JET Program for Closing Gaps to Fusion Energy

Xavier Litaudon on behalf of the JET Contributors*

***Abstract*— The European program foresees that a pure tritium (T–T) and an extended deuterium–tritium (D–T) experimental campaign will be carried on Joint European Torus (JET). The realization of the JET T–T and D–T experimental campaigns is an important contribution to fill major physics and technological gaps for the development of fusion energy. The JET-ITER-like wall experiment provides an ideal test bed to study in an integrated way the interplay between the plasma surface interactions and the plasma operation with the ITER plasma-facing materials (beryllium wall and tungsten divertor). Recent and significant progresses of the ITER scenarios development on JET are reviewed together with their extrapolation by assuming a 50%–50% D–T plasma mixture. The JET performance has been recovered at a plasma current up to 2.5 MA for both the ITER baseline and the hybrid scenarios. These two scenarios will be pursued at higher fusion performance by increasing the applied powers and/or plasma current in the coming experimental campaign in the conditions compatible with the W-divertor. The D–T experimental campaign will provide a unique opportunity to benchmark the ITER relevant 14-MeV neutron detection calibration procedures, neutronic codes for calculating the neutron flux, and machine activation, and to investigate the radiation damage of ITER functional materials. In view of the full tritium and D–T experimental campaigns, the JET tritium plant is being upgraded with an increase in the T-fuelling capability using different gas injection modules, with an improved T-accountancy and a new water detritiation system to fully close the T-cycle at JET.**

***Index Terms*— Fusion energy, ITER plasma scenarios, neutronics, tokamak, tritium technology.**

I. Introduction (the Gaps to Fusion Energy)

The path toward a power plant yielding economical attractive thermonuclear fusion energy should address three major critical gaps.

1) The sustainment of stable high-gain burning plasma for long duration. This requires the control of the burning plasma where the core fusion performance is optimized while integrating the edge conditions imposed by the plasma-facing components (PFCs), i.e., highly radiative regimes, where heat flux and actively cooled divertor/wall temperatures are controlled with reduced material erosion [1].
2) The development of materials and components that are simultaneously resistant to high heat flux (5–10 MW/m^2) and high neutron fluence. The continuous bombardment of high-energy 14-MeV neutrons issued from the deuterium–tritium (D–T) thermonuclear reactions leads to combined material displacement damage and helium bubble formation in the bulk structural materials that alter the material properties (e.g., embrittlement and modification of tritium permeation/retention properties).
3) The realization of a self-sufficient tritium fuel cycle. The continuous operation of the thermonuclear reactor generating 1000 MW of electric power requires approximately a level of tritium throughput of 12 kg per month. Since the estimated available tritium produced in the world is of the order 30 kg (from the CANDU fission reactors), fusion power plants should ensure tritium self-sufficiency by developing tritium breeding blankets and should master the tritium cycle in a safe manner (tritium extraction, separation, purification, and waste reprocessing). This also requires that tritium retention in plasma-facing materials is kept to a minimum to reduce the required tritium breeding ratio.

These three critical issues represent major gaps between our present and the required know-how on the route toward the development of reliable and safe fusion power reactors. These issues are addressed in dedicated experimental facilities combined with a significant effort in terms of modeling (plasma physics, plasma wall interactions, materials under neutron irradiation, and so on). With the decision of building the ITER tokamak in 2006, fusion research is entering in a nuclear era. More recently, Europe has elaborated a roadmap to the realization of fusion energy [2], in which ITER is a key facility. The European fusion roadmap sets out a pragmatic approach to achieve the goal of generating fusion electricity with a demonstration fusion power plant (DEMO), which is the only step between the ITER and a commercial fusion power plant.

In this paper, we review the contribution of the Joint European Torus (JET) program for: 1) minimizing the risks in ITER's scientific exploitation, both in the nonactive and active nuclear phases and 2) addressing the critical gaps to fusion energy. In this paper, the JET program for closing gaps to fusion energy is reviewed in three main sections structured along the three main issues briefly highlighted in this section. In Section II, the objectives and the status of the preparation of the next D–T experiment campaign (DTE2) with the JET ITER-like wall (JET-ILW) is reviewed [3]–[6]. In Section III, the validation of the neutronics codes and the foreseen tests of ITER functional materials under the flux of the 14-MeV fusion born neutrons is presented [7], [8]. In Section IV, recent developments in mastering the complete

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The author is with the EUROfusion Consortium, JET, Culham Science Centre, Abingdon OX14 3DB, U.K. (e-mail: xavier.litaudon@euro-fusion.org).

