OVERVIEW OF R&D ACTIVITIES WITHIN IFERC IN SUPPORT OF FUSION DEVELOPMENT IN THE CONTEXT OF THE BROADER APPROACH AGREEMENT PHASE II

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Abstract

The International Fusion Energy Research Centre (IFERC) is one of the three projects executed by EU and Japan under the Broader Approach Agreement. IFERC has three lines of activity: the DEMO Design and DEMO R&D activities, the Computational Simulation Centre (CSC) and the ITER Remote Experimentation Centre (REC). In 2020, the review of the IFERC Project Plan to orient the IFERC activities was performed following the priorities given by the Steering Committee (SC) for BA phase II; (1) to provide the support for ITER, IFMIF/EVEDA, and JT-60SA, (2) to consolidate know-how for future fusion reactors through the production of databases, inputs to engineering hand books, and review of lessons learned in the existing fusion projects. In this paper, the progress of DEMO Design and DEMO R&D activities in BA Phase II are briefly overviewed, stressing on DEMO R&D activities.

1. INTRODUCTION

The International Fusion Energy Research Centre (IFERC) is one of the three projects executed by EU and Japan under the Broader Approach Agreement [1]. The IFERC Project supports the other joint fusion projects (ITER, IFMIF/EVEDA, JT60-SA) [2-4] and contributes to the development of the next generation of fusion devices after ITER, such as DEMO [5-6]. IFERC has three lines of activity. The DEMO Design and DEMO R&D activities aim to share and develop the design of the next generation of fusion devices, and to study and develop materials for these devices, towards a fusion reactor producing electricity. The Computational Simulation Centre (CSC) supports the EU and JA fusion communities with super-computer resources to design components for present day and future machines, to interpret plasma physics data, and to model and design the future operation of ITER and DEMO. The ITER Remote Experimentation Centre (REC) aims to develop remote participation techniques, to give access to ITER scientists to the future ITER operation results and facilitate world-wide collaboration in the ITER exploitation.

In Broader Approach (BA) Phase I, the research achievements are briefly summarized as follows; (1) DEMO Design: the pre-conceptual design including device parameters was shown. The critical issues were identified and R&D tasks to find feasible solutions were specified. (2) DEMO R&D: the various techniques for Tritium technology were developed by using the world-class unique facilities in Rokkasho Institute for Fusion Energy. Reduced Activation Ferritic Martensitic (RAFM) steels F82H in JA and EUROFER97 in EU are characterized and optimized for structural applications and qualified material properties are published in dedicated material property handbooks. In-house fabrication techniques of advanced neutron multiplier and advanced T-breeders are established, and the characterization was implemented for functional materials. R&D results in fundamental database of mechanical/physical/chemical properties for SiC/SiC compositions. (3) CSC: Helios was provided and operated with a very high availability and usage rate as a dedicated machine to magnetic fusion research.

Published papers are seen in a wide range of activities. (4) REC: Remote facility for remote participation in ITER experiment was prepared and the functions of hardware and software developed were verified with WEST, JET and JT-60SA.
In 2020, the review of the IFER Project Plan to orient the IFERC activities was performed following the priorities given by the Steering Committee (SC) for BA phase II; (1) to provide the support for ITER, IFMIF/EVEDA, and JT-60SA, (2) to consolidate know-how for future fusion reactors through the production of databases, inputs to engineering hand books, and review of lessons learned in the existing fusion projects. In this paper, the progress of DEMO Design and DEMO R&D activities in BA Phase II are briefly overviewed, stressing on DEMO R&D activities.

2. DEMO DESIGN

The DEMO Design Activities in BA Phase II have addressed the following joint activities with high priority: (1) Plasma scenario development, (2) Divertor and power exhaust, (3) Breeding blanket design and tritium extraction and removal, (4) Remote maintenance and (5) Safety.

2.1. Plasma scenario development

The required heating and current drive (H&CD) power level has been assessed for the L–H transition and pulsed operation for more than two hours in JA DEMO, supposing the use of electron cyclotron wave injection and neutral beam injection [7], and similarly in EU DEMO, leveraging simulations with ASTRA/Simulink and the DEMO Flight Simulator [8]. EU efforts focused on optimizing H&CD systems for both ramp-up and flat-top phases, demonstrating effective strategies to control fusion power fluctuations (~100 MW for tens of seconds) through coordinated use of plasma elongation, ECRH power, and pedestal top density. The H&CD power required for the L-H transition and pulsed operation for more than two hours in JA DEMO are evaluated to 100 and 80 MW, respectively, by performing integrated modelling simulations, with similar levels estimated for EU DEMO through ASTRA and CREATE-NL-based modelling. The charged particle heat loads to the plasma facing components during the steady-state and off-normal transients have been evaluated, developing models and conducting the inter-code benchmarking. Based on these evaluations, the discrete limiter concept has been proposed for EU DEMO [9], while a poloidally continuous limiter is favoured for JA DEMO [10], to protect the first wall against high heat loads.

