Annex 4

EUROfusion Engineering Grants AWP2022 (FP9-1)

List of positions

Contents

EEG21-01 DEMO nuclear waste management

Contact persons: Maria Teresa Porfiri (Project Leader WPSAE) – mariateresa.porfiri@enea.it

Background:

The breeder blanket, divertor and vacuum vessel are the components exposed to a high neutron fluence in a DEMOnstration Fusion Reactor. The structural materials that are foreseen for these components are austenitic stainless steel AISI 316 for the vacuum vessel and reduced-activation ferritic martensitic steel (EUROFER) for the breeding blanket and divertor. In addition, other materials such as tungsten and copper-alloys are used for the plasma facing components (PFC), ceramic or liquid metal breeders (LiPb) and beryllium as multiplier are used for the breeding blanket. The in-vessel components (breeding blanket and divertor) will be replaced according to a maintenance plan. The vacuum vessel is a permanent component and is supposed to work for the whole plant life.

Objectives:

The study will have the scope to perform a waste management study starting from the current DEMO material activation data. It will include the handling of the components, their packaging and transfer / transport to processing facilities (e.g. via cask), processing and storage/disposal. This will be done for the operational waste, the refurbishment waste (exhausted components), the waste in consequence of Design Basis Events and accidents and the decommissioning waste.

Among operational waste, particular attention should be given to tritium, which is not strictly a waste as an element but requires treatment and conditioning processes of all tritiated materials favouring tritium recovery.

Identification and separation of waste materials with low activation or contamination, suitable for recycle will be a key part of this work because minimization of long-lived activation products is an important objective for DEMO.

Tritium recovery from these components is important for economic, safety/environment and waste minimization reasons. Means of tritium recovery have to be assessed to determine those most applicable to DEMO for the tritiated streams identified. An important early step to define such streams will be to review the bounding Design Basis Events including accidents so that the waste consequences can be evaluated. For this, it will be important to examine the impact that safety measures may have on waste production such as the use of water, which will increase the demands on the management of tritiated water and other waste. Technical and economical solutions will be sought throughout, for example in elaborating the needs for increased tritiated water storage and clean up.

Competence development:
The candidate will be required to become proficient in waste management and its application to large-scale fusion plants (DEMO). He/she has to gain knowledge of and/or experience in the following main areas:

- becoming fully acquainted with the relevant regulatory codes and standards especially the requirements for classification, clearance and recycling, transport and disposal of radioactive wastes.
- developing waste processes to address the waste types and quantities identified, including the assessment of secondary waste.
- gaining experience with relevant computer codes e.g. those for neutron activation, calculating waste generated in accident sequence, and tritium recovery.
- understanding the experimental work in support of the above computer code calculations and identifying any gaps where this should be strengthened for application to this work. Of particular importance are the opportunities to recycle metals used in the DEMO reactor and future power plants. The work to be carried out will aim to identify and focus on developing solutions to these issues.
- working with experts in the DEMO design to assist in determining feasibility of proposed approaches and in particular their integration into the DEMO design,
- developing and pursuing contacts with other specialists at laboratories and industries.

Competences required:
Engineers holding a Master Degree

Work packages involved:
WPSAE (Primary), DEMO WPs

Facilities to be used:
N/A

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. This project foresees multiple stays in the DEMO Central Team in Garching.

EEG21-02 Design of DEMO Magnet Systems

Contact person: Valentina Corato (Project Leader WPMAG *), valentina.corato@enea.it
** on behalf of the PSDD Head of the DCT*

Background:
The Toroidal Field and the Poloidal Field magnet systems, including the Central Solenoid, are the core of the DEMO machine and must be designed to satisfy the system requirements and operating environment in compliance with the design criteria. Because of the limited possibilities in remote maintenance and replacement of coils, a long-term reliability of magnets at full performance is required. This must be achieved by designs based on relevant experience, comprehensive analyses, and component verification in a well-coordinated development program.

Objectives:
The objective of this call is to provide the suitable candidate with (a) sufficient engineering skills both for the coarse dimensioning of the magnet structures in operating conditions and for refined analyses (e.g., FEM) to assess the detailed designs; (b) the ability to develop design concepts to satisfy manufacture, assembly and maintenance requirements.

Competence development:
In particular, the candidate will develop his/her skills on:

- Design of magnet systems, including conductor and winding pack design. Proposals of design concepts to satisfy manufacture, assembly and maintenance requirements.
- Scoping analyses, using crude representations of the magnet system but considering a range of major geometric variations and magnet concepts.
- Detailed electro-magnetic, structural calculations for various load cases, to look in depth at the most promising concept.
- Design of magnet support and inter-coil structures, including stress calculations.
- Fault analyses, to look at the magnet behavior under fault conditions that may act in some areas as design drivers.

Competences required:
Engineers holding a Master Degree.

Work packages involved:
WPDES (Primary), WPMAG, WPPES

Facilities to be used: N/A

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. This grant foresees multiple stays in the DEMO Central Team in Garching. The planning and budget should be aligned with the relevant Project Leader.

EEG21-03 DEMO RAMI Analyst

Contact persons: Sergio Ciattaglia (PMU) – Sergio.ciattaglia@euro-fusion.org

Background:

The Demonstration Power Plant (DEMO) will be a prototype of fusion reactor designed to prove capability to produce electrical power in a commercially acceptable way. Then, key elements of any engineering development of the reactor are definition of reliability and availability requirements (or targets), reliability and availability analysis, reliability testing, and "reliability growth", the structured process of finding root causes for reliability problems and predicting and monitoring the increase of system's reliability through successive phases. Since reliability and availability are strictly related to maintenance and inspection activities performed on the plant during the operating phases, the integrated approach in reliability and availability optimization is based on the four issues: Reliability, Availability, Maintainability and Inspectability (RAMI).

Many factors are important to achieve a satisfying RAMI level: design of systems; manufacturing quality; the operational environment; the design and development of the support systems; the level of training and skills of the people operating and maintaining the system; the availability of spare parts to repair the system; and the diagnostic aids and tools (instrumentation) available to check system processes and capability to detect normal and abnormal operating parameters. All these factors must be understood to achieve a plant with a desired level of RAMI. During Pre-Concept Design phases, the most important activity is to understand the rationale of the plant, the related functions, requirements and constraints for the different systems. During plant development, the most important RAMI activity is to identify potential failure mechanisms and to make design changes to remove them or to mitigate consequences of the failures. During realization and installation, the most important RAMI activity is to ensure quality in manufacturing so that the inherent RAMI qualities of the design are not degraded. Finally, in operations and support, the most important RAMI activity is to monitor performance in order to facilitate retention of RAMI capability, to enable improvements in design (if new plant upgrading will be foreseen), or of the support system (including the support concept, spare parts storage, etc.).

Inadequate reliability or failed failure indications of components deemed safety critical items may directly jeopardize the public and worker safety. For that reason deterministic and probabilistic safety assessments (PSA) have to be strongly integrated with the RAMI assessments.

Once reliability and safety of plant operation is assured, further objective is to obtain plant's product and/or plant's mission at minimum cost. Then, cost/benefit analysis to justify and prioritize plant changes, modifications and enhancements during design and operation have to be undertaken and have to be matched with RAMI and PSA during the overall phases of plant life.

The above relationship between the facility design process and the parallel development of the facility safety analysis lets us to set the RAMI programme in a widest context called more generically the reliability assurance programme for DEMO.

Objectives:

The objective of this EEG is to perform RAMI analyses of a few DEMO systems, particularly critical from a RAMI and safety point of view because of their complexity, the challenging operation conditions and/or for their safety functions. The RAMI analyses will provide indications on the possible design improvements in order to reach an acceptable RAMI target for the overall DEMO, meeting the safety requirements. The grantee will work with RAMI and nuclear safety specialists in the DEMO-Design Central Team (DCT) and in the Research Unit (RU) hosting the Engineering Grant to develop the analysis process required for DEMO.

Scope/Activities/Outcomes (foreseen):

Interim and final reports on the work will be written by the grantee. The work may also be presented at international fusion conferences and papers published in their proceedings. Attendance a two/three such conferences in the course of the grant will also give the grantee valuable exposure to the broader world of scientific and engineering research for fusion power.

Competence development:

The candidate should have a knowledge of nuclear reliability and safety issues, be able to identify components and their related operating functions from design documents and design drawings, be competent in scientific/engineering computing, with the capability of setting up complex computer models. In the course of the grant, the successful applicant will gain a good understanding of fusion systems and aspects of the DEMO design, will further deepen his/her knowledge of RAMI and nuclear safety principles and their application to fusion, and gain good experience of setting up and using complex engineering computer models for RAMI analyses.