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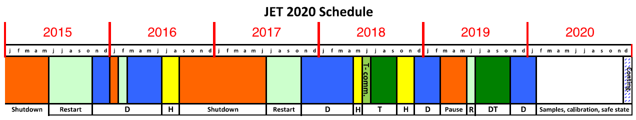

Fig. 1. JET 2020 time schedule (pending decisions on funding EUROfusion and the JET Operating Contract in 2019–2020).

tritium cycle, including tritium retention, tritium handling, and tritiated waste reprocessing in JET, are presented.

II. JET Deuterium–Tritium Integrated Operation With the ITER-Like Wall

The European program foresees that a D–T experimental (DTE2) campaign will be carried out in on JET in support of ITER. A previous experiment (DTE1) was carried out in 1997 with the carbon wall [9]. During the period 2014–2020 (pending decisions on funding EUROfusion and the JET Operating Contract in 2019–2020), a coherent set of experimental campaigns and shutdowns will be executed and further planned that includes the experiments covering the entire range of hydrogen isotopes (hydrogen, deuterium, and tritium), culminating in an extensive D–T campaign (Fig. 1). In this context, the next experimental campaigns are devoted to the optimization of the ITER regimes of operation in the presence of the ILW and to the optimal preparation of the D–T campaign. The JET-ILW experiments provide an ideal test bed to study in an integrated way the interplay between the plasma–surface interactions and plasma operation with the ITER plasma-facing materials (beryllium wall and tungsten divertor) [10], [11]. After an initial experimental phase, focusing more on the characterization of the ILW (reduction of the retention, material migration, dust production, and so on) [5], [12]–[14], JET program is moving toward the full development of ITER plasma scenarios at high fusion performance within the constraints imposed by the new wall and the inertial cooled PFCs [15]–[20]. The D–T campaign will provide an integrated test of ITER scenarios with the ITER material mix, including the isotope effects on confinement, L–H threshold, ELM, and pedestal physics [3], [20].

The high-level scientific objectives of DTE2 in terms of physics and technology are stated as follows [21].

1) Develop the ITER baseline ELMy H-mode and hybrid scenarios at full fusion performance with the ILW: study the core, edge physics, and plasma-wall integration, assess the transferability of the developed scenarios when varying the deuterium isotope during deuterium (D), hydrogen (H), tritium (T), and DT campaigns.
2) Investigate the hydrogen isotope effects (D, H, T, and D–T) on the core and pedestal confinement (check the mass dependence, scaling or not?), on the L to H transition, and on Edge Localised Modes (ELM) physics.
3) Characterize the T-cycle in a complex tokamak environment, the T-fuelling, the T-retention and migration, the T-recovery, the T waste management and detritiation process, the dust production during T, and D–T experimental campaigns.
4) Investigate the alpha-particle physics and the access to thermonuclear burn, including fast alpha-particle effects on impurity transport, on thermal and particle confinement (fast particle effects on microstability and plasma turbulence) and on macrostability during the D–T campaigns.
5) Exploit the high neutron flux to test ITER structural materials under the fusion born 14-MeV neutrons and validate the neutronics codes and nuclear data applied in ITER nuclear analyses.

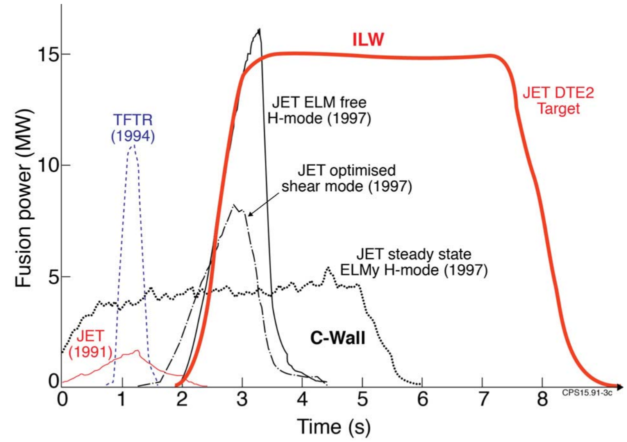

Fig. 2. Achieved fusion power in TFTR (1994) and in JET DTE1 in 1997 [9] with the C-Wall and projected goal for JET DTE2 with the ILW.

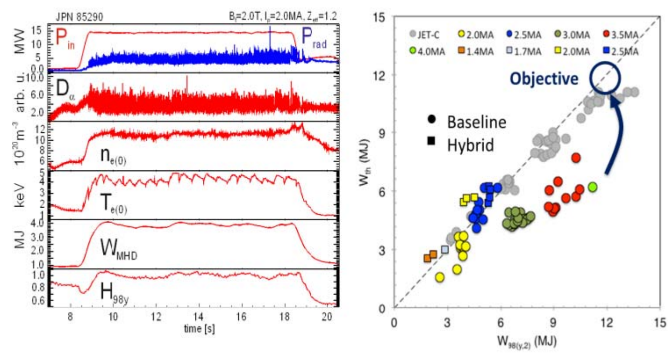

Fig. 3. Left: time evolution of a stationary ELMy H mode scenario at 2 MA with $H_{\mathrm{IPB98(y,2)}} \sim 1$. Right: thermal energy content versus the predicted IPB98(y, 2) energy content for the C-wall and the ILW [17].

