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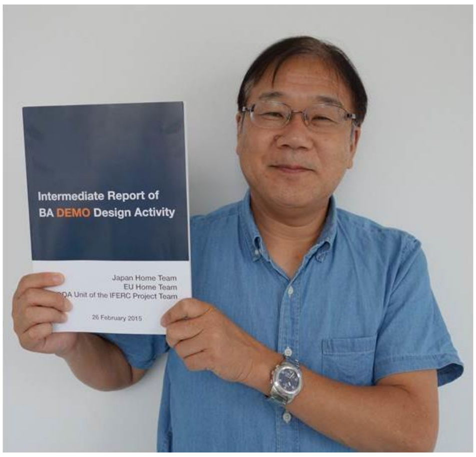
DEMO Design Activity

Intermediate Report of BA DEMO Design Activity



Dr. Asakura, DDA Leader, holding the DDA Intermediate Report

This document is an intermediate report of a DEMO Design study, which began in 2010 with the basic goals of (i) providing a technical perspective of a Demonstration Fusion Reactor (DEMO) with a capability of generating a few hundred MW of net electricity and operating with a closed fuel-cycle tokamak consistent with credible operating scenarios and feasible engineering solutions to critical design issues, (ii) and preliminarily identify a representative range of machine parameters.

Europe (EU) and Japan (JA) are undertaking, as part of the Broader Approach (BA) Agreement between Japan and EURATOM, joint common DEMO Design Activities (DDA) to address the most critical design issues for DEMO and to investigate feasible DEMO design concepts. The work is being conducted by a Project Team as part of the International Fusion Energy Research Centre (IFERC), with strong support from the EU and JA Home Teams. The study will be completed in May 2017 and a final report will be issued then.

Emphasis in this report is on the identification and analysis of key design issues and R&D needs.

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

<u>System Codes and Analysis of DEMO Design Points</u>

The system study work under the BA DDA has been beneficial for both parties, leading to refinements and improvements of used codes. Work was done to study the effects of varying major input physics parameters - e.g. H-factor, Greenwald fraction, and beta limit on the machine performance, to understand the impact of uncertainties and to explore and narrow down the design space and identify/ select attractive design points. A DEMO design, allowing both pulsed and steady-state operations, has also been investigated.

<u>Physics Basis and Scenario Modelling</u>

Several key DEMO physics issues have been analyzed. Assessment of vertical stability for DEMO with R = 8.2 m and A = 3.2 suggests that elongation of κ_{95} = 1.65 could be attainable in the case of the conducting shell position of r_W/a=1.35 (where r_W is the radial position of the inner vessel wall at the mid plane, and a is the plasma minor radius). Another focus of DEMO physics integration has been simulation and preliminary validation of DEMO discharges with state of the art tools. For instance, the core transport module TGLF coupled to ASTRA and the frameworks JINTRAC and CHRONOS.

<u>Divertor and Power Exhaust</u>

For SlimCS with 3 GW fusion power, it is difficult to find a feasible solution for power handling. Recent studies with reduced fusion power design options (e.g., 1.5 GW) showed that solutions based on available power exhaust physics and technology basis are expected for a machine with major radius of 8-9 m. Good progress has been made on the experimental side in EU. In particular the demonstration of P_{sep}/ R = 10 MW/m in ASDEX-Upgrade and experiments with high core radiation (ASDEX-Upgrade/JET-ILW) are important for development of robust DEMO scenarios. Initial scoping engineering studies of advanced divertor configurations (e.g., snowflakes, super-X divertors) showed critical issues and difficulties, while simulations showed relatively small advantages on divertor performance so far.

<u>Vessel and In-Vessel Components</u>

Initial engineering assessments have been carried out to understand the implications on the design coming from requirements, such as tritium self-sufficiency, cooling of breeder units with high temperature coolant, and choice of reduced activation steel as structural material.

Breeding blankets are required on the inboard as well as on the outboard to achieve the tritium breeding target. In addition, breeding blanket design must aim at a reduction of steel in both the first wall and the breeding area. The engineering limits of the first wall cooling technology based on reduced activation ferritic-martensitic (RAFM) steel rather than Cu-alloys indicate that some areas of the first wall will likely require different technologies with higher power handling capabilities.

Remote Maintenance

There is now a consensus between the EU and JA that a multi-module segment (MMS) approach is more suitable. Furthermore, both studies have independently verified the importance of separating the blanket and divertor maintenance activities. The second maintenance option proposed by JA is a saddle-shaped segment scheme accessed via the horizontal port, which reduces the number of pipe connections and facilitates the installation of conducting shells to stabilize the plasma. The EU MMS maintenance scheme was improved, most notably potentially de-coupling the blanket and divertor maintenance. An important finding of the RM design study is that DEMO requires huge areas for hot cell and storage to handle a large amount of used blanket modules and divertor cassettes, which are generated in every replacement.

Superconducting Magnets

Calculation/ design methodologies have been well established independently in Japan and Europe and have been adopted for initial conceptual design of low temperature superconducting (LTSC) TF coils. Currently both EU and JA consider Nb_3Sn as the primary candidate material for strands of LTSC TF coils. Additionally, JA considers cryogenic steels with allowable stress higher than those used in EU. Finally, common work is needed to determine the impact of the field ripple requirements (coming from Physics) on the TF coil design.

Structural Material Design and R&D

The knowledge on RAFM steels (F82H in JA and EUROFER in EU) has significantly increased during the past decade. Nevertheless, there are still issues, at both ends of the design temperature window. In particular, this includes loss of ductility (e.g., uniform elongation), fracture toughness (DBTT) and creep-fatigue (softening). In particular, there are some issues, like fatigue with its asymmetric loading with regard to neutron irradiation, cannot be predicted with today's experiments and modelling tools. Design rules to cope with the issues have to be further developed starting from the Structure Design Criteria for ITER, etc.

Safety

For a water-cooled DEMO with a fusion power of 1.5 GW and average neutron wall load of 1 MW/m^2, source terms and energies that could mobilize radioactive materials were determined for safety analyses. The analyses have focused on upper bounding sequences to outline the confinement strategy, prevention and mitigation systems against accidents and to assess the worst public dose in hypothetical accidents. Analyses included the temperature evolution of main tokamak components after the total loss of coolant, in large-scale ex-vessel and in-vessel LOCA. It was confirmed that no melting of in-vessel components and the first confinement barrier (vacuum vessel) occurs in the total LOCA. In addition, it was suggested that large-scale ex-vessel and in-vessel LOCAs would be formidable accident scenarios to ensure structural integrity of confinement barriers.

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