2.2. Divertor and power exhaust

Power exhaust concept and the appropriate divertor design are common critical issues for EU DEMO and JA DEMO, and physics study and engineering strategy on the DEMO divertor designs were reviewed as joint work [11,12]. Power handling and particle/He exhaust for EU DEMO divertor were simulated by SONIC as an important collaboration work, and the performance was compared with those obtained by SOLPS-ITER simulation [13]. Energy dissipations in ion and electron transports were different between the two codes while the total energy dissipation was comparable. He/impurity concentrations in SOL and plasma edge were also different. These produced difference of the heat load profiles, particularly, in the partially detached (outer) divertor. Further investigations of neutral and impurity transport models have been carried out.
The divertor design was revised for new EU DEMO plasma concept, which considered a low-aspect ratio, and strategy for simulation tools was summarized. Faster mappings of divertor operational space and fast edge models have been developed to determine the core plasma requirements, coherent with the plasma exhaust part (e.g. Zeff vs divertor heat-flux).
The SONIC simulations including He exhaust for JA DEMO divertor was carried out. Both Ar impurity shielding and He exhaust could be acceptable for large gas puff case of the JA DEMO high-κ option design [14-15], which can accept Ar concentration (c_{Ar}^{edge}) level (0.6%) larger than the JA DEMO 2014 design (0.23%). Sub-divertor gas pressure became ~2 Pa for the large gas puff case, which will satisfy the requirement of the pump system. Effects of neutral-neutral collisions (NNC), photon absorption and impurity thermal force were investigated to determine feasibility of power and impurity/He exhaust scenarios. Progress of engineering design for EU and JA DEMO divertors was summarized. For redesigning campaign for a new EU DEMO baseline, R&D of cassette and target technologies, and structural integrity assessments were carried out. An alternative option to adopt PWR (or WCLL) cooling condition for the cassette body (CB) was investigated together with the EUROFER97 steel structure such as a ribbed-box architecture and an array of multiple circular cooling channels. Divertor structure design and coolant circuits for JA DEMO divertor concept, i.e. double-coolant circuits of 200◦C coolant for high heat load target PFUs (CuCrZr-pipe) and PWR coolant for high neutron load PFUs (F82H-pipe) and CB, were

recently revised [16]. CFD calculation for all PFU support designs and stress analysis for CB and target PFU were carried out.

2.3. Breeding blanket (BB) design and tritium extraction and removal

In Japan, the conceptual design of the breeding blanket (BB) was developed [17]. The design of the water-cooled solid breeder (WCSB) blanket was updated to a cylindrical design using beryllide blocks as neutron multiplier and heat transfer material. The influence of magnetic fields on ferritic steel structures was investigated, and the application of probabilistic design methods was attempted. In the EU, the design of the three BB concepts, i.e. the Water-Cooled Lead-Lithium (WCLL) [18], the Water-cooled Lead Ceramic Breeder (WLCB) and the Helium Cooled Pebble Bed (HCPB) has successfully progressed [19-20], and a complete set of analyses has been preliminary concluded in view of the next design gate. A preliminary BB attachment system design for the VV has been developed, considering off-normal plasma operation modes. The conceptual design of DIR loop for JA-DEMO was investigated. Some investigation was also carried out into the water detritiation system (WDS) as a coolant purification system (CPS). A comprehensive simulation tool for the EU-DEMO fuel cycle is being developed to predict dynamic behaviour and assess key metrics like operational tritium inventory.

