO Design Activity (DDA) Phase-I

The collaborative DEMO Design Activities (DDA) conducted by Europe (EU) and Japan (JA) in the Broader Approach (BA) framework from 2010. The final report of BA DDA Phase-I was published in Feb. 2020, which summarized achievements of common research topics. The DDA final report emphasizes the design integration and impacts on DEMO system design in the nine critical areas. Several joint works between EU and JA were carried out, and the conclusions were highlighted.

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

Work to date has led to the identification of a number of fundamental DEMO design points. These do not represent fixed and exclusive design choices but rather "proxies" for possible plant design options to be used for further investigation. The defining design drivers that have led to the initial selection of design features and parameters are requirements for (i) protecting the divertor from excessive heat loads, which is one of the greatest challenges in DEMO, (ii) achieving a level of self-sufficient tritium production (tritium breeding ratio: TBR >1) in the breeding blanket (BB), and (iii) converting thermal power into electricity, by relying on a realistic Balance of Plant (BoP) configuration. Integrating these constraints and assumptions into systems codes has led to a remarkable similarity in the design configurations identified by EU and JA as shown in Table 1.



DDA final report and N. Asakura (DDA leader in Phase-I)

Table 1: main design parameters of EU and JA DEMOs

Parameters	EU	JA
Major/minor radius, R_p/a (m)	9.1, 2.9	8.5, 2.42
Aspect ratio, A	3.1	3.5
Toroidal magnetic field, B_T (T)	5.9	5.94
Number of TF coils	16	16
Plasma current, I_p (MA)	17.8	12.3
Safety factor, q_{95}	3.9	4.1
Elongation, κ_{95}	1.65	1.65
Triangularity, δ_{95}	0.33	0.33
Fusion power, P_{fus} (GW)	2	1.5
Net electricity, P_{el} (MW)	500	250
Auxiliary power, P_{aux} (MW)	50	83.4
Beta normalised β_N	2.5	3.4
Confinement enhancement, HH_{y2}	1.1	1.3
Bootstrap current fraction, f_{bs}	0.39	0.61
Normalized density, n_e/n_{GW}	1.2	1.2
Primary operation concept	Pulsed/ 2 hrs	Steady-state

Key findings and achievements in nine areas are summarized:

Systems codes

Updated systems codes of PROCESS (EU code) and TPC (JA code) were applied independently to the EU and JA DEMO concepts, and came to similar DEMO reactor concepts as shown in Table 1. JA has focused on a steady-state DEMO reactor concept with slightly smaller R_p = 8.5

m and P_{fus} = 1.5 GW, that provides sufficient operational flexibility, including pulsed operation in the initial phase. EU focused on a pulsed DEMO, but a so-called flexiDEMO that allows pulse and steady-state operation is also studied. EU work also included extensive sensitivity analyses to determine the impact of key uncertainties in the physics assumptions on the overall DEMO reactor design operating performance.

Benchmarking of the PROCESS and the TPC led to refinements and improvements in both systems codes, enabling reliable results from the application of the codes to the exploitation of various DEMO design parameters. Recent improvements include: i) the power balance model relating to radiation power, definitions of energy confinement time and H-factor (HH_{98y2}), ii) the calculation model of the normalized beta value for fast particle contribution, and iii) the divertor prediction model, i.e. physics-based approach underpinning the divertor model in PROCESS was confirmed with results by the JA SONIC divertor simulation code. The benchmarked system codes were also applied to conduct uncertainty analyses to determine the robustness of the assumptions used for each DEMO design. For the parameters and assumptions of the EU DEMO baseline design (DEMO1 2017) examined, a likelihood of reaching the acceptable performance was estimated to be a 60-70%. Over the whole period of the BA collaboration, this exercise has greatly improved understanding of how initial assumptions about device and system performance impact the final global power plant parameters and therefore the overall design strategy.

Core members joined in EUROfusion, Garching

DEMO Physics basis

Common physics issues for the EU and JA DEMO designs have been identified:

(1) The assessment of the ramp-up scenario for highly elongated plasma by means of plasma equilibrium simulators, taking into account the 3-dimensional eddy current effect and the acceptable coil power for vertical position control. The plasma with lower elongation (κ_{95} < 1.65) does not require a dedicated conductor shell, but it is necessary for the higher κ_{95}.

(2) EU and JA developed analysis codes for the plasma heat loads on the first wall, based on magnetic field line tracing. The heat load carried by charged particles tends to be larger around the top region, near the baffle plate, and the inner mid-plane for the present design of the first wall shapes. Requirements for the dedicated wall-protection limiters should be continued in the future.

(3) In view of the challenging requirements on active control schemes in ITER, it is concluded that DEMO needs to assume a scenario with no/small ELMs. The ELM mitigation strategies, including R&D, needs experimental support from the existing tokamaks/ITER and modelling/simulation.