To reach these high-level objectives, the development of operational scenarios based on the conventional and hybrid ELMy H-mode should provide fusion power in a JET D–T experiment in the range of 10–15 MW in stationary conditions ($\approx$5 s) representing a fusion energy between 50–75 MJ, as shown in Fig. 2.

A. Development of ITER Scenarios With the ILW

The development of stationary scenarios [Fig. 3 (left)] with the ILW requires maintaining the W-concentration at a sufficient low level, typically below $\sim 5 \times 10^{-5}$, to avoid W accumulation that leads to the discharge termination with a core radiative collapse when the radiated power exceeds the

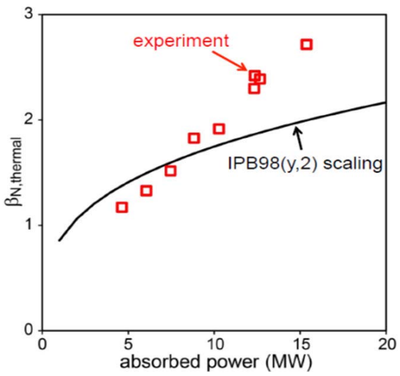

Fig. 4. Thermal normalized beta $\beta_{N,\text{thermal}}$ (open squares) versus the absorbed power in the dedicated unfueled power scan performed in scenario with on-axis q close to unity (constant current/field, 1.4 MA/1.7 T, shaping, density). Full line is the predicted $\beta_{N,\text{thermal}}$ from the IPB98(y,2) [19].

injected power [15], [17]. The control of the W-concentration is achieved by: 1) increasing the level of neutral deuterium gas fuelling at the plasma periphery to reduce the W-source; 2) increasing the ELM frequency to regularly flush the W impurity; 3) increasing the core electron heating thanks to minority ion cyclotron resonance heating; and 4) controlling the divertor surface temperature via strike points sweeping or/and extrinsic impurity seeding. These operating conditions imposed by the metallic wall materials lead to a significant reoptimization of the plasma scenarios to reach similar level of fusion performance and thermal energy content as previously observed for similar operational parameters (toroidal field, plasma current, and applied power) but with the C-wall. Indeed, it was found for the ITER baseline scenarios ($q_{95} \sim 2.7$–3 and $\beta_N \sim 1.8$–2) that efficient pumping conditions, with the strike points of the magnetic configuration close to the divertor pump duct entrance leading to a reduction of the divertor neutral pressure, are required to recover the fusion performance with an enhancement confinement factor close to unity [15]–[20]. With the available applied power in the range of 26–30 MW (consisting typically of 22–27 MW of neutral beam power and 4 MW of Ion Cyclotron Resonance Heating (ICRH) power), the JET performance has been recovered up to a plasma current of 2.5 MA for both the ITER baseline ($\beta_N \sim 2$) and hybrid scenarios ($\beta_N \sim 2.3$–3, with on-axis safety factor above unity), as shown in Fig. 3 (right) [17].

It is worth noting that in plasma regimes exceeding typically $\beta_N > 2$, the thermal confinement is above the value predicted by the thermal H-mode scaling law with $H_{\text{IPB98(y,2)}} = 1.1 - 1.4$ at $\beta_N \sim 2$–3.4 in conditions of low D_2 gas fuelling rates (Fig. 4) [19]. So far, in the hybrid regime, the highest fusion performance phase at $\beta_N > 2$ is limited in duration (0.5 s) due to a deleterious interplay between the core MHD limits and the core impurity influx and as well by the divertor temperature limit and high W concentration in the core are reached in low fuelling conditions. The future challenge will consist in combining the good core confinement with the divertor constraints. Access to high confinement at high beta is consistent with an increase of the pedestal pressure consistent with peeling/ballooning modeling leading to higher core electron temperature and lower core collisionality resulting in a higher electron density peaking. The plasma core energy is further increased due to the reduction of the core plasma turbulence at high beta associated with the presence of fast ions [22], [23] and a high rotational ExB shear. The relevance of this promising effect in burning plasmas needs to be investigated experimentally during JET DTE2. In addition, this virtuous cycle that leads to high predicted fusion performance should be included in the time-dependent modeling for the prediction of the fusion performance in JET, ITER, and DEMO.