2.4. Remote maintenance

The purpose is to identify and study solutions that have the potential to solve maintenance strategy challenges for the ex-vessel and active maintenance facilities of EU DEMO and JA DEMO. Aim was also to identify engineering tools for remote maintenance design, safety and optimization. In parallel with this approach, a major effort has been made between JA and EU to clarify one unified vocabulary and common maintenance indicators, enabling us to jointly define the objectives underlying the achievement of DEMO's operational availability. What is the maintenance policy and strategy needed to yearly guarantee the generation of 2628 hours electricity with a DEMO Power Plant? To guarantee the long-term economic viability of nuclear fusion this implies, that the integration of physics and all technological systems must enable a reliable and safe DEMO fusion power plant ensuring a challenging 30 % for the required availability. For these purposes several studies were conducted to identify maintainable architecture supported by the establishment of a remote maintenance facilities test strategy. Studies about application of condition monitoring in DEMO remote maintenance system or development of RM tools related to the active maintenance facility within different maintenance scenarios were beneficial for both EU and JA in improving their maintenance approach at plant level but also design creativity as RM systems level. This effective collaboration has enabled us to draw up a robust and synthetic functional analysis of a clear EU/JA DEMO maintenance transversal function, integrating industrial and regulatory constraints. This supports the necessary demonstration the nuclear safety demonstration driven by the Reliability, Availability, Maintainability, Inspectability (Regulatory & Statutory Inspections) and Security which are among DEMO maintenance management key pillars.

2.5. Safety

To enhance comprehension of the safety characteristics of fusion reactors, the safety system was developed by first identifying initial events and accidental sequences in JA and EU fusion DEMO reactors. Safety analyses were then carried out for these events [21]. In addition, an inventory of radioactive source terms such as tritium (T), active impurity gases, and active corrosion products was identified to ensure the safety of the DEMO reactors. In JA, a 2-D coupled analysis code for nuclear, thermal, and tritium diffusion has been developed to evaluate tritium retention in the in-vessel components. In EU, a working group has identified preliminarily all the inventories present in DEMO: both the radioactive ones (dust, ACP, Tritium, etc.) and non-radioactive ones that can initiate or propagate an accident as, e.g. inventories of cryo-fluids, magnetic energy, enthalpy of coolants. A radwaste management scenario with recycling and reuse strategies for volume reduction was proposed for JA DEMO. In EU, a complex R&D programme has been set up to validate the models adopted in the safety analyses and to minimize the radioactive wastes amount. In the assessment of licensing constraints, finally, JA have considered the process of attracting experimental reactor ITER to Japan and investigate the licensing constraints on the fusion plant design at that time. EU is considering the experience of Licensing of ITER, and the relevant lessons learned, as a good reference to start the development of standards for Fusion Power Plants as EU DEMO [22].

3. DEMO R&D

DEMO R&D tasks started as the Phase II activity in 2020 in the following 4 areas: Task 1 (T1): R&D on Tritium Technology, Task 2 (T2): Development of Structural Material for Fusion DEMO In-Vessel Components, Task 3 (T3): Neutron irradiation experiments of Breeding Functional Materials (BFMs), Task 4 (T4): Development of material corrosion database.

3.1. T1: R&D on Tritium Technology

For DEMO design, it is extremely important to develop and validate the technology for the tritium continuous processing and accountancy. This system shall handle the tokamak exhaust and the flow of tritium extracted from the breeding blanket. The main purposes are: 1) Development of Tokamak Exhaust Processing (TEP) design and corresponding technology focusing on the common development of analytical tools and analysis systems for tritium accountancy. For the most urgent needs in terms of analytics development an associated hardware development will be assessed and carried out, and 2) Set-up of detailed models in the fuel cycle simulator. In addition, 3) Analysis of plasma wall interaction using JET ILW and JET DT samples for evaluation of tritium inventory and tritium recovery.

3.1.1 Analysis of plasma wall interaction using JET DT samples for evaluation of tritium inventory and tritium recovery

Joint European Torus (JET) was operated with metallic plasma-facing components (PFC): beryllium (Be) tiles in the main chamber and tungsten (W) in the divertor: JET with ITER-Like Wall (ILW). Studies of materials retrieved from JET comprised all aspects of morphology changes, from macro to nano, and quantitative assessments of fuel retention with particular emphasis on tritium both in plasma-facing components (PFC) and in dust.

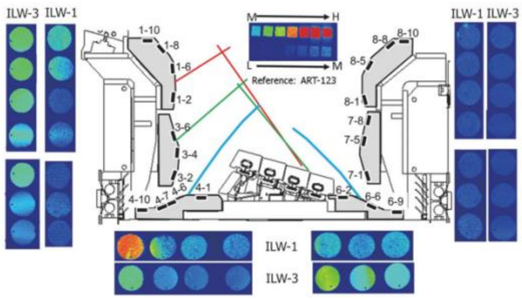

Fig.1 TIPT results of cored samples in ILW-1 and ILW-3 and for a reference T sample. Colored lines show the positions of the separatrix in three typical configurations, ILW-1 (green), ILW-3(blue). Reproduced from Fig.7 in Ref. [24].