Future work is focus on (i) the identification of heating and current drive (H&CD) requirement for DEMO, (ii) the evaluation of plasma facing components heat loads during transients. In particular, a strategy will be proposed to allow protection of the first wall by the installation of discrete and possibly sacrificial limiters.

It is foreseen that ITER and JT-60SA, together with other devices worldwide, will play a major role and establish a robust experimental basis.

Divertor and Power exhaust

Concepts for the power exhaust and divertor design, consistent with the respective plasma scenarios for JA and EU DEMOs have been developed.

The increase of the radiation power fraction in the main plasma (f_{rad}^{main}) to ~0.67 was proposed for the EU DEMO in order to employ ITER-level power handling in the divertor (power exhaust parameter: P_{sep}/R_p ~17 MWm^{-1}) and to reduce the divertor baffle, i.e. more open geometry. On the other hand, for the JA DEMO, a divertor design appropriate for high power handling (P_{sep}/R_p ~30 MWm^{-1}) in the ITER-like, closed geometry was proposed in order to achieve a higher plasma performance at the ITER-level f_{rad}^{main} (~0.4). The divertor simulation codes showed that detached plasma operation with a peak target heat load (q_{target}) less than 10 MWm^{-2}-level was obtained in the both divertor design concepts. As a joint work, effect of the divertor geometry on the EU and JA divertor performances, e.g. reduction in the peak q_{target} and T_e^{div}, were compared for identical P_{sep}/R_p at ITER-level.

Design considerations focused on water-cooled divertor. Coolant routes for the divertor target (high heat flux component) and the other components including the cassette were an important common issue. Potential material degradation of the CuCrZr pipe under neutron irradiation and appropriate water temperature (130-200°C) are common issues to be addressed. At the same time, several improved concepts of high-heat flux components were developed for the EU divertor, and testing of mock-ups was successful at 20 MWm^{-2} and beyond. A target design with F82H steel coolant piping (290°C, 15MPa) was proposed for the high neutron areas such as baffles and dome in the JA divertor.

Assessments of advanced magnetic configurations for the DEMO divertor were carried out in EU and JA, pointing out major design integration issues in physics and engineering.

In-vessel and Tritium breeding blanket (BB) design

Three different BB design and technologies options have been developed by EU and JA. EU blanket concepts are the Helium Cooled Pebble Bed (HCPB) and the Water Cooled Lithium Lead (WCLL). The HCPB blanket has undergone drastic design changes, adopting a fuel-pin based architecture and opening-up to the possibility of using Beryllides as neutron multipliers. JA is developing a Water Cooled Ceramic Breeder (WCCB) blanket concept. The WCCB blanket uses a honeycomb structure, which acts against the in/box loss-of-coolant accident (LOCA) ensuring pressure tightness. It has a simple interior with a Li_2TiO_3 (T-breeding) and $Be_{12}Ti$ (n-multiplier) mixed pebble bed. Neutronic evaluations showed that the target of the overall TBR (>1.05) is achievable with all the three blanket concepts.

Common design requirements of these BB concepts have been systematically cross-checked with particular focus to the shielding performances for the vacuum vessel (VV) and the toroidal field (TF) coil insulator (JA refers to 0.1 dpa/FPY in the VV, EU to 2.75 dpa over 6 FPY with a nuclear heating target of 0.3-0.5 MWm^{-3}).

For the cooling water activation issue, an enhanced numerical methodology has been developed, and the nitrogen isotopes (^{16}N and ^{17}N) concentrations have been benchmarked (concentrations of the order of ~$3\text{-}4\times10^{16}$ [m^{-3}] and of $1\text{-}2\times10^{12}$ [m^{-3}] for ^{16}N and ^{17}N, respectively). The impact of water activation on the BoP design was analyzed, and some mitigation designs like protecting the valves were proposed.

BB concepts have shown issues in terms of fabrication and reliability. Future work is devoted (i) to improve the BB RAMI, (ii) to demonstrate the integral manufacturing of JA and EU BB, and (iii) to reduce the costs, e.g. recycling for Reduced activation ferritic/martensitic (RAFM) steels, T-breeding and n-multiplication pebbles, etc.

Remote maintenance (RM)

The main differences from ITER are that the mass and size of components utilized in DEMO are much larger; the in-vessel environment for DEMO is more severe, in particular for radiation levels; the remote maintenance strategy for DEMO requires faster maintenance operations, and more operations to be undertaken in parallel compared to ITER. The current view from the EU and JA design studies is that the vertical maintenance scheme is considered the most viable blanket maintenance scheme for DEMO and future reactors.

The shutdown dose rate was analyzed in VV and surrounding ports by simulating different removal scenarios for the blanket segments, divertor cassettes and remote maintenance equipment. Radiation analyses were conducted in JA and EU and the obtained results were broadly similar. On this basis, concepts have been developed for two different blanket and divertor handling systems with two outline options for the transporter engineering design. The design issues associated with the removal of the full blanket segments have triggered the investigation of alternative architectures (blanket segmentation) for risk mitigation.