Competences required:

Engineers holding a Master Degree

Work packages involved:

WPDES (Primary), WPSAE and the DEMO WPs involved by the RAMI analyses

Facilities to be used:

N/A

Mobility:

During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. This grant foresees multiple stays in the DEMO Central Team in Garching. The planning and budget should be aligned with the relevant Project Leader.

EEG21-04 A methodology for cracks tolerance assessment in irradiation embrittled EUROFER Reduced Activation Ferritic Martensitic (RAFM) Steel.

Contact person: Giacomo Aiello (FTD Materials RO) – giacomo.aiello@euro-fusion.org

Background:
EUROFER is a 9%Cr Reduced Activation Ferritic Martensitic (RAFM) steel specifically developed for application as structural material for In-Vessel components of future fusion power plants. Neutron irradiation significantly affects the mechanical properties of steels causing, among other, hardening and embrittlement. RAFM bcc steels intrinsically have a steep gradient between brittle and ductile behavior, in brief often represented by a ductile to brittle transition temperature (DBTT) as a quick measure of embrittlement. Fracture Toughness (FT), the material property to resist to crack-extension, under neutron irradiation shows generally, both, a reduction and a DBTT shift towards higher temperatures, commonly addressed in the Master Curve approach. Both phenomena strongly depend in particular on a) the neutron irradiation fluence, usually measured as displacements per atom (dpa) and b) the local irradiation temperature (history); FT depends as well as on the actual operational temperature. Irradiation-induced embrittlement, like loss of FT and shift in DBTT is very pronounced at lower operating temperatures (<350°C) and is effective even at low n-fluence.

Fusion In-Vessel components (IVC) are operated under strong n-fluence and temperature gradients as well as subsequent strong stress- and strain- gradients in few mm-cm thin structures. Through thickness and for given cracks, this leads to strong variations of, both, the local loading parameter (represented by stress-intensity or J-Integral) and the material property FT at the crack-tip. Areas of ductile or brittle material behavior are difficult to determine and criticality of a crack depends on the T-irradiation history, the (actual) load and temperature. In particular, maintenance and start-up/shut-down phases, when the components are cooled-down to ambient temperatures and then re-pressurized before operation, are identified as critical. The development and validation of new tools and methodologies, capable of assessing and predict crack behaviour such as initiation/propagation/instability or arrest under such conditions, are urgently required to review current engineering DEMO IVC designs against safe operation.

Objectives:
Applications are invited from individuals interested in the development of novel approaches in Fracture Mechanics to investigate the mechanisms of crack growth under representative loading and strong gradients in the governing material property FT at the crack tip. The work shall ideally combine experimental testing with modelling to develop in the end an integrated design tool for defects assessment and acceptability in In-Vessel components.

Scope/Activities/Outcomes (foreseen):

The selected candidate is expected to focus his/her research in the following areas:

- Development of damage models to describe the fracture process during crack propagation in neutron-irradiated materials taking into account steep gradients of FT and transition from brittle to ductile behavior.
- Development of experimental testing and evaluation procedures to calibrate and validate the damage models.
- Implementation of the models in Finite Elements codes (ABAQUS, ANSYS, etc.) to simulate crack propagation in components and assess them with respect to acceptability of defects as function of crack size, relative depth, FT and FT gradients.
- Apply the methodology to First Wall and Breeding Blanket components and conduct parameter studies on them deducing simplified criteria and rules for designing In-Vessel components against fast fracture.

Competence development:

During the course of the Grant, the candidate will gain knowledge and experience in the following main areas:

- Effects of neutron irradiation on mechanical properties of materials.
- Linear and non-linear Fracture Mechanics.
- Mechanical testing of materials.
- Numerical simulations.
- Design of nuclear fusion components.

Competences required:

Engineers holding a Master Degree. A recipient of a Master Degree or PhD in a University of Technology will also be considered as an eligible candidate.

Work packages involved:

WPMAT (primary), WPBB

Facilities to be used:

N/A

Mobility:

During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. This grant foresees multiple stays in the DEMO Central Team in Garching. The planning and budget should be aligned with the relevant Project Leader.

EEG21-05 Novel DEMO Divertor Concept Solutions for Simplified Maintenance and Exchange

Contact person: Antonino Cardella*– antonino.cardella@gmail.com
** on behalf of the PSDD Head of the DCT*

Background:
One of the crucial points in the dimensioning of a power producing fusion plant remains the amount of power that can be reliably produced and controlled within it. This heavily depends on, amongst other things, the power handling i.e the heat load that can be tolerated by the divertor under normal and off-normal operation.

The divertor design solutions adopted for ITER and foreseen for DEMO are based on water-cooled plasma facing components (PFCs) mounted on a supporting divertor cassette. The PFC concept is based on tungsten monoblock that surrounds the heat sink, made from copper-alloys, obtaining a particularly robust design in particular for high number high heat flux cycles. However, under intense peak heat loads, which may arise, for example, during loss of detachment or transients, the lifetime of the divertor target would become rather short, demanding the early replacement of the complete unit. This will lead to long machine downtimes, the generation of a significant amount of radioactive waste and significant investment cost penalties.

Objectives:
The main aim of this work is to explore and develop novel ideas for the design of practical sacrificial engineering solutions to be used for the areas of the strike points of the DEMO divertor that would enable local replacement without affecting other parts of the divertor system that are subjected to lower heat loads. This could include for example, but not only, enhanced heat transfer and heat transport schemes and/or innovative maintenance schemes, namely:

1. Heat pipe designs or thermoelectric magneto-hydrodynamic (TEMHD) driven flowing liquid metal as a heat sink to remove the heat from the divertor surface.

2. Easily removable plasma facing tiles removal, either leaving the cooling system in place or including the substitution of cooling pipes up to regions a low neutron fluence zone to allow re-welding of the servicing hydraulic connections.

Scope/Activities/Outcomes (foreseen):
This work should focus on the development of sound engineering solutions, its numerical simulation and at least partial experimental validation under relevant heat load conditions. This should address, in particular, design aspects of structural integrity, reliability, design integration, maintainability, material compatibility, safety and fabrication. The work shall be carried out by the selected candidate in close collaboration with the DEMO Central Team (DCT) [1] well as laboratories supporting the numerical validation and the experimental characterisation of the concept solutions developed. It will also be aimed at the scientific exploration and understanding of physics effects by conducting all the required theoretical and numerical simulation to predict the performance and operation limits. The main aim of this project to produce sufficient reliable domain data so that we have a convincing basis for developing the concept further. If proven feasible, such solutions would provide an attractive and easily demountable concept that would enable replacement without the need to cut and re-weld hydraulic connections in high radiation areas.

Competence development:
- Obtain a full knowledge of severe operation conditions of a divertor target in DEMO and future fusion power plants and understanding of limitations of current divertor designs and technologies
- Develop ability to identify and analyze alternative divertor target concepts by prioritizing the prospects for their practical implementation. Also define an R&D programme, if required, for the validation of proposed novel technologies
- Strengthen ability to work in a design team. Frequent interactions with experts on plasma facing component design, materials, divertor plasma physics and divertor remote maintenance will provide a very important training opportunity to the selected candidate.

Competences required:
Engineers holding a Master Degree.

Work packages involved:
WPDIV, WPMAT, WPRM

Facilities to be used:
TBD

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. This grant foresees multiple stays in the DEMO Central Team in Garching. The planning and budget should be aligned with the relevant Project Leader.

[1] In FP9, the DCT is foreseen to advance the design basis (physics and technology) of a DEMO fusion power plant, by implementing and agile architectural design capability, impartial analysis of options, and quick access to the expertise distributed in the EU fusion laboratories, universities and industry. This is needed to ensure the rapid convergence towards a feasible DEMO plant architecture (see G. Federici, C. Baylard, DEMO Project Charter Proposal, IDM reference: 2P3ZEP. April 2020).

EEG21-06 High Current Conductors for DEMO magnets

Contact person: Valentina Corato (Project Leader WPMAG) – valentina.corato@enea.it

Background:

Plasma in a tokamak must be confined in a finite volume, which is achieved by applying strong magnetic field. In the largest tokamaks, the required field can only be generated using superconducting magnets. Several magnet systems are necessary to shape the plasma and control its position.

The largest of these magnet systems are the so-called Toroidal Field (TF) coils. In DEMO, there are 16 D-shaped TF coils with the perimeter of approx. 40 meters. Every TF coil consists of more than 200 turns.