Divertor titles after JET-ILW campaigns and dust collected after JET-C and JET-ILW operation were examined by a set of complementary techniques (full combustion and radiography) to determine the total, specific and area tritium activities, poloidal tritium distribution in the divertor and the presence of that isotope in individual dust particles [23]. The study was carried out on two types of samples: (a) JET-ILW divertor tiles W-coated carbon-fibre composites (W/CFCs) after ILW-1 and ILW-3 campaigns; (b) samples of dust retrieved after the operation in JET-C and then in ILW-1 and ILW-3. Tritium analyses were performed by means of radiography using a tritium imaging plate technique (TIPT), full combustion method (FCM) and liquid scintillation counter (LSC). Figure 1 shows results obtained with TIPT for both ILW-1 and ILW-3. The images of cored samples show the tritium distribution on the titles. Larger amounts of T accumulated on the inner vertical divertor tiles than on the outer ones for both ILW-1 and ILW-3. However, the area with T is expanded downwards in the ILW-3 case. The intensity is the highest for sample 4-10 (ILW-1) located in the inner divertor inside the pumping duct. This result can be associated with the presence of tritiated carbon co-deposit, i.e., legacy after JET-C. Other differences in the T distribution on tiles 4 and 6 can be attributed to the difference positions of the typical divertor strike points in the two campaigns, as marked with green and blue lines.

The main findings and conclusions are listed below:

(1) The central part of outer poloidal (OPL) showed the largest T retention in the plasma-facing surfaces of the Be limiter tiles in the main chamber. The erosion was dominant over the deposition in this region, and hence the large T retention was explained by the implantation of high energy T (up to 1.01 MeV) produced by DD fusion reactions. The cross-sectional TEM observation of the subsurface region showed the presence of nanoparticles enriched with N and O. Understanding of the influence of these impurities on T retention is a future task.

(2) The crucial contribution of this work to fusion science technology relates to the first-ever determination of the distribution and retention of T on the surfaces in the castellation grooves of JET-ILW Be limiters. The

deposition profiles of T in the castellation grooves of the Be limiter tiles after ILW-3 were like those after ILW-1; The large T retention at 1–2 mm depth from the entrance of grooves was observed together with the gradual decrease with increasing depth in the deeper region. The variation in T retention was explained by the difference in discharge conditions between ILW-1 and ILW-3. The results complement earlier data for PFS and those obtained in deuterium retention studies.

(3) In the divertor region, the T-containing co-deposits were preferentially formed on the W-coated CFC tiles located at the upper-inboard region (Tile 1) and shadowed regions of floor tiles (Tile 4 and Tile 6). The implantation of high energy T occurred more uniformly. Hence, the difference in the T retention between inboard side and outboard side was smaller than the difference in the D retention.

(4) The co-deposit formed on the shadowed region of Tile 4 (Sample 4-10) in ILW-1 showed outstandingly high T concentration. The small but well detectable amount of T remained in the co-deposit even after heating in a vacuum to 1273 K for the TDS measurement. The analysis of chemical composition of the co-deposit using XPS indicated high C fraction at this particular position. Hence, the high T concentration and the presence of residue T after TDS measurement were ascribed to trapping of T by C. Such co-deposits with high C and T contents were not observed after ILW-3.

(5) The T retention in the W-coated CFC divertor tile after ILW-3 was higher than that after ILW-1. This observation was explained by the larger amount of T produced by DD fusion reactions due to larger number of shots and input energy in ILW-3. Using FCM it was found that the poloidal distributions of the surface and bulk trapped tritium are quite different. The total tritium activities show significant differences between the JET operation with ILW and the earlier operation with the carbon wall (JET-C) indicating that tritium retention has been drastically decreased in the operation with ILW.

(6) The T retention on PFS of the bulk W tile lamellae was negligibly small, while co-deposition of T with Be and O were observed on the side surfaces facing to toroidal and poloidal gaps. A part of O was present as OH, and hence it is plausible that a part of T was in the form of OT. No significant difference was observed between ion drift side and electron drift side, while the outboard side showed larger T retention than the inboard side.

(7) First-ever detailed studies of the deposit structure in the gaps were performed. Two kinds of deposition structures are distinguished: "homogeneous" and "directional". The directional ones are characterized by nanoscale inclined vertical stripes in the deposition layer. An uneven surface (hills and valleys) of only a few micrometers makes a significant difference for the deposition. The results clearly indicate the impact of the surface finish even in the tile gaps, both on the qualitative and quantitative aspects of deposition.

(8) By using a metallic wall instead of a carbon wall at JET, the amount of dust generated is drastically reduced, and the amount of tritium retained in the dust is also reduced. Therefore after approximately 62 hours of operation, the JET ITER-Like Wall has demonstrated a reduction in dust generation and has been effective in reducing the activity of tritiated dust in the fusion reactor.