As part of the scenario development to increase the fusion performance, it is essential that the attention is given to minimize the occurrence of disruptions. That is applying disruption avoidance techniques, reducing the plasma disruptivity and, thus, minimizing the consequences to the device [24], [25]. In this context, it is recalled that inadequate disruption mitigation is the highest programmatic risk in the ITER research plan. With the ILW, lower radiation and, hence, higher plasma temperatures are observed during unmitigated ILW disruptions compared with C-wall. The resulting slower current quench induces larger vessel forces and causes greater heat loads [26], [27]. On the other hand, the longer current quench rates and, thus, the lower toroidal electric fields have made it more difficult to generate runaway electron beams with ILW [28]. In this context, mitigation by massive gas injection (MGI) became a necessity for ILW operations at plasma currents higher than 2 MA [28], [29]. Active mitigation requires, together with appropriate avoidance schemes, the development of reliable real-time detection/prediction of disruption, which activates the MGI system. Recent efforts have developed predictors that, taking the advantage of the statistical learning methods, have a very high prediction performance (>90%) [30], [31].

B. Impurity Seeding to Reduce the Divertor Heat Load

For future burning plasmas in ITER and in a fusion reactor [1], [32], a high divertor radiation level (70%–90%) will be required to maintain the heat flux on the divertor at an acceptable level ($\sim$10 MW/m^2), i.e., to avoid thermal overload of materials and minimize the material erosion in severe conditions. This issue drives the need to characterize and understand the conditions for divertor detachment, and the effect of extrinsic impurity seeding in the core plasma in different scenarios, and with different machine sizes, such as ASDEX-Upgrade [33], [34] and JET [35], [36] with W divertor.

To address this critical issue for the development of fusion energy, dedicated impurity seeding experiments have been performed on JET [34], [35] to approach power plant like divertor conditions with detached divertor and to complement recent ASDEX-Upgrade experiments [33], [34]. On JET, it was found that the total radiated power fraction, f_{rad}, reached a maximum value of 75% independently of the level of applied powers (below 30 MW) and type of extrinsic impurity (N_2, Ar, Ne, or a mixture of Ne and N_2), as shown in Fig. 5 [35], [36]. The reason of the observed limitation is not yet understood and this is a subject of dedicated modeling activities.

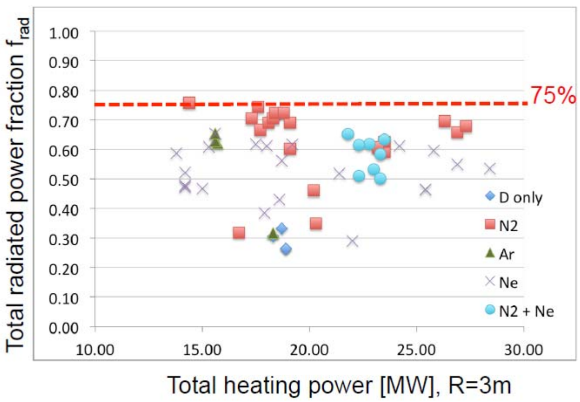

Fig. 5. Total radiated power fraction, f_{rad}, versus the total applied power for different injected impurities [35], [36].

The highest values of f_{rad} were obtained with only N_2 seeding, where stable completely detached (outer and inner) divertor conditions have been achieved. These highly radiative regimes have been obtained in the conditions of enhancement confinement factor in the range of 0.7–0.8. The important challenge remains to combine highly radiative operation with good thermal confinement. It will also be important to check that the observed f_{rad} limitation remains at higher power in the range of 30–40 MW.

C. Present Status and Prospects for D–T Operation

One of the main objectives of the next deuterium campaigns is to further extend the performance of the ILW at higher plasma current (>2.5 MA) by fully exploiting the JET machine capability at high additional powers in the range of 40 MW with up to 34 MW of neutral beam injector (NBI) power and 6-MW ICRH (the ICRH ITER-like antenna has been reinstated during the 2014–2015 JET shutdown). The quantitative objective is to reach deuterium plasma scenarios with thermal energy content of the order of $W_{\text{th}}(\text{DD}) \sim 12$ MJ with $H_{\text{IPB98(y,2)}} \geq 1$ [Fig. 3 (right)] generating a stationary fusion D–D neutron rates of $R_{\text{DD}} \sim 6 \times 10^{16}$ n/s during 5 s.

To reach these objectives, two main approaches will be pursued: 1) the ITER baseline scenario by simultaneously increasing the current, toroidal field, and applied powers at $q_{95} \sim 3$ and $\beta_N \sim 1.8$–2 and 2) the ITER hybrid scenario at slightly reduced plasma current and higher $q_{95} \sim 3$–4 but at $\beta_N > 2$ to enter in the virtuous cycle where confinement is increased at high beta through the interplay between the core and edge confinement optimization, where fast particles may play a key role as previously reported (Fig. 4) [19]. The challenge will be to reach and sustain the fusion performance while not exceeding the temperature conditions imposed by the inertially cooled W divertor with acceptable low W concentration in the core.