During the regular operation, the superconducting cables of the TF coils carry current, typically around 60 kA, without any resistance and voltage along the winding pack (WP). Only during ramp up and ramp down of the current, small inductive voltage over the coil terminals is generated. The situation is different if the superconducting cable accidentally quenches, due to some external heat deposition, i.e. when part of it becomes normal conducting. The quench propagation is a complex transient whose evolution typically requires sophisticated thermal-hydraulic codes to be predicted. As the cable reaches the current sharing temperature it starts heating up and this can initiate a divergence of the conductor temperature and of the supercritical helium coolant pressure. To avoid damage of the quenched coil, the current has to be switched off in a fast way, on a time scale of ~30 s. High voltage in the order of several kV is generated at the coil terminals during this fast discharge. The resistive voltage remains negligible with respect to the inductive voltage that opposes the change in current and magnetic field. Though quenches in the magnets are very rare, they represent a serious risk for the magnet system, e.g. for the integrity of its electrical insulation.

The high voltage generated during the fast discharge is proportional to the coil inductance, which can be reduced by designing the TF coil with lower number of turns, however carrying higher current.

The reduction of the maximum terminal-to-terminal voltage can be similarly pursued also for Central Solenoid and Poloidal Field coils

Objectives:

The objective of this call is to provide the suitable candidate with the ability:

(1) to develop innovative design proposals based on high-current conductors, supported by analysis,

(2) to lead the manufacturing of prototypes and the experimental tests for validating the innovative design.

Competence development:
The candidate will develop his/her skills on:

- Design Nb_3Sn, NbTi and HTS conductors for high operating current (e.g. 90-110 kA).
- Design WPs based on these high-current conductors.
- Perform thermal-hydraulic and electro-magnetic analysis of the designed coils to assess the conductor and WP design performances.
- Lead manufacture of the conductor prototypes and samples.
- Participate in the testing in SULTAN facility.
- Perform data reduction and interpretation of the results, with suitable numerical models when needed.
- Publish the results of the R&D and sample testing.

Competences required:
Engineers holding a Master Degree. A recipient of a Master degree or PhD in a University of Technology or Physics will also be considered as an eligible candidate.
The candidate will operate in a broad range of engineering fields, covering material science, cryogenics, numerical analyses, experimental work, and data acquisition and analysis. The candidate will be embedded in a project-oriented team and required to interact with other international groups and laboratories. As such, despite the majority of his/her activities will be conducted independently, the candidate must be capable of rapidly adapting to different technical contexts.

Work packages involved:
WPMAG (primary), WPDES

Facilities to be used:
Sultan facility (SPC, EPFL Switzerland)

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-07 Numerical optimization for advanced divertor engineering design with SOLPS-ITER

Contact person: Sven Wiesen (FTD Plasma Exhaust RO) – s.wiesen@fz-juelich.de

Background:

The SOLPS-ITER code [1] and its predecessors have been employed for the ITER divertor operational design studies [2]. The plasma-fluid/neutral-kinetic code is state-of-the-art with respect to included physics relevant for detached/dissipative divertor plasma scenarios [3] and has undergone a strong validation effort inside the fusion community (e.g. neutral models and convective plasma fluid drifts). Given the continued success for advancing the ITER divertor design studies (see e.g. [4]), the code is now also being employed for the assessment of the EU-DEMO divertor design under the auspices of both, WPDES/FTD and WPPWIE/FSD.

The ITER divertor design was a very extensive and time-consuming activity, including manual selection of promising candidates for the divertor shape and magnetic geometry and long convergence times of the SOLPS code. A more modern approach (numerical based) should allow for a faster cycle of advanced simulations for a more rapid divertor design study. In the forthcoming years, the assessment of the EU-DEMO divertor design candidate under the auspices of WPDES and the DEMO Central Team (DCT) should be supported by adequate numerical optimization techniques allowing faster turn-around times. Such numerical methods have been developed over the past 5-6 years, but the testing of these techniques were mostly based on simplified divertor plasma geometries (e.g. slab models, or reduced divertor geometries).

Objectives:

The goal of this combined WPDES/WPPWIE is to support further the development of advanced numerical optimization tools for the EU-DEMO divertor design (a *Design Tool for Reactors*), and also for the development (and testing) of improved physics analysis tools (i.e. SOLPS-ITER) for advanced divertor configurations (ADCs). As an example one could predict the relative change in the divertor dissipative performance (e.g. when moving from a single-null (SN) divertor to a X-divertor (XD) or Super-X divertor (SXD) with long divertor legs) provided the required engineering constraints (e.g. system size, pumping speed, magnetic field coil current limits and location, etc.).

Usually, the employment of SOLPS-ITER is slow given its small convergence rate when modelling the required physics details for detached divertors involving a coupling to the EIRENE Monte-Carlo neutral code. Contrary to forward simulations for the purpose of discharge analysis requiring a number of fixed inputs and other constraints (e.g. magnetic equilibrium and vessel geometry) that produce a reduced set of outputs (e.g. heat flux profiles), an optimization technique must be able to modify the initial inputs. So called *adjoint simulations* have the capability to take the difference between forward simulated output of a simulation and a target output (e.g. a broad and flattened heat flux profile) - the *cost functional* - that is then used to inform a change in the original input (e.g. a tilting of the divertor plate or change of angle of magnetic field line incidence).

Scope/Activities/Outcomes (foreseen):

Several applications/tasks of the use of the adjoint optimization method for divertor design are proposed which should be furtherly developed under this EEG:

- Task 1: To develop, test and employ improved divertor shape design techniques in SOLPS-ITER as initially proposed and developed by Dekeyser et al [5] employing the adjoint method. The idea is to optimize the SOLPS-ITER divertor solution towards, for example, to a uniform/flat target heat load (the cost functional) by modifying the divertor shape (plate tilt, size/extent of plate, neutral conductance toward pump etc.) It should be possible to include realistic engineering design constraints (e.g. for a EU-DEMO: system size, neutron shielding, pump speeds etc)
- Task 2: To develop improved optimization capabilities for the magnetic field configuration in the divertor. Based on earlier assessments of this technique by Blommaert et al for WEST [6] the optimization technique should include also design constraints for the field coil conductors, limits and system size. The cost functional of task 1 is thus to be extended to include a minimization of the joule loss through the coil currents.
- Task 3: Automatic grid generation. Task 2 (and sometimes also task 1) requires changes in the magnetic field configuration and subsequently for SOLPS-ITER new grids must be produced at each optimization cycle. Fast automatic grid generators must be developed or existing extended. This is also helpful for an automatic grid generation for more complex ADC topologies like snow-flakes ([7]).
- Task 4: The developed optimization tools in SOLPS-ITER should be tested. In fact, any code validation attempt that attempts to calibrate a-priori unknown physics parameters (e.g. anomalous transport) can be seen as nothing else but an optimization problem (c.f. Baelmans et al [8]). In this case, the cost functional is again a target variable like the (measured) heat flux at the divertor plates, the variational parameter are then transport coefficients or other (physics) model constraints. A task proposal is to predict in an optimized way the physics model setup for DTT and Compass-U.

Competence development:

Tasks 1-4 require some modifications in the SOLPS-ITER code that should be strongly supported and supervised by the KU Leuven team that developed the original optimization techniques.

Competences required:

A Master or PhD degree in Engineering. In this case, a recipient of a relevant Master degree or PhD in a University of Technology – such as Technical/Applied Physics – will also be considered as an eligible candidate.

Work packages involved: WP PWIE and WP DES.

Facilities to be used: TCV, WEST, ETASC-ACHs

Mobility:

During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. This grant foresees multiple stays in the DEMO Central Team in Garching. The planning and budget should be aligned with the relevant Project Leader.

[1] S. Wiesen et al., J. Nucl. Mat. 463 (2015) 480
[2] A. S. Kukushkin et al., Fus. Eng. Des. 86 (2011) 2865
[3] X. Bonnin et al., Plasma and Fusion Research 11, 1403102-1403102
[4] E. Kaveeva et al., Nucl. Fus. 60 (2020) 046019
[5] W. Dekeyser et al., Nucl. Fusion 54 (2014) 073022
[6] M. Blommaert et al., J. Nucl. Mater. 463 (2015) 1220
[7] S. Van den Kerkhof et al., Contrib. Plasma Phys. 58 (2018) 696
[8] M. Baelmans et al., Plasma Phys. Control. Fusion 56 (2014) 114009

EEG21-08 Nuclear engineer in support of DEMO nuclear safety studies

Contact person: Maria-Teresa Porfiri (Project Leader WPSAE) - mariateresa.porfiri@enea.it

Background:
Of main importance for the final development of DEMO fusion reactor is the ability to meet the safety objectives both for equipment and the external environment.