Results of this work on dust have answered at least two questions regarding the accumulation of tritium in dust: (i) where is the isotope retained; (ii) how does the accumulation evolve over time, i.e. in consecutive operation periods? It has been clearly shown that the T retention is predominantly associated with C grains, while nearly two orders of magnitude lower T intensity was recorded in Be- and W-dominated particles. Secondly, the number of C grains decreased from ILW-1 to ILW-3 and, it was accompanied by a significant drop of the specific T activity. It is a positive message for ITER.

A global fuel retention pattern in JET-ILW could be constructed. The measured differences between the retention after ILW1 and ILW3 points once again to the role of C being a legacy after the JET-C operation. This, in turn, reinforces a positive message with respect to fuel retention in gaps for the all-metal ITER machine where in-vessel C impurities will be minimal.

3.1.2 Development of conceptual design of Tokamak Exhaust Process (TEP) systems for DEMO

In JA, catalyst development for TEP is underway to optimize the process and refine its concept. Traditionally, methane decomposition required high temperatures (~500°C) and additive gases. However, catalysts using supported noble metals like rhodium or palladium enable decomposition at lower temperatures (~200°C). These catalysts were therefore studied to evaluate how temperature and additive gas composition affect the product gas and methane conversion rate. Development and demonstration of remote gas analysis system using a laser Raman system was done for fuel cycle system including TEP. The system has been successfully downsized to be a practical scale and can evaluate the concentration of each gas in-situ in an interval of a few seconds. By contrast, the EU-DEMO fuel cycle has been structured into a three-loop architecture, comprising the Direct Internal

Recycling Loop (DIRL), Inner Tritium Plant Loop (INTL), and Outer Tritium Plant Loop (OUTL). This architecture enables separation of core tritium processing functionalities: DIRL supports immediate recycling of the bulk exhaust gas, INTL manages impurity removal and isotopic rebalancing, and OUTL is responsible for tritium recovery from breeding blanket and coolant systems and for monitoring net tritium production. Performance of the fuel cycle was evaluated using five dimensionless performance metrics (fuel recirculation, rebalancing, extraction, recovery, and direct internal recycling) derived from the plant parameters and tritium handling functions. Operational tritium inventories for each of these functions were calculated for the reference design point the cumulative inventory across all subsystems was approximately estimated.

3.1.3 Development of T Inventory evaluation tool for DEMO fuel cycle design

JA's original calculation codes were specialized for ITER and needed to be generalized for the DEMO reactor. Therefore, they were redeveloped. For the hydrogen isotope separation system (ISS), which handles large amounts of liquid tritium, experimental data is limited due to safety concerns. The new ISS code offers improved operability and has been validated against previous experimental data. Its results match earlier calculations within a ±1.2% margin, confirming its equivalence to the old code. By contrast, the preparatory R&D work has been carried out within the framework of the Work Package Tritium Matter Injection, Vacuum of the European Fusion Programme. This included the development of purpose-built, physics-based unit operation models for all primary fuel cycle technologies. These models have been implemented in the Aspen® Custom Modeler platform and integrated to reproduce the full EU-DEMO fuel cycle architecture at the unit operation level. The resulting Fuel Cycle Simulator enables accurate prediction of the fuel cycle's transient characteristics and the evaluation of key performance metrics, including the operational tritium inventory. Furthermore, its architecture allows close coupling with the sizing and design activities of individual subsystems, making it a central tool for the integrated design and assessment of the DEMO fuel cycle in the ongoing concept design phase.

3.2. T2: R&D on Development of Structural Material for Fusion DEMO In-Vessel Components

The most significant issues for developing structural design codes (SDC) for fusion DEMO in-vessel components include that existing design codes must be extended to be applicable for highly irradiated structures under fusion neutron spectra. Existing code frameworks are based on long-term experience in fission as well as for different classes of materials. As fusion DEMO SDC for in-vessel components must be developed (nearly) without any experience and feedback from facilities under similar operating conditions a clear strategy and a strong international collaborative effort are mandatory. The final objective of this R&D is to develop the technical basis for DEMO SDC for in-vessel components, i.e., Breeder Blanket and Divertor and respective material Annexes for RAFM steels, Cu-alloys, and W-based materials. Probability-based design methodologies are considered in parallel to the conventional deterministic design method. An irradiation database and respective methodologies to best estimate the fusion neutron irradiation effects shall be developed considering both design approaches. Collaborative neutron irradiation campaigns shall be elaborated to qualify and validate the whole set of data, design methodologies, and the structural design rules developed.