The status of the maximum fusion performance achieved at the end of the 2014 experimental campaign for the two scenarios is summarized in Table I [37]. The fusion power projections in D–T plasmas do not include the alpha power and the isotope effect on confinement. The D–T projections of

TABLE I

D–T PROJECTION OF THE MAXIMUM D–D FUSION PERFORMANCE FOR BOTH THE BASELINE AND HYBRID SCENARIO AS ACHIEVED AT THE END OF THE DEUTERIUM 2014 EXPERIMENTAL CAMPAIGN, AND, AS EXTRAPOLATED AT HIGHER APPLIED POWERS, FIELD, AND CURRENT. THE D–T PROJECTIONS DO NOT INCLUDE ALPHA POWER AND ISOTOPE EFFECTS ON CONFINEMENT [37]

	Baseline		Hybrid	
	2014	DT projection based on present plasma performance	2014	DT projection based on present plasma performance
I_p (MA)	3.2	4.5	2.5	2.5-3.0
B_t (Tesla)	3.3	3.85	2.9	2.9-3.45
q_{95}	3	2.8	3.7	3.7
P_{NBI} (MW)	27	34	22	34
P_{ICRH} (MW)	4	5	4	6
H	0.8	0.8	1.1	1.1
Duration (s)	2	5	0.5	5
P_{fus} (MW)	3.2	7.5	5.5	10-13

the presently achieved highest D–D neutrons rates scenarios predict, respectively, 3.2 and 5.5 MW of fusion power for the baseline and hybrid scenarios, i.e., a factor of 2–3 with respect to the high level D–T objective previously discussed. In addition, D–T projection of the expected fusion performance has also been calculated based on present plasma scenario but assuming that the experimental profiles and confinement enhancement factors could be translated at higher field, current, and applied powers (i.e., temperature and density profile shapes are kept constant with a fixed ratio of the density normalized to Greenwald density when increasing the plasma current). With these assumptions, the predicted fusion performance is 7.5 MW for the baseline scenario (at $H_{\text{IPB98(y,2)}} = 0.8$) at high current (4.5 MA) and 13 MW for the hybrid regimes with 40 MW of applied power at lower current (2.5–3 MA). The main challenge is to increase the confinement at higher field/current and applied powers, while reaching stationary conditions compatible with the W divertor [38].

III. NEUTRONICS AND CODE VALIDATION

The D–T campaign will provide a unique opportunity to benchmark ITER relevant 14-MeV neutron detectors calibration procedure and neutronics codes for calculating the neutron flux and machine activation and to investigate radiation damage mechanisms of ITER functional materials [7], [8].

A. 14-MeV Neutron Detectors Calibration

The 14-MeV neutron rates issued from the D–T fusion reaction should be accurately measured for the scientific exploitation of the D–T experiment (code validation and performance optimization), for the accurate measurement of the fusion power (including tritium burn and tritium accountancy), and, for a precise estimate of the machine activation within the available neutron limit. On JET, an accurate calibration of the neutron detectors at 14-MeV neutron energy

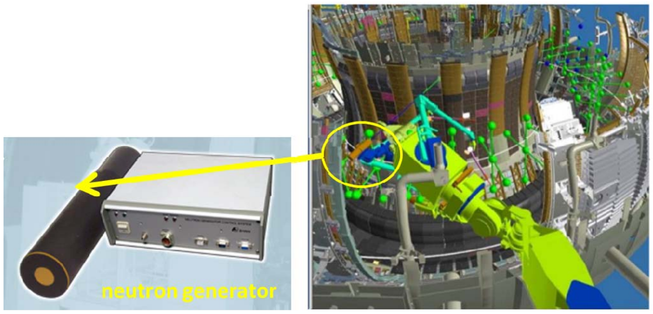

Fig. 6. Sketch of the neutron generator to be mounted on the remote handling MASCOT robotic arms that will be deployed inside the JET vessel.

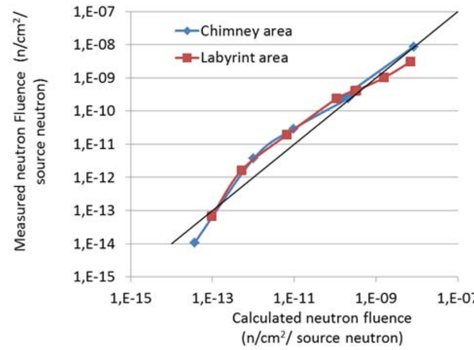

Fig. 7. Comparison between measured and calculated neutron fluence along two penetrations through the JET torus hall biological shield: the access labyrinth and the air duct chimney [8].

