OVERVIEW OF THE WEST-ITER DIAGNOSTIC INSTRUMENTATION (WIDIA) COLLABORATION ACTIVITIES

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Introduction

The ITER organization is actively preparing for the implementation of high-performance real-time measurement systems that will monitor and optimize the plasma while ensuring the operational safety of the machine's components.
With this in mind, joint activities with the CEA have been carried out for several years, consisting of testing, designing and/or developing often innovative diagnostics in a dedicated CEA laboratory or on the WEST tokamak. All of these activities are grouped under the name WIDIA (WEST-ITER Diagnostic Instrumentation agreement).

In the following a review of the main different type of WIDIA activities is presented.

1. I&C for the sensors located at the ITER diagnostics port plugs
2. Leak detections and localisation
3. Magnetic sensors calibration and tests.
4. Thermavip as a visualization tools and warning supervision.
5. Design of the Port Plug Test Facilities (PPTF) for testing the port plug and in vessel components

I&C activities

Preventing disruption

- Development of the I&C for the sensors located at the ITER diagnostics port plugs, to avoid overheating and huge mechanical stresses that could occur during disruption.
- Several thermocouples and accelerometers/strain gauges are installed in different port plugs to monitor in real time both temperatures and mechanical movements
- Correlations are made to detect potential hot points in specific locations of the machine and to benchmark the