Safety analysis is an important tool for justifying the safety of nuclear power plants. Typically, this type of analysis is performed by means of system computer codes with one dimensional approximation for modelling real plant systems. However, in the nuclear area there are issues for which traditional treatment using one dimensional system codes could be considered inadequate for modelling local flow, fluids distribution and heat transfer phenomena. The increasing interest, in nuclear safety analyses, for the application of three dimensional computational fluid dynamics (CFD) codes as a supplement to or in combination with system codes, makes it necessary to reduce uncertainty and conservativism characterizing most of the actual tools. There are a number of both commercial (general purpose) CFD codes as well as special codes for nuclear safety applications available.

For this kind of analyses, it is necessary to have a deep knowledge of the nuclear plant system and a number of different codes covering fluid-dynamics, thermal hydraulics, chemical reaction, fire and explosion scenarios. Combination and integration of accident progression codes, system codes, computational fluid dynamics and structural tools are necessary and unavoidable. Coupling of two or more different computer codes is a challenge for the entire scientific community.

Objectives:
This call, that could be take advantage from the complementary work started with other previous grants, is aimed at:

- providing a broad knowledge of the systems of a tokamak plant that are important for the safety performance, together with familiarization of all parts of the safety analyses, with particular reference to the calculation of the potential consequences of accidents.
- providing a good comprehension of the coupling capabilities of different tools for safety analyses in nuclear fusion plants.

Scope/Activities/Outcomes (foreseen):

- The scope of the activity is to support the accident analyses carried on the WPSAE by applying different simulation tools than those normally used in safety studies and to verify.
- The activities will include accident analyses on DEMO main systems.
- The outcomes are reports to document abnormal events complemented by sensitivity and uncertainties studies.

Competence development:
During the training period, special focus will be put on integration of accident progression tools and CFD methodology for selected accidental sequences, which can be defined as dominant in the entire safety of DEMO. In particular, the candidate will work with different experienced safety experts in the selection of the accidental scenarios to be analyzed. He or she will be involved in the modelling of

primary and auxiliary systems with the aim of obtaining information useful to the safety design of the systems themselves. At the same time, it is expected that the candidate undertakes one or more short internships in universities and/or research centers, aimed at training on various computer programs, methodologies and their application in the field of nuclear fusion.

Competences required:
Engineers holding a Master Degree.

Work packages involved:
WPSAE (primary), DCT/WPDES and other DEMO WPs

Facilities to be used:
To be determined

Locations:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-09 Development of Tungsten Plasma Facing Components for W7-X

Contact person: Rudolf Neu (Project Leader WPDIV) – Rudolf.Neu@ipp.mpg.de

Background:

The transition to reactor relevant metal plasma facing components in W7-X will be a major and necessary step in providing the proof of principle that the stellarator concept can meet the requirements of a future fusion reactor. During the OP1.2 campaign in W7-X a lot of knowledge about the limitation of plasma operation has been already obtained, in particular, regarding the overloading of several baffle elements and divertor modules, excess plasma heat loads onto leading edges and in the pumping gap in certain configurations as well as heat loads by fast ion losses onto wall and diagnostic components. Therefore, restrictions of the plasma operation will be unavoidable in the next operational phase (OP2) with the current design. On the other hand, decisions of required design changes are already possible at this stage, knowing that their realization demands several years of development. Experience gained during OP2.1/2.2 can probably be taken into account in the detailing of the final design of new plasma facing components with a tungsten armour. However, the required development and project work should start as early as possible.

Objectives:

New high heat flux target elements with tungsten as plasma facing material have to be developed together with EUROfusion and with qualified external companies in order to meet the heat flux requirements of up to 10 MW/m^2 in steady-state operation. Different technical concepts (flat-tile design, mono-block design) have to be evaluated with respect to a possible usage in W7-X aiming to the development and fabrication of mock-ups, which have to be tested in relevant high heat flux facilities. Depending on the specific area of expertise of the applicant, also optimization of the W7-X 3D-divertor design and baffle area could be envisaged, involving validated physics tools into the mechanical design (e.g. by employing MHD equilibrium codes and field line extension) or expected SOL transport. These developments and associated tests should provide a solid decision basis for the technology selection and prototype fabrication of tungsten target modules being part of the W7-X W-Roadmap activities.

Scope/Activities/Outcomes (foreseen):

The successful candidate should provide the link between the W7-X activities within WPDIV and the activities within the W7-X project team “Divertor Design” towards a carbon-free W7-X device.

Competence development:

The successful candidate will work at the interface of engineering and physics. Depending on her/his initial abilities she/he will learn the view, techniques and tools of the respective other area. Combined with the original strengths of the candidate this will synergistically increase his/her expertise providing her/him an ideal profile for the future work in fusion research. This development will be strongly fostered by the team of experienced engineers and physicists working on the development of the transition of W7-X towards a full metal device.

Competences required:
A Master or PhD degree in Engineering. In this case, a recipient of a relevant Master degree or PhD in a University of Technology – such as Technical/Applied Physics – will also be considered as an eligible candidate.

Work packages involved: WPDIV (primary), WPMAT, WPW7X, WPPWIE

Facilities to be used:
Computer and experimental facilities at the hosting (and collaborating) Research Units

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-10 DEMO coolant purification system

Contact person: Christian Day (Project leader WPTFV) – christian.day@kit.edu

Background:

DEMO has to demonstrate tritium self-sufficiency, which is one of the 8 missions stated in the European Roadmap for the realisation of fusion energy. Hence, DEMO will be the first fusion facility that includes a closed fuel cycle and features a complex breeding blanket system. To achieve this goal, tritium has to be recovered not only from the tokamak exhaust stream and from the blanket units, but also from the entire primary coolant loops. The latter is of great relevance for the safety aspects to avoid the release of tritium inside and outside the buildings.

In the previous years of DEMO programme, main features of the Coolant Purification System (CPS) have been identified for both helium and water coolant options. For the case of helium CPS the amount of hydrogen has an important role in the size and operation of the several components, while for the water coolant CPS it has been demonstrated that the existing water detritiation facilities do not allow the tritium control inside the coolant loop without the presence of appropriate anti-permeation barriers. It was one of the main recommendations of the recent design review, to integrate all coolant functionalities and bring the system into a more mature level.

Objectives:

This call aims to advance the current design with the following aspects: i) include the contribution of the tritium permeation rate coming from first wall and divertor regions; ii) take into account the most recent results of the tritium simulation analysis at system level and review the CPS accordingly, iii) initiate the integration activity of the CPS units in the primary coolant loop and iv) consolidate the knowledge of water detritiation systems necessary for the water coolant loop. In this view, the candidate will be firstly required to become proficient with the complex issue of the tritium management in primary coolant and then support the CPS design activities in the following main areas:

- Implement the tritium mass balance analysis by taking into account all the different tritium sources (blanket, first wall, divertor, etc.);
- Assess the water and helium CPS size, according with the outcomes of the tritium mass balance and perform the dimensioning of different main components;
- Improve the knowledge with the existing water detritiation facilities;
- Provide the CAD layout of the CPS;
- Identify possible integration solutions for the CPS units.

Scope/Activities/Outcomes (foreseen):
Due to the fact that the correct tritium management in primary coolant can be achieved exclusively with an integrated approach, the candidate should be able to constructively interact with other experts of DEMO design with the aim to understand the needs and the limits of the other subsystems connected to the CPS. This work will also have to be closely linked to the parallel development of an integrated fuel cycle simulator. Main activities carried out under this EEG are:

- Task 1. Acquire the knowledge of principal technologies proposed for helium and water CPS;
- Task 2. Update the current CPS pre-concept design by taking into account most recent outcomes of the KDII#2 activity;
- Task 3. Consolidate and implement the CPS design also with CAD drawings and identify possible CPS integration solutions inside the tokamak and tritium buildings.
- Task 4. Analyse the impact of different coolant medium (helium and water) on CPS and, more in general, on the fuel cycle outer loop technologies.

Main outcomes of this EEG are the consolidation of the CPS design and the evaluation of the impact in using helium or water primary coolant on the fuel cycle outer loop technologies.

The successful applicant will be closely involved in the CPS team located at ENEA, but strong interactions with the other Research Units included in the project will be required.