(^{235}U fission chambers and the in-vessel activation system) will be performed using a D–T neutron generator to be deployed inside the JET vacuum vessel by remote handling [39]. The JET calibration procedure will also provide a benchmark of the procedure envisaged in ITER, where the fusion power should be measured within less than 10%. The deployment of the D–T neutron generator (10^8 n/s) inside the JET vacuum chamber is foreseen in different toroidal and poloidal positions (Fig. 6). The operation time required for the neutron generator in each position is in the range of 0.3–4 h during which the neutron emission rate by the generator will be monitored by *ad hoc* monitoring detectors. As a calibration source, the neutron generator must fulfill some minimal characteristics related to sufficient neutron source intensity, stability, lifetime, and simplicity of configuration. Precise characterization of the neutron generator will be performed prior the in-vessel calibration. The characterization tests will be performed in a standard neutron laboratory, well equipped to perform such characterization.

B. Neutronics: Code Validation for ITER

Dedicated neutron measurements around JET are performed to validate the various codes used in ITER to predict quantities, such as the neutron flux along streaming paths, the activation of materials, and the resulting shutdown dose rates. Validation of the neutron streaming calculation in large and complex fusion facility such JET is an important step to gain confidence in the safety assessment calculations made for ITER.

For this purpose, several streaming experiments have been carried out, consisting of measuring the neutron fluence and dose rates in the torus hall and along its ducts. It was demonstrated that the neutron fluence measurements along the penetrations of the JET torus hall biological shield are well reproduced by the Monte Carlo codes over six orders of magnitude of the neutron fluence, as shown in Fig. 7 and reported in [8].

Shutdown dose rates have also been calculated with various codes (Advanced D1S, R2Smesh, MCR2S, and R2S-UNED). A good agreement is found among the codes that reproduce the measured gamma dose rates along the midport and in several cells outside the vessel another set of neutron fluence and activation measurements is foreseen during the 2015–2016 D–D campaign. These measurements will be

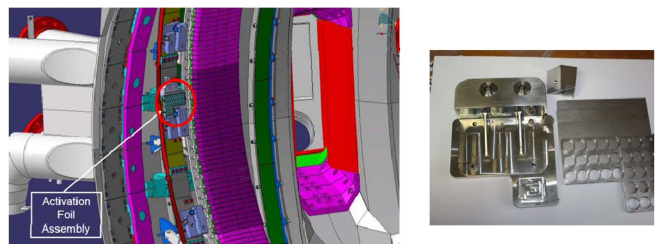

Fig. 8. Sketch of the long term irradiation station (left) and one of the sample holder (right) to be installed inside JET close to the first wall.

repeated during the DTE2 to further validate the neutronics codes with experimental data obtained from the streaming of 14-MeV neutrons and activation of materials.

C. Materials Neutron-Induced Activation and Radiation Damage Studies for ITER

The JET DTE2 campaign will also provide an important opportunity to investigate the radiation damage of some selected ITER functional materials. Indeed, JET 14-MeV neutron fluence of 10^{20} neutrons/m^2 on the first wall is of the similar order of magnitude as the one expected for the functional materials in the ITER environment in the rear part of ports. The materials are the ones to be used in ITER diagnostics and heating systems. The effects of 14-MeV neutron radiation on the physical properties of the functional materials used in ITER, such as cables and windows (e.g., optical and dielectrical properties), need to be investigated in conditions as close as possible to the one expected on ITER in terms of neutron flux, fluence, and 14-MeV neutron energy spectrum. The JET DTE2 campaign will constitute an ideal opportunity to investigate the evolution of the material properties under real tokamak environment. In this context, two in-vessel long-term irradiation stations with dedicated sample holders will be installed in JET by remote handling (Fig. 8).

These holders will also host samples of structural materials being used to manufacture ITER in-vessel components to investigate the 14-MeV neutron-induced activation and validate the activation predictions in ITER. A list of potential ITER real materials to be investigated was agreed in 2014. Fusion for energy will provide the samples of such

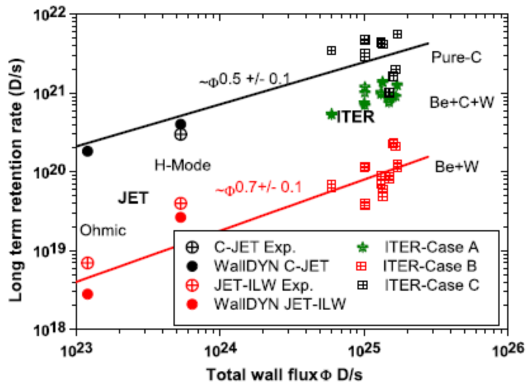

Fig. 9. Measured and calculated long-term retention rate versus the total wall flux for the JET–C wall and ILW. Calculated retention rates for ITER with a C wall and with a Be wall and W divertor [43].

ITER-grade materials as well as the chemical composition, including impurities.