3.2.1 Development of Material Properties Handbooks (MPHs)

Work is progressing towards the definition of the 2nd version of MPH for structural materials, also with consideration of probability function of properties for engineering design of DEMO. Bayesian method is applied to determine reference standard strength for neutron-irradiated F82H [25]. It is a typical RAFM steel developed in Japan, which was developed to reduce induced radioactivity by replacing Mo and Nb with W and Ta, among the main additive elements of modified 9Cr-1Mo steel. Dose dependence of tensile properties of F82H was investigated considering two prediction functions, e.g., Power and Logarithm, and two distribution models, e.g., Normal and Weibull. It is found that the 'Log-Normal' distribution gave better predictions for 0.2% proof strength and tensile strength (Figure 2). By contrast, the 'Power-Normal' and 'Power-Weibull' models gave comparable criteria for the distribution of total elongation data. It should be noted that the possibility of Bayesian inference for predicting fusion neutron irradiation effects is limited to critical neutron fluence, and beyond this threshold, it is important to obtain data from 14 MeV irradiation.

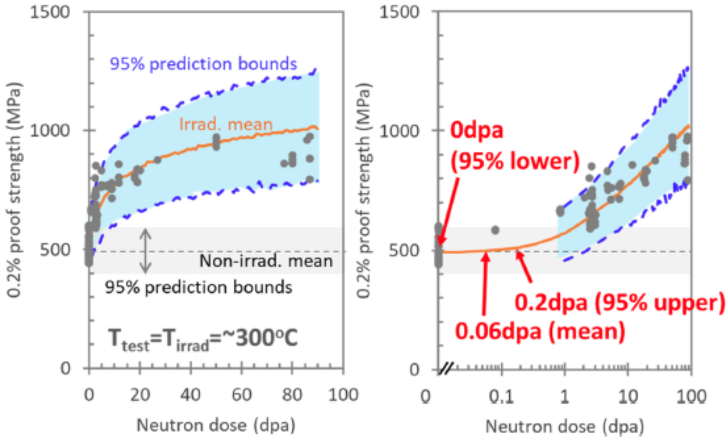

Fig.2 0.2% proof strength vs. neutron dose dependence with mean(orange) and 95% Bayesian prediction bounds(blue). Reproduced from Fig.8 in Ref. [25].

In summary, development of a database for irradiated F82H and EUROFER97 is being implemented, including irradiation experiments and statistical analyses of the data. The main achievement is a common material property handbook (MPH) for RAFM steels.

In parallel, the development of MPHs of baseline Cu alloys and W materials is being pursued. The final objective is to characterize industrially available materials – like the new rolled W products from A.L.M.T – under DEMO relevant conditions, including the effects of manufacturing processes that can lead to modifications of the as-received properties, like the Hot Radial Pressing process used to join the W monoblocks to CuCrZr pipes in the divertor. This is achieved through an extensive characterization program for both materials, including thermo-physical and mechanical properties both before and after irradiation. . The main achievement is a common MPH for divertor materials to the DEMO SDC.

3.2.2 Small Specimen Test Techniques (SSTT)

Both EU and JA pursued the development and validation of SSTTs for tensile, creep, fatigue, and fracture toughness properties, in alignment with the IAEA coordinated research project on "Testing and Standardization of Small Specimen Techniques for Fusion Applications". JA has formally agreed to participate in the EU-led interlaboratory study on miniature tensile test standardization under the ASTM framework, with QST serving as a contributing laboratory. In addition, more advanced miniature specimen tests such as multi-axial creep and brittle/ductile fracture were proposed.

3.2.3 Multiscale modeling of irradiation effects

Multiscale modeling of irradiation effects was pursued, employing atomistic and meso-scale simulations benchmarked against experimental data. A mechanism model incorporating cavity nucleation phenomena targeting He cavity formation was considered, identifying the effects of irradiation parameters on void swelling. We obtained physical insights contributing to the advancement of irradiation material microstructure evaluation models, including the statistical properties of atomic ejection phenomena and the thermal stability of He and H bubbles. Focusing on the He cavity formation in the irradiation fields, quantitative data on the He effect on void swelling and peak behavior were obtained.

3.2.4 Structural Design Criteria (SDC) and statistical evaluation strategies

The project pursued two complementary approaches: Europe's deterministic method aligned with established nuclear codes (e.g., RCC-MRx, ASME), and Japan's probabilistic approach utilizing statistical techniques (Bayesian inference, MCMC, bootstrap methods) to address data uncertainties. The development of DDC-IC (DEMO Design Criteria for In-Vessel Components) is being pursued in the EU with particular focus on the inelastic assessment route, supported by structural validation tests of fusion material to characterize critical IVC failure modes like exhaustion of ductility, multi axial fatigue, creep-fatigue interaction, fatigue crack growth and mixed-mode fracture. This dual strategy ensures that both conservative and probabilistic methodologies can be applied depending on the availability and quality of the underlying data.