Competence development:
During the training period, the candidate will acquire knowledge of tritium transport phenomena, tritium processing technologies and principal tritium safety requirements. He/she will implement design skills on tritium technologies by working with expert in the field and by attending dedicated CAD/CATIA courses. Short internships in universities and/or research centres are foreseen for improving the knowledge of principal CPS interfaces.

Competences required:
Engineers holding a Master Degree.

Work packages involved:
WPTFV (primary), WPBB, WPSAE, WPBOP

Facilities to be used:
TBD

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-11 Development of Optical Diagnostics for ITER and DEMO

Contact person: Wolfgang Biel (Project Leader WPDC) – w.biel@fz-juelich.de

Background:

The international fusion experiment ITER as well as the European demonstration fusion power plant (DEMO) under development require a number of accurate measurements related to the plasma properties to be investigated and controlled. Compared to existing fusion devices, additional limitations exist for the diagnostic systems in the more demanding environment of ITER and DEMO. Especially the diagnostic front-end components will be subject to extreme load conditions including high flux and fluence of neutron and gamma radiation as well as fast neutral atoms, thermal and mechanical loads. Additionally, on DEMO the space for diagnostic implementation is limited to facilitate other goals such as tritium self-sufficiency and limited maintenance. As a consequence, diagnostic systems on ITER and DEMO have to be optimised for robustness and reliability rather than only for performance, and be mounted in protected locations where they can withstand the loads.

Objectives:

The training and work within this fellowship concentrates on the development of durable and reliable optical diagnostic systems including spectroscopy from X-ray to IR wavelength, radiation power measurements, Thomson scattering and IR interferometry/polarimetry for plasma measurements and control on the fusion devices ITER and DEMO under development. The fellowship will be closely related to the EUROfusion work package diagnostic & control (WPDC). Moreover, one or several longer stays at the ITER site in Cadarache are foreseen to participate in optical diagnostic developments and training actions.

Scope/Activities/Outcomes (foreseen):

Work topics are:

- Cooperation in review and assessment of plasma diagnostic methods relevant to plasma control, with respect to the measurement process and the technical realisation.
- Systems level design of optical diagnostics and quantitative prediction of the expected measurement performance for accuracy, time resolution and spatial resolution.
- Coordination or execution of specific studies on the design and integration of optical diagnostics compatible with remote handling with CAD, optical design software and neutronics simulations.
- Concept design of optical and mechanical components such as mirrors, filters, detectors and spectrometers.
- Quantitative analysis of the degradation of diagnostic components under typical load conditions for lifetime and reliability.

Competences required:
Engineers holding a Master Degree. A recipient of a Master degree or PhD in a University of Technology or Technical/Applied Physics will also be considered as an eligible candidate.

Work packages involved:
WPDC

Facilities to be used:
To be defined as part of the proposal, as required e.g. to contribute simulations, enable prototype testing or demonstration experiments.

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-12 Test technologies for the characterization of in situ irradiation effects under applied loading conditions

Contact person: Gerald Pintsuk (Project Leader WPMAT) – g.pintsuk@fz-juelich.de

Background:

Among the technological challenges on the road to commercial fusion power, developing plasma-facing materials (PFMs) and components (PFCs) with sufficient longevity in a power-generating reactor's first wall and divertor is one of the most critical, since a catastrophic failure of materials at these locations would likely compromise the viability of the entire fusion power plant. Yet, the quantitative evaluation of PFC designs is hindered by the lack of design-relevant data for irradiated materials. A not yet fully resolved question concerns the combined synergetic effect of high dose irradiation and external loading, for example particle loading from the plasma and/or mechanical stress resulting from thermal expansion or radiation-induced swelling. The development and validation of new tools and methodologies, to assess and predict the radiation-induced material changes occurring under loading conditions, are urgently required to provide input to the engineering design studies and dedicated material models required for the development of a DEMO design framework.

Objectives:

Applications are invited from individuals interested in the development of novel research systems to investigate the irradiation induced degradation of materials for in-vessel and in particular highly loaded plasma facing components. The focus should be on the microstructural modification and mechanical alteration / degradation of materials due to the synergetic interactions of radiation damage with other factors such as embedded hydrogen and helium as well as superimposed mechanical stresses induced for example by thermal expansion effects. The work shall combine experimental testing with modelling to develop transferable understanding of mechanisms leading to material degradation.

Scope/Activities/Outcomes (foreseen):

The selected candidate is expected to focus his/her research on one or more of the following areas:

- Development of testing technologies (both experimental methodology and related theoretical concepts) for the combination of irradiation with other loading types either using neutron, proton or ion irradiation determination of mechanical and/or physical data relevant for engineering design.
- Determination of microstructural changes and mechanical property data due to the applied loading conditions.

Competence development:
In particular, the candidate will develop his/her skills on:

- Design of in-situ mechanical test systems to be used within irradiation facilities
- Manufacturing and set-up of these facilities and perform related operation
- Detailed analyses of the interaction between externally applied stresses and irradiation induced material damage causing microstructural changes as well as changes in the mechanical response of the material.
- Physical understanding of the irradiation induced material modifications towards implementation into materials modelling.

Competences required:
A Master or PhD degree in Engineering. In this case, a recipient of a relevant Master degree or PhD in a University of Technology – such as Technical/Applied Physics – will also be considered as an eligible candidate.

Work packages involved:
WPMAT (primary), WPPRD

Facilities to be used:
Relevant facilities dealing with ion irradiation

Mobility:
During the work, collaborations with and visits to laboratories involved in irradiation and modelling of irradiation effects should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-13 DEMO Design of the EC Mid Steering Antenna

Contact person: Jean-Philippe Hogge (Project Leader WPHCD) – jean-philippe.hogge@epfl.ch

Background:

The baseline heating technology for all plasma phases of DEMO is Electron Cyclotron Wave (ECW). It will be used for the break-down of the gas and creation of the plasma, during the ramp-up phase to bring the plasma to conditions where fusion reactions occur, during the flat-top phase (lasting two hours), and during ramp-down. During the flat-top phase, the functions of ECW are multifold: heating the plasma core, controlling magnetohydrodynamic instabilities which occur at specific locations of the discharge (the “neo classical tearing modes (NTM)” at q surfaces = 3/2 and 2/1) and a thermal instability at near to the edge. A total of about 130 MW is foreseen to be deposited for these various functions.

ECW is ideal to meet these requirements. At the right frequencies, the wave launched from the antenna propagates without reflection and is fully absorbed. The location of the absorbing layer is determined by the frequency of the wave, the launch geometry and the local magnetic field in the plasma. This allows, for example, in the case of NTM to select a very precise absorbing layer.

Physics studies have led to precise requirements on the ECW beam in the plasma. During Horizon 2020, design studies have provided the essential components of the antenna itself. In one of the variants under consideration, the millimeter wave power is brought from the sources (known as gyrotrons) to the antenna located near to the plasma. In the antenna, open oversized waveguides transmit the power to a couple of mirrors, the first one fixed and the second one steerable. This set of mirrors allows then to direct the beam to the correct location in the plasma.

Objectives:

Scope/Activities/Outcomes (foreseen):

The design of these mirrors and the actuator(s) of the mobile one(s) is the scope of the EEG call. The work shall encompass the participation to the definition/revision of the load requirements (such as mm wave losses, nuclear, plasma, cooling and electromagnetic loads), requirements from remote maintenance, safety and integration into the port plug, as well as the thermomechanical design of the mirrors and of the actuators compliant with the loads requirements. The design of the mirrors and actuators shall be integrated in the launcher plug, which is under the responsibility of WPHCD. It is expected that the work will be iterative to be able to cope continuously with DEMO physics and integration requirements. Interaction with the WPHCD team, the DEMO Central Team located in Garching and other collaborating Work Packages (Remote Maintenance, Breeding Blanket, Nuclear calculations) is foreseen during the execution of the grant.

Competence development:
During the training, the grantee will develop a large spectrum of skills, aiming at giving him/her the necessary competencies for the mechanical design of te EC mid-steering antenna on the one hand, and to related domains such as cooling, thermo-mechanical simulations, handling of mm-wave radiation, neutron irradiation impact assessment on the antenna components and material selection.

Competences required:
Engineers holding a Master Degree.

Work packages involved:
WPHCD

Facilities to be used:
N/A

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-14 Non-destructive testing for joints and measurements of neutron induces material degradation

Contact person: Oliver Crofts (Project Leader WPRM) – oliver.crofts@ukaea.uk

Background:
Demonstrating joint integrity to the satisfaction of a nuclear regulator is critical and suitable NDT (Non-Destructive Testing) techniques must be understood at an early design stage because changes to meet regulatory requirements could have a significant impact on the plant and maintenance designs. This includes welded, brazed and mechanical connections in pipe and other critical joints.