IV. Tritium Cycle

JET is the only tritium compatible magnetic confinement fusion device currently active. JET DTE2 campaign will require using 60 g of tritium in the active gas handling system (AGHS) plant, while the total releasable tritium inventory allowed outside the tritium plant is limited to 11 g on the various cryopanels and to 4 g on the mobile in-vessel inventory.

A. Deuterium and Tritium Retention in Metallic Walls

Both postmortem and gas balance analyses have concluded on a significant reduction in ASDEX-Upgrade and JET of the deuterium retention with metallic walls compared with C walls [13], [40]–[43]. The remaining retention is still dominated by codeposition with residual C and boron from boronization in ASDEX Upgrade and intrinsic Be in JET. More recently, it has been shown that the WallDYN simulations could reproduce the retention reduction (Fig. 8) and the migration pattern with both the JET-C wall and ILW [43]. WallDYN couples the state-of-the-art models for the surface processes (e.g., erosion, reflection, implantation, and sublimation) with material redistribution data from trace impurity plasma transport models in a fully self-consistent simulation [43]. Applying the same model and process physics as for the JET calculations, the impurity migration and resulting fuel species codeposition in ITER for different wall configurations and background plasmas have been calculated. The simulations show that the ITER T-limit of 700 g is reached with only 100 to 700 full 400 s ITER discharges with a C-wall, whereas for the ITER material choice (Be wall and W divertor), 3000 and 20 000 discharges are possible depending on the plasma scenario (Fig. 9) [43].

In order to get a better understanding of the tritium behavior in JET and complement the investigations of the fuel retention through gas balance of the postmortem analysis, dedicated laboratory studies will be carried out on tritium short term permeation, outgassing rates versus the T content, retention on Be and W samples. In this context, due to the different mechanisms involved in tritium transfer (tritium ions implantations and tritium gas permeation), two facilities are being designed, will be built and run in the CCFE Material Research Facility-based at Culham close to the JET facility. Both preliminary designs are being undertaken of the: 1) tritium loading facility, aiming to reproduce tritium ion implantation in the in-vessel materials and 2) the tritium permeation facility, aiming to simulate tritium gas permeation from the regeneration of JET cryopumps. The experimental data will be used to validate the tritium diffusion and permeation codes before ITER extrapolation. The objective is to provide a better understanding of the relevant mechanisms with experimental tests on samples at different temperatures and loading conditions, for selected material, such as Be/W tiles and in-vessel structural materials.

B. Tritium Fuel Cycle and Detritiation

The AGHS was constructed to process and recycle the gases from the torus and NBIs [44], [45]. The AGHS is located in a separate building equipped with its own ventilation system and connected with the JET torus via gas transfer and pumping lines. For JET DTE2, the AGHS will be further upgraded with the possibility of using different gas injection modules (GIMs), with an improved T-accountancy and a new water detritiation system to fully close the T-cycle at JET [44], [45]. The valuable experience gained both in this preparation and in the execution of the JET DTE2 campaign is contributing to train the future ITER operator of the tritium plant in terms of operation and maintenance and will provide quantitative information for the nuclear regulator (e.g., radioactive waste management contaminated with tritium).

For DTE2 campaign, it is foreseen a factor 16 increase in the tritium throughput and feed requirements on the AGHS compared with DTE1 performed in 1997. The present tritium inventory in the AGHS is ∼6 g and the first batch of 7.5 g of tritium for the T, and D–T campaigns has been received in spring 2015. The site inventory will be up to 60 g of tritium and will be reprocessed up to 16 times by the AGHS, making the equivalent of up 1000 g permitting a significant number of tritium (750 pulses) and D–T (∼80 pulses) experiments. During the 2016 experimental campaign, the AGHS plant will be exercised with deuterium only and the procedures will be fully reviewed and assessed well before the T and D–T campaign.

An upgrade of AGHS is being undertaken to enable four existing GIMs, instead of one in DTE1 to be fed by the AGHS with pure tritium. This upgrade will ease the transfer from deuterium to tritium plasma scenarios initially developed using different GIMs and will provide a better control of the D–T mix within the JET plasmas.

In addition, before the next tritium campaign on JET, there is a requirement to improve our tritium accountancy techniques to allow the quantities of tritium being transferred to, and returning from, the JET machine to be accurately traced at a level below 1%. The first steps have been made to improve the tritium accountancy with an upgrade of the instrumentation

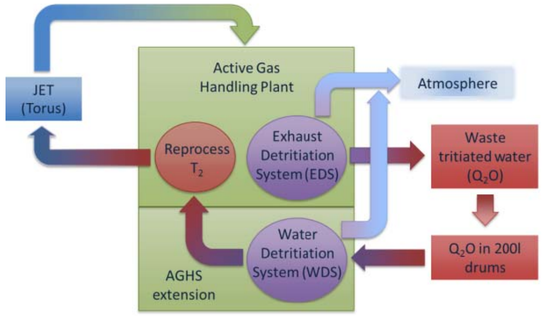

Fig. 10. Sketch of the JET tritium cycle with the new water detritiation system where T waste will be reprocessed and recovered to close the on-site tritium cycle [45].

for T measurement, e.g., by the development of a solid-state-based detector for tritium.