Furthermore, extensive proposals on different views of in-vessel maintenance scheme, service connection and hot cell facility concepts were beneficial for both EU and JA for their design optimization.

Superconducting magnets

Superconducting magnet design has been studied based on current DEMO design targets that assume a toroidal magnetic field (TF) on axis of larger than 5.3 T both in the EU and JA. A comparison of the TF coil design options (2 in JA, 4 in EU) was performed. Previously, the EU was only considering winding pack (WP) options without radial plates (RPs), while solely options with RPs were studied in JA.

In order to develop feasible WP concepts based on the comparisons of designs, EU and JA have included additional concepts (EU: using RPs, JA: without RPs, respectively) as a joint work. This should be based on the identified common design criteria and assumptions, while taking into account the known differences.

In both EU and JA, the basic concept of central solenoid (CS) and poloidal field (PF) coil design is similar to ITER. In addition, both sides are investigating a concept of a hybrid CS as an advanced option, using high temperature superconductors (HTS) to optimize the CS flux and compactness.

A design of the error field correction coil for mitigation of fabrication tolerance was investigated for JA DEMO. As a common issue, EU and JA plan to adopt an error field correction coils installed at the outside the vessel.

Plant design, Balance of plant (BoP) and Tritium process

EU and JA have developed some initial plant layouts including the BoP concept, and several options of the plant system have been assessed. As EU DEMO is based on the pulse operation, options including an energy storage system (ESS) are contemplated. By keeping a short dwell time around 10 min, a direct cycle between Primary Heat Transfer System (PHTS) and Power Conversion System (PCS) was possible. JA DEMO has chosen a two loop-PHTS directly coupled with PCS, and it also plans pulsed operation during commissioning phase (power-up phase). Possible dwell time can be designed to about 14 minutes for the two loop PHTS.

Comparison of tritium (T) management strategies revealed several common understandings. The Permeation Reduction Factor (PRF) and scale of Coolant Purification System (CPS) have a large impact on control for T-management in PHTS. The potential problem of T-permeation was investigated at the steam generator, especially for the case of helium as a coolant in EU. Further work is also required on the design and technology development of the T-Extraction and Removal (TER) and CPS, as ancillary systems of the BB. The Tokamak Exhaust Processing (TEP) system should be also designed to consolidate design of the fuel system. The development of dynamic T-cycle simulator is necessary to support the fuel system design.

Finally, the tokamak complex layout was compared between EU and JA. Particular issue highlighted was (i) qualification of PHTS valves (isolation and leak detection) and radiation doses, which became a comparable level to the staff and (ii) the availability of the DEMO plant.

Safety

Works was focused on the two water cooling systems, the JA WCCB and the EU WCLL systems. Radioactive source terms and energies were determined and cross-checked. Failure Mode and Effects Analyses (FMEA) were carried out for most systems, postulated initiating events were selected and a few deterministic analyses made. For the In-VV LOCA, both JA and EU consider a Vacuum Vessel Pressure Suppression System (VVPSS) to accommodate the overpressure of the VV. In an External-VV LOCA, JA adopts a guard pipe for the PHTS, whereas EU provides adequate volumes in the tokamak building to maintain the overpressure below the containment design pressure. The characteristics of confinement strategies are shared between JA and EU DEMO concepts, and the impact on plant design was analyzed (e.g., the pressure at the bellows of the guard pipe for the JA concept, and the dimensions of the connection between the VV and the VVPSS for EU DEMO). The assessment of licensing constraints on the plant design and the definition of a credible waste management and disposal strategy were also in progress.

Structural materials R&D

RAFM steels, F82H and EUROFER97, were specified for DEMO application, and non-irradiated material properties have been extensively accumulated and summarized in the material property handbook (MPH). The present version of RAFM MPH has been prepared using common evaluation procedures as part of the joint work. The technical issues on specifying divertor materials, Tungsten as a plasma facing materials (PFM) and CuCrZr as a High-Heat Flux Material (HHFM) were investigated, and a preliminary MPH provided. Work should continue to identify a common CuCrZr alloy and W-material as baseline.

A DEMO Design Criteria document was developed. Rewriting and adapting the existing criteria to fusion specific load conditions and materials was attempted as a short-term approach, and developing design criteria specifically tailored to capture inelastic areas was conducted as a mid-term approach. The use of probabilistic design methods was investigated to enable appropriate consideration of uncertainties in the design of DEMO in-vessel components and its related load cases as well as the limited quality of post-irradiation properties of materials.

Future plans for the DEMO design are determined such as (i) a proper estimate prediction of material degradation under fusion representative environment and unknown loading conditions. (ii) Best estimate data and material properties for in-vessel materials. (iii) Best design criteria and methodologies for materials for nuclear environment and under representative loading.

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