ITER will need to develop NDT to qualify a range of welds and other joint types before it can be licenced. DEMO may require significantly different welds and joint types due to differences in requirements, licencing arrangements, materials, environment and performance requirements and new techniques are becoming applicable that may be available to DEMO such as phased-array ultrasonics, laser shearograph and laser ultrasonics.

NDT techniques, such as magnetic hysteresis, can also be used to indirectly measure material degradation due to neutron exposure and allow an assessment of the condition of actual components and, in conjunction with the testing of material samples, removed from the tokamak, it may be able to reduce the frequency of planned maintenance.

Objectives:
This EEG would build on work started in the past:

- Establish the state of the art for both joint integrity, using ITER as the starting point, and for neutron induced material degradation
- Determine difference in regulatory requirements for the demonstration of the integrity of joints – both welded, mechanical and alternative technologies, as well the cross-over into pipe lifetime monitoring, valves and other features that must be dealt with remotely
- Define and describe the material degradation mechanism due to neutron induced damage
- Establish the technology gaps and technical risks in the field for implementation on DEMO
- Mitigate the technical risks by developing methods to fill technology gaps and create outline concept designs
- Where suitable novel technologies have been identified and where the scope of the grant permits, develop and test proof-of-principle trials
- Define the roadmap for use of these techniques on DEMO

Competence development:
During the execution of the project, the candidate is expected to work with DEMO design experts to determine the feasibility of the proposed approaches and their integration into the overall DEMO design. This also requires the candidate to develop and pursue collaboration and cooperation with other specialists at relevant research centres, and in industry, to achieve the designated goal of the project.

Competences required:
Engineers holding a Master degree.

Work packages involved:
WPRM (primary), WPMAT, WPDES

Facilities to be used:
Small test rigs may be developed to test solutions or testing may be performed in suitable existing industrial facilities.

Mobility:
During the work, collaborations with and short visits to laboratories with expertise on the subject matter should be foreseen. The planning and budget should be aligned with the relevant Project Leader.

EEG21-15 EU enhancement project for JT-60SA: Thomson Scattering Diagnostics

Contact person: Carlo Sozzi (Project Leader WPSA) – carlo.sozzi@istp.cnr.it

Background:

The series of Enhancement Projects that are being carried out by Fusion for Energy (F4E) and EUROfusion to prepare the European exploitation of JT-60SA, the large tokamak jointly built by Europe and Japan at QST (National Institutes for Quantum and Radiological Science and Technology, Naka, Japan), includes the procurement of large part of the whole Thomson Scattering system, which is a state-of-art diagnostics design developed within the severe constraints of a large superconducting machine.

EU is contributing to the design and procurement of several key components of the core and edge Thomson Scattering (TS) diagnostics: optical fibres and polychromators of both systems, collection optics with its supporting mechanical structure and the laser source for the edge system. Procurement is expected to be completed in 2022 as well as assembly and laboratory tests, installation and commissioning during the shutdown in 2022-23, in order to be available at the restart of operation in the second half of 2023. The edge TS system will give important inputs for pedestal physics and its operation and scientific exploitation is expected to be led and largely carried out by the EU.

Objectives:

In this context the grantee will be trained by an experienced TS team and will receive the opportunity of actively contribute to the success of the TS enhancement project, also as a bridge to design, development and operation of TS systems in other experiments, especially ITER, by closely following the project during important phases:

- assisting the Project Leader of the TS Enhancement Project (located in Consorzio RFX, Italy) in:
  - optimizing and finalizing the engineering design
  - interfacing with F4E and EUROfusion for coordination of the project
  - interfacing with QST for management of interfaces and for collaborative efforts
  - interfacing with suppliers and project team for the follow-up of components procurement;
- under the coordination of the Project Leader and assisted by the project team, the grantee will participate in the pre-assembly, in the supervision of the installation and in the commissioning and calibration of the TS diagnostic at QST;
- under the coordination of the Project Leader of the TS enhancement and in liaising with the Physics Team, he/her will have the opportunity to initiate the edge TS system exploitation and data production;
- carry out investigation of polarimetric TS and dual wavelength calibration, with possible application to ITER.

Competence development:
The trainee is expected to acquire extended hands-on experience on technical and management aspects of diagnostics engineering, with insights in the operation of the system during the commissioning and plasma operations.

Competences required:
A Master or PhD degree in Engineering. In this case, a recipient of a relevant Master degree or PhD in a University of Technology – such as Technical/Applied Physics – will also be considered as an eligible candidate.

Preferable:

- Expertise in experimental physics
- Previous expertise in fusion devices and plasma optical diagnostics

Work packages involved: WPSA (primary), WPPrIO

Facilities to be used: JT-60SA

Mobility:
Visits at RFX consortium (Padua, Italy), QST-Naka (JT-60SA), and at ITER are envisaged. The planning and budget should be aligned with the relevant Project Leader.

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EEG21-16 Advanced control of gyrotrons for heating, plasma stabilization and diagnostics

Contact person: Andreas Dinklage (Project Leader W7X) – andreas.dinklage@ipp.mpg.de

Background:

Electron cyclotron heating (ECRH) is a versatile actuator for next-step fusion devices. Since ITER foresees ECRH-based control of plasmas, there is a long-term strategic need to ensure full capabilities of heating systems in support of the scientific program of ITER. In ITER, Electron cyclotron heating is required for plasma heating, current drive in the plasma core, control of MHD instabilities, and plasma start-up.

The specific needs lie in field of expertise in high-power microwave technology and its control (electric engineering). The vision of this proposal is to initiate and foster a long-term engineering support for the control of ECRH making use of the world's most powerful ECRH on W7-X. The existent environment on W7-X, unique development capabilities at KIT provide an ideal background aiming at systematic coverage of requirements for ITER. Developments for plasma control benefit from present activities conducted by F4E. Moreover, high-field side launch with the remote steering launcher on W7-X and gained experience in terms of RAMI will allow an assessment for applications on DEMO.

Advanced control of gyrotrons for electron-cyclotron (EC) heating, plasma stabilization and diagnostics improves three major quality factors of any EC system in particular for long pulse or cw operation: total available output power, reliability during operation, and possible modes of operation. Advanced gyrotron control systems, however, are missing today, but are essential for plasma experiments with EC heating on such as W7-X, ITER and future DEMO.

Bridging this gap is in reach on W7-X where the advanced gyrotron control system allows for increasing the achievable output power while maintaining (or even increasing) operational reliability. An automated, fast recovery system of the output power has been developed and successfully tested. Additionally, the development of a fast feedback gyrotron controller allowing to minimize the probability of mode loss or allowing to avoid any mode loss by taking countermeasures in advance has been started.

Considering the possible integration and transfer concept to ITER in strong collaboration with the ITER HCD/ECS team right from begin of the EEG offers the possibility for later integration of the successfully developed concepts to ITER with minimum effort and maximum benefit. Hence, the EEG does not replace but complements the EUROfusion Engineering Traineeship for basic implementation of the control system in an ideal way. It allows for the advance of the basic installation by an additional option for higher output power (up to 10% using the initial mode recovery concept already demonstrated in W7-X), more reliable operation, increased efficiency at the highest output power and completely new operating possibilities.

Objectives:
Given successful prototypical solutions, the challenge is to demonstrate high reliability and availability through advanced control. Therefore, this proposal aims at the continuation of the work already initiated on W7-X. The focus is set on a first advanced phase-/frequency-control to increase the possible modes of operation of the heating gyrotrons for specific tasks, which are directly relevant to the work plan of W7-X. Examples are Collective Thomson Scattering (CTS), frequency sensitive power switches and combiners, and, eventually, beat wave operation using several gyrotrons at defined frequency steps. Finally, it might allow to find operating points of superior operational stability and higher efficiencies.

Scope/Activities/Outcomes (foreseen):
Engineering in support of the frequency stabilization for high power gyrotrons will need to cover:

- Simulation and experimental testing of the frequency behavior of high power gyrotrons
- Theoretical investigations of possible frequency stabilization methods
- Experimental investigations on the stability and dynamic capabilities of the gyrotron high voltage power supply
- Design and implementation of an external feedback control system for the frequency stabilization of high power gyrotrons
- Stability analysis of the feedback control system, demonstration of opportunities and limits
- Integration into the W7-X control system
- Integration and transfer concept to ITER (w/ F4E)
- Assessment for DEMO requirements

Competences required:
MSc Electrical Engineering

Work packages involved: WPW7X, other experimental WPs

Facilities to be used: N/A

Mobility:
The conduction of work is most efficient employing capabilities at W7-X (Greifswald), KIT (Karlsruhe), F4E (Barcelona) and ITER (Cadarache). The planning and budget should be aligned with the relevant Project Leader.