3.3. T3: Neutron irradiation experiments of Breeding Functional Materials (BFMs)

3.3.1 Neutron irradiation experiments for PIEs at BR2 reactor (SCK·CEN)

The final design of capsules for neutron irradiation of Li ceramic samples (JA and EU) (including Gd and Pt shielding and chemical compatibility and calculation of burn-up of Gd based on the previous feasibility studies in 2023) have finished. Accordingly, the neutron irradiation of Be relating materials started in April 2024 and continued during the year. Discussion on the preparation, transportation and PIEs for (2025) is under discussion. The transportation of irradiated EUNM and JPNM samples will be organized in the same container and at the same time because the irradiation periods for EUNM and JPNM samples in BR2 reactor were practically synchronized. According to a preliminary timeline for the PIE of irradiated EUNM samples, it can start in the beginning of 2026 depending on a delivery time of the container with irradiated samples from SCK·CEN to KIT. Therefore, the main PIE results on irradiated EUNM samples can be obtained by the end 2026.

3.3.2 In-situ T release experiments at WWR-K reactor (INP)

After some preparatory tests and technical considerations, the Tritium detection and disposal systems were installed, and several rig designs have been suggested based on the thermal calculations. By using the most proper concept among several concepts, the mock-up tests (gas supply system, TUS, etc.) were implemented and following by the final rig and capsule designs and fabrication and assembles of the final design were finished. Preparation for irradiation experiments has been continuously conducted and irradiation experiments by using the final concept started in December 2024.

3.4. T4: R&D on Development of Material Corrosion Database

3.4.1 Development of corrosion handbooks

To clarify the effects of fusion-relevant environments on the corrosion characteristics of RAFM, e.g., magnetic fields, hydrogen peroxide generated because of water radiolysis, etc., we have advanced equipment preparation and conducted impact assessments to obtain the key initial results for DEMO design. It is noteworthy that the amount of ACP migration outside the blanket system is decreasing. Characterization of corrosion layers in the EUROFER-water system with different pH moderators and different water chemistry was carried out. It was found that both LiOH and KOH are effective in controlling the pH and the addition of Zn was negligible on the overall Fe dissolution. On the other hand, a addition of H_2 is needed to control the concentration of oxidizing species and suppress the effects of radiolysis.

3.4.2 Development of Activated Corrosion Product (ACP) evaluation model

Concerning the T permeation behavior from high temperature, high-pressure tritiated water in RAFM, we identified the effect of an oxide film on the permeation rate using deuterium. In parallel, we developed a high temperature, high-pressure tritiated water permeation test apparatus, recently providing the permeation rate as one of the important parameters to facilitate the ACP model.

4. SUMMARY

The main achievements of T1-1 are 1) global quantitative characterization of T distribution and content in the divertor and limiter tiles, and in erosion products, i.e. dust particles, 2) correlation of T distribution with the structure of plasma-facing components (PFC) with particular emphasis on the microstructure of erosion and deposition zones, and in the grooves of castellation, 3) quantification of T (total and specific activity) in dust particles from JET operated with the metal wall and those from JET operated formerly with the carbon wall (JET-C), 4) determination of T content in individual dust particles from JET operated with metal walls, discrimination between metals (tungsten, W and beryllium, Be), and carbon being a legacy from JET-C. Of particular emphasis is that the T content in Be and W was two orders of magnitude lower than C grains. In addition, preparation advanced for analyzing materials retrieved from JET after full D-T campaigns completed in 2023.

In T1, we pursued complementary strategies - such as different modelling approaches or design choices, and their assessment through distinct statistical and deterministic methodologies. This not only reduced duplication but expanded the scope of investigation, allowing a more effective validation of concepts and methods. The originally planned program has been extended to new areas, namely T depth profiling by chemical etching, and testing of that approach for detritiation of heavy metal wall components. This work is directed towards future DEMO needs. The updated TEP system concept can be expected to avoid complex system configurations in the original concept, and the revised ISS code enables rational design. These achievements will play an important role in designing the fuel system for DEMO.