The AGHS exhaust detritiation system produces tritiated water at a rate averaging 9200 L peryear. This is collected in a tank and stored in stainless steel drums. These drums have historically been shipped to the Ontario Power Generation Inc. in Canada for disposal and recycling. It has been decided that an on-site water detritiation system will be constructed to allow the processing of tritiated water and the recovery of contained tritium. The tritium recovered from the water will be fed back into the AGHS plant for processing (Fig. 10). The water detritiation system is designed to have an annual throughput of ∼15 000 kg with activities of up to 200 GBq/L. It is planned to start nonactive commissioning of the water detritiation system by 2016.

C. Tritium Compatible Pumping System for DEMO

It is foreseen to test at JET during the DTE2 campaign a novel and innovative torus vacuum pumping system that is being developed for DEMO [46]–[48]. The objective is to demonstrate the feasibility of a continuous, noncryogenic, T compatible pumping system. The new pump should overcome the T inventory issues in long pulse operation. The new system comprises three stages, one of which is a liquid ring roughing pump that uses mercury as working fluid [47], [48]. JET will provide a unique opportunity to test and validate this pumping solution under tritium operation and fusion relevant conditions. In this context, a fully tritium compatible rough pumping system, already in DEMO relevant scale, will be installed in the AGHS and operated during DTE2.

V. Conclusion

The exploitation of JET with the ILW, which mimics the ITER first-wall material configuration (Be wall and W divertor), provides a major contribution to the ITER design and operation. In addition, it provides an ideal environment to validate the predictive capabilities of the engineering and physics codes for ITER application and to test new technological developments. European scientists and engineers are preparing the future JET T–T and D–T experimental campaign. We have reported that the preparation, exploitation, analysis, and modeling of the JET D–T campaign require to address and to solve, in an integrated manner in terms of physics and technology, crucial issues for the development of fusion energy, in view of the following.

1) the development of ITER scenarios by integrating ILW constrains with impurity seeding or *X*-point sweeping, the SOL physics, pedestals with the core fusion performance optimization (low W concentration) in a relevant D–T mixture and burning plasma conditions.
2) the validation of neutronics codes in the real tokamak environment for future ITER application and for providing essential information to the safety regulators; the integrated tests of ITER functional materials under relevant 14-MeV neutron fluxes and energy spectra;
3) the mastering of the full tritium cycle in a real tokamak environment and the trititiated waste reprocessing.

Progress along these lines of research is a direct contribution to fill the three major gaps for the development of fusion energy as reviewed in the introduction of this paper. Since the installation of the JET-ILW in 2011, JET has made significant contribution in the characterization of the ITER wall and has recovered the fusion performance previously achieved with the C-wall at a plasma current up to 2.5 MW with typically 30 MW of additional powers. Important scientific challenges will be addressed in the following experimental campaigns:

1) to improve the integrated fusion performance with the ILW for D–T operation;
2) to achieve stationary operation at full fusion performance in condition compatible with the ILW;
3) to increase our level of understanding of plasma operation with ILW and its implication for ITER operation and DEMO design.

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Xavier Litaudon was the Head of the Heating and Confinement Department with the Institute for Magnetic Fusion Research, CEA, Cadarache, France, from 2010 to 2014, the Leader of the ITER Scenario Modeling Group from 2010 to 2013, and the Leader of the Joint European Torus (JET) Task Force from 2004 to 2007. As a Young Physicist, he was part of the 1997 Deuterium-Tritium Campaign at JET. He has been the Head of the ITER Physics Department with the EUROfusion Program Management Unit since 2014 and he is based with the JET Facility Site, Culham Science Center, Abingdon, U.K. He is in charge of coordinating the European physics program in support of ITER: JET program, physics program of European Medium-Size Tokamaks, preparation of scientific exploitation of JT-60SA, preparation and exploitation of W-7X campaigns, and coordination of the modeling activities. The European fusion program is based on the Roadmap to the realization of fusion energy. His field of expertise is thermonuclear fusion, tokamak physics, physics and technology of heating and current drive systems, integrated tokamak modeling, plasmas operational scenarios for long pulse and steady-state regimes, and plasma confinement and real time control.

Mr. Litaudon is one of the EU members of the Coordinating Committee of the International Tokamak Physics Activity and he is a member of the Editorial Board of *Plasma Physics and Controlled Fusion*.