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EEG21-17 Development of Infra-Red monitoring system using artificial intelligence techniques in view of ITER application

Contact person: Andreas Dinklage (Project Leader W7X) – andreas.dinklage@ipp.mpg.de

Background:

Reliable surveillance of plasma facing components and the first wall is key to safe and efficient operation of high-power, long-pulse fusion devices. The pace and the success of ITER's research program will depend on the reliability and robustness of its control systems to cope with overloads and operation monitoring. Therefore, ITER is developing imaging systems to monitor temperatures of the plasma facing components as well as the plasma emission and flow through infrared and visible measurements. While physics driven, forensic studies are central to the identification of overloads and the interpretation of large amounts of data, latest developments in artificial intelligence may provide approaches for reliably using the data for plasma control and investment protection systems. The vision is a reliable long pulse monitoring of plasma facing components capable to detect exceptional events like hot spots or dust interaction with the plasma.

While R&D in the field is subject to several efforts in different EUROfusion work packages, systematic but customized software engineering is necessary to develop cross-machine toolboxes putting the focus on AI agents to cope with unprecedented amounts of surveillance data. Moreover, short response times need to be technically provided to induce appropriate counter actions in case of harmful events. These requirements make latest software and control engineering a prerequisite for a successful delivery of required tools. Consequently, this proposal is to initiate and maintain the required and new software engineering back-bone for a successful application of artificial intelligence for wall protection on ITER and long-pulse and metallic wall EU fusion facilities (W7-X, WEST).

Objectives:

The EEG proposal is to develop a wall monitoring system for real time protection based on infra-red and visible cameras observing the plasma facing components in support to ITER operation. The system will be implemented and validated on W7-X (IPP/Greifswald (Germany)) and WEST (CEA/Cadarache (France)) to provide safe and high performance steady-state long duration operation. Synergies of both institutes for this common objective lead to develop unique expertise and methods to prepare the ITER first wall and divertor protection.

Scope/Activities/Outcomes (foreseen):

The internal components inside present and future fusion experiments (ITER, DEMO) are exposed to large amount of power (~100MW) and energies and are designed to protect vacuum chamber of the machine from high fluxes coming from the confined thermonuclear plasma. The concentrated thermal load could exceed the limits set by the component's design, causing irreversible damages. To avoid such damages, the internal components are monitored during plasma operation using a set of infrared (IR) viewing system. Since human operators could not handle and process a large numbers of thermal events in real time operation, the development (including design, implementation and exploitation) of automated detection and analysis tools is an active domain of research in the context of the nuclear operation of ITER and future DEMO reactors [1-3]. W7-X and WEST teams have already joined their efforts on designing and building the infrared acquisition and data handling tools. There is a significant collaboration between the two EU teams, through the development and exploitation of ThermaVIP acquisition, visualization and analysis software. Both groups share their new developments, where machine generic applications and routines developed for one machine are easily transferred to the other one before a future implementation on ITER.

This proposed EEG is for bringing the collaboration to the next step of development, by developing and implementing advanced data analysis techniques for the management of thermal events in both machines. The tools shall address the challenges of hotspots classification, including automatic detection of false positive and irrelevant hot spots, hot spot rating (criticality, urgency, relevance) by using techniques of artificial intelligence and machine learning (object detection & tracking, instance segmentation, expert system, fuzzy logics, deep learning, other…). The workflow will be designed with a system engineering approach, so that its relevance goes beyond the exploitation and validation on W7-X and WEST, and could be implemented in other magnetic fusion machines (JT-60SA, ITER, DEMO).

Complementing a broader attempt for a European framework in computer vision for wall protection, a tight collaboration with the proposal on video protection is envisaged.

Competences required:
Master degree in Engineering or equivalent in a relevant discipline; Excellent interpersonal, teamwork and communication skills, including a good level of spoken & written English. Preferably also experience working in an international environment

Work packages involved:
WP-W7X (primary), WPPrIO, WPTE (optional), WPPWIE (optional)

Facilities to be used:
WEST, W7X, collaboration with ITER-CT

Mobility:
The conduction of work is most efficient employing capabilities at W7-X (Greifswald), CEA (Cadarache), and ITER (Cadarache). The planning and budget should be aligned with the relevant Project Leader.

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EEG21-18 Engineering support on the wall conditioning and ITER GDC design

Contact person: Xavier Litaudon (Project Leader WP PrIO) – xavier.litaudon@cea.fr

Background:
In the coming years, ITER operation will have a growing focus on (integrated) commissioning and wall conditioning leading to its first plasma operation. Conditioning techniques such as Glow Discharge Cleaning (GDC) will be used in ITER. Indeed, for a successful operation of ITER and future magnetic fusion reactors, it is crucial (i) to minimize the radiation losses by controlling the impurity content released from the vacuum vessel, and, (ii) to ensure a good plasma density control by reducing the level of particles that will recycle at vessel's surface. The design of the ITER GDC system is currently under development and could strongly benefit from the commissioning and operational experience of presently operating devices. The WEST facility at CEA-Cadarache (close to ITER), a superconducting machine equipped with a tungsten wall, offers an excellent test-bed to develop conditioning strategies, and expertise as well as to engineer ITER-relevant equipment(s) in collaboration with the ITER team. Furthermore, the grantee could have the opportunity to participate in the first commissioning of a new GDC system at the JT-60SA tokamak in Japan, a joint EU-Japanese project and the world's newest largest tokamak.

Objectives:
The high-level objectives of this EUROfusion Engineering Grant proposal are twofold:

(i) to establish protocols and strategies to assess both the quality of the vacuum and the conditions of the plasma-facing components based on existing European facilities in preparation for ITER operation, and,

(ii) to inform and support the design of the specific ITER GDC system.

Scope/Activities/Outcomes (foreseen):
The successful candidate will have the opportunity to become familiar with a complete set of standard tools for the conditioning of magnetic fusion devices and to rigorously assess their impact on plasma operation on WEST and to compare with other EUROfusion facilities. In particular, an optimization of conditioning strategies will be carried out by defining precise requirements on the maximum tolerable level of key impurity signals to achieve tolerable radiated power fractions during plasma discharges. These criteria will be used by the candidate to optimize the GDC strategy in current facilities and on ITER to ensure in a systematic manner long-lasting and spatially homogeneous conditioning effects. (S)he will propose alternative and improved GDC strategy dedicated to the ITER environment as well as to support the design of the ITER GDC electrodes. This activity could lead to the conception of prototypes to be tested in dedicated facilities and/or in WEST. The ITER GDC system consists of seven electrodes located in port plugs as well as their supporting systems. These electrodes are plasma-facing components that are currently undergoing a significant re-design to simplify the initial one and to address integration issues in the ITER port plug. The redesign of the ITER GDC electrodes requires a trade-off between the demanding and challenging thermal, electrical, materials, and radiation requirements

Competence development:

- Vacuum and associated conditioning techniques.
- Electromagnetic engineering expertise.
- Experience from design, testing, commissioning to operation and application (data analysis and implication for plasma operation).
- Experience of working in an international environment.
- Training and network with European experts through a wall conditioning workshop planned by the EUROfusion Operations Network (EON) under the EUROfusion WPPrIO work package.
- Participation in an engineering-oriented Fusion School.

Competences required:
Master degree in Engineering or equivalent in a relevant discipline.
Excellent interpersonal, teamwork and communication skills.
Excellent level of spoken & written English.

Work packages involved: WPPRIO, WPPWIE, WPTE

Facilities to be used:
WEST and EUROfusion facilities, JT-60SA, Collaboration with ITER-IO.

Mobility:
Visits to CEA (Cadarache), other EU facilities and ITER (contact So Maruyama (IO) So.Maruyama@iter.org) are envisaged. The planning and budget should be aligned with the relevant Project Leader.