The main achievement of T2-1 are as follows: The update of MPHs for DEMO in-vessel component (IVC) materials (RAFM, Cu-based alloys, and W) has been supported by extensive neutron irradiation campaigns conducted in parallel by JA and EU in HFIR (USA) and BR2 (Belgium) reactors. These campaigns covered industrial and lab-scale heats of RAFM steels, e.g., F82H (JA) and EUROFER97 (EU), Cu-alloys, and W, both from JA and EU. Specifically, advanced grades of W from A.L.M.T. (JA) were tested and compared under the same conditions with the reference products from Plansee (EU). The latest CuCrZr alloy produced by Yamato Gokin Co., Ltd. (JA) showed good neutron irradiation resistance up to 5 dpa, providing the possibility of the use in wider temperature ranges than expected, which gives more freedom to the design of the DEMO divertor. This parallel approach enabled not only direct cross-validation of results but also the identification of the most promising material grades for further development and potential industrial qualification.

In T2, we pursued complementary strategies, such as conducting irradiation campaigns in parallel reactors or approaching design rule development through distinct statistical and deterministic methodologies. This not only reduced duplication but also expanded the scope of investigation, allowing a more effective validation of materials and methods. In other cases, the collaboration enabled a deeper understanding of shared topics. The joint work on W and RAFM steels showed how cross-benchmarking improves confidence in determining material properties' trends. Similarly, JA's participation in the EU-led interlaboratory study on miniature tensile testing ensures consistency in SSTT development and supports future standardization. By collaborating in the areas of expertise of both the EU and JA in the characterization of the latest materials for the divertor and material development for DEMO, which are the primary objectives, we were able to acquire and organize a large amount of data. In addition, through a new neutron irradiation campaign, we were able to identify issues for DEMO and provide new guidelines for DEMO divertor design. For example, for the case of W, as a plasma-facing material, JA was able to manufacture and provide the EU with high-quality industrial-grade W with excellent heat load tolerance. Characterization of these materials in the EU led to a dramatic increase in the basic material property data. In addition, neutron irradiation data obtained for CuCrZr, which is considered a heat sink material, revealed the possibility of expanding the allowable operating temperature range of copper alloys and provided insights into their irradiation tolerance. In the field of neutron irradiation modelling, this program contributed to understanding He cavity formation and its computational modelling, mainly implemented by JA, brought significant added value to the development of irradiation-induced microstructure prediction technology from micro-scale to macro-scale, which has been mainly developed by EU.

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REFERENCES

[1] IFERC: https://www.iferc.org/wp/.
[2] ITER: https://www.iter.org/.
[3] IFMIF/EVEDA: https://www.ifmif.org/.
[4] JT-60SA: https://www.jt60sa.org/wp/.
[5] JA-DEMO: https://www.fusion.qst.go.jp/rokkasyo/ddjst/reactor/index.html (in Japanese); Y. Sakamoto et al., 27th IAEA Int. Conf. on Fusion Energy (Gandhinagar, India, 2018) FIP/3-2.
[6] EU-DEMO: https://euro-fusion.org/programme/demo/.
[7] S. Sugiyama, et al., Nucl. Fusion 64 (2024) 076014.
[8] L.E. di Grazia et al., Fusion Eng. Des. 215 (2025) 114976.
[9] F. Maviglia, et al., Fusion Eng. Des. 177 (2022) 113067.
[10] W. Chen, et al., IEEE Trans. Plasma Sci. **50** (2022) 4233.
[11] N. Asakura et al., Nucl. Fusion **61** (2021), 126057.
[12] N. Asakura et al., Nucl. Mater. Energy **35** (2023) 101446.
[13] N. Asakura et al., IFERC CSC Workshop (2022).
[14] N. Asakura et al., Nucl. Fusion **57** (2017) 126050.
[15] N. Asakura et al., Nucl. Mater. Energy **26** (2021) 100864.
[16] N. Asakura et al., Fusion Eng. Design, **206** (2024) 114595.
[17] Y. Someya et al., Nucl. Fusion 64 (2024) 045025.
[18] P. Arena et al., Appl. Sci. 2021, 11, 11592.
[19] F. Hernandez et al., FED 157, 2020, 11614.
[20] G. Zhou et al., Appl. Sci. 2023, 13, 2081.
[21] Y. Someya et al., 28th IAEA Int. Conf. on Fusion Energy (2021) TECH/2-1.
[22] J. Elbez-Uzan, et al., 2024 Nucl. Fusion 64 037001.
[23] Y. Torikai et al. Nucl. Fusion 64 (2024) 016032.
[24] S. E. Lee et al. Nucl. Mater. Energy 26 (2021) 100930.
[25] T. Nozawa et al., J. Nucl. Mater. 604 (2025) 155486.