EEG21-19 Refactoring and deployment of the TOKES tokamak plasma transient code

Contact person: Sebastijan Brezinsek (Project Leader WP PWIE) – s.brezinsek@fz-juelich.de

Background:
The TOKES ("Tokamak Equilibrium and Surfaces") code (I. Landman, G. Pestchanyi et al.) is a comprehensive, parallelized tokamak fluid plasma description, particularly suited to the simulation of fast transient heat loads on plasma-facing components (PFC) for engineering and physics studies. It computes multi-fluid processes (including impurities and neutrals) in the core and SOL plasmas, accounting for the dynamics of magnetic fields and currents in the plasma and in the tokamak poloidal field coils. The code features a numerical meshing out to all wall surfaces with the possibility of spatially variable grid resolution on the mesh. It includes standard surface interactions such as sputtering, but also surface vaporization and, importantly, a vapour shielding module. Developed over more than decade at KIT, TOKES has been extensively used for several years in specific ITER studies, covering in particular simulations of disruption mitigation by massive gas and shattered pellet injection and the impact of heat fluxes due to non-mitigated disruptions and ELMs (Edge-Localized Modes), including vapour shielding effects. The code is also being applied to JET-ILW (ITER-like wall) and to the EU DEMO PFC design activities considering in particular the impact of disruptions and sacrificial limiters. Compared with more conventional boundary codes, it has the advantage of rapid run times, permitting extensive parametric studies even at the reactor scale like required in the DEMO design phase.

Objectives:
Unfortunately, the future of TOKES is critical as the principle developers and users are retired, though G. Pestchanyi would be available for a period of time to support a code refactoring. This EEG proposal aims to secure the continued deployment of TOKES for the good of ITER and DEMO by a two stage process in which the code is first refactored from Delphi into a new format like C++ or another language. This transfer would also allow the use of the code in the ITER Integrated Modelling Analysis Suite (IMAS) in future and thus capabilities to the novel developed codes under different TSVVs. In particular it could contribute to the TSVV 7 addressing the plasma-wall interaction in DEMO in steady-state and during transients.

Scope/Activities/Outcomes (foreseen):
In a second stage, the new code would be employed to examine specific engineering use cases for transient heat flux analysis on ITER and/or DEMO within e.g. WP PWIE and WP DES. Within this activity, the code will then be benchmarked regarding experiments in plasma gun facilities including the Ukrainian QSPA facilities as well as MAGNUM-PSI (NL) within the WP PWIE addressing in particular tungsten vapour shield dynamics. Complementary studies can be done at the MK 200 plasma gun in Russia which is an ITER partner in the ITER W divertor qualification and risk assessment processes.

At least 50% of the project would focus on the code refactoring, benefitting from the continued presence of S. Pestchanyi to ensure knowledge transfer and to identify the key components/modules of TOKES to be reprogrammed. Experience in the use of Delphi and more modern languages would be required to execute the transfer.

Competence development:

Competences required:
A Master or PhD degree in Engineering. In this case, a recipient of a relevant Master degree or PhD in a University of Technology – such as Technical/Applied Physics – will also be considered as an eligible candidate.

Work packages involved:
WPPWIE (primary), WPDES. The project would involve direct participation of the ITER Organization.

Facilities to be used:
QSPA, MAGNUM-PSI and others within WP PWIE

Mobility:
Visits to JSI, KIT and ITER are envisaged. The planning and budget should be aligned with the relevant Project Leader.

EEG21-20 Development of software tools for ECH exploitation (JT-60SA and ITER)

Contact person: Carlo Sozzi (Project Leader WPSA) – carlo.sozzi@istp.cnr.it

Background:

JT-60SA is a large tokamak device built jointly by Europe and Japan at the Naka site of QST (*National Institutes for Quantum and Radiological Science and Technology*) under the Broader Approach agreement. Machine operations for the integrated commissioning have started in September 2020 and will include a plasma phase with use of the ECH system.

In both ITER and JT60-SA the Electron Cyclotron Heating (ECH) system will play an important role starting from the first plasma where it will be used to assist the break-down phase and continuing in most of the plasma operations including wall conditioning, plasma scenario control, current drive in the plasma core, impurity and MHD control and transport studies in the subsequent research phases.

IN JT60-SA, during initial operations, 1.5MW ECRH power from two dual-frequency gyrotrons will be the only auxiliary plasma heating available. Once fully developed, the ECRH system will deliver up to 7MW to the plasma over three different wave frequencies and with wide steering capabilities for the injected beams. In ITER, the full 20MW system will be operational for PFPO-1.

The final design phase and the following exploitation of the ECH system require an extended application of a suite of performance analysis tools, with the capability of integrating both physical (plasma-EC waves interaction, heating and current drive) and technical aspects (EC beam optics in the launcher, thermal and mechanical effects, stray radiation). This kind of analysis is a constant need accompanying the preparation of the ECH application in the various plasma scenarios and operational phases of a machine in order to optimize the use of the system and to prevent and mitigate the safety issues on the launchers and vessel components related with the high power density waves.

Objectives:

In particular, with the supervision of expert personnel the trainee will be involved in several of the following activities

- Implementation of a model for the EC launchers predicting the beam characteristics for the given launcher setup;
- Combination of the use of software tools related to CAD information (e.g. Catia®) with rays and beam propagation (e.g. OpticsStudio-Zemax®, TICRA-Grasp®) in optical systems
- Development of an interface for existing beam-tracing codes (e.g. GRAY) with the experimental database;
- Coupling of the launchers' model with the beam-tracing code to achieve a prediction of power absorption and current drive in the plasma consistent with the launchers' configuration;
- possibly extension of the EC beam-tracing code's capabilities to target low absorption scenarios with greater detail in order to help the evaluation of the EC-stray load.
- Development of a user interface to ease the analysis of ECH scenarios in all their relevant features (EC profiles location in the plasma, presence of multiple resonances, shine-through hot-spots on the inner wall);
- interfacing the tools above with the ECRH control system;
- integration with discharge simulator tools and pulse editor tools

Scope/Activities/Outcomes (foreseen):

During the JT-60SA operations with plasma the trainee will have the opportunity of gain hands-on experience in

- exploiting such tools for the interpretation of data and to assist the preparation of experiments for the research phase of JT-60SA.
- Be involved in the operation of the ECH system;
- contributing to ensure that safe operation limits are determined and enforced;
- contribute to the development of the operating procedures for the ECH system

Competence development:
The grantee will receive training on the tools and codes needed to describe EC waves propagation in quasi optical systems and in plasma, in interfacing them with the data system, and will be assisted in their application during ECH operations.

Competences required:

- Master degree in Engineering, or equivalent in a relevant discipline
- Excellent interpersonal, teamwork and communication skills, including a good command of spoken & written English
- Experience in programming languages for scientific applications (e.g. Fortran, C/C++, Python, MATLAB)

Preferable also:

- Experience in ECH physics, and/or mm-wave systems design, optical systems
- Experience in any of ECRH operation in plasma, including assisted break-down, wall cleaning, or MHD control
- Experience working in an international environment
- Knowledge of optical software
- Knowledge of mechanical design software

Work packages involved: WPSA (primary), WPPrIO

Facilities to be used: JT-60SA

Mobility:
Visits to CNR (Italy), Naka (Japan) and to ITER (France) are foreseen. For extended visits the planning and budget should be aligned with the relevant Project Leader

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Annex – relevant acronyms

Work packages acronyms used in this document:
WPSAE: Safety and Environment
WPMAG: Magnets
WPDES: Design
WPMAT: Materials
WPDIV: Divertor
WPRM: Remote Maintenance
WPPWIE: Plasma Wall Interaction and Exhaust
WPW7X: Wendelstein 7-X
WPTFV: Tritium, Fuel Cycle, Vacuum
WPBB: Breeding Blanket
WPBOP: Balance of Plant
WPDC: Diagnostics and Control
WPPRD: Prospective Research and Design
WPHD: Heating and Current Drive
WPSA: JT-60SA
WPPrIO: Preparation ITER Operation
WPTE: Tokamak Exploitation

Role/PMU acronyms:
DCT: DEMO Central Team
FSD: Fusion Science Department
FTD: Fusion Technology Department
KDII: Key Design Integration Issue (*you will be hearing more of this, or maybe not anymore…*)
PMU: Programme Management Unit
RO: Responsible Officer

Facility/Activity acronyms:
ACHs: Advanced Computing Hubs
ETASC: EUROfusion Theory and Advanced Simulation Coordination
JT-60SA: Tokamak in Japan – collaboration with EUROfusion
MAGNUM-PSI: Linear Plasma Generator for Plasma Surface Interaction (DIFFER, NL)
QSPA: Quasi-Stationary Plasma Accelerator – Ukraine
TCV: (from: Tokamak à Configuration Variable) – Tokamak at Swiss Plasma Centre
WEST: (from: W Environment in Steady State) – Tokamak at CEA Cadarache
W7-X, ITER, DEMO: (*if you don't know what these are, then you should not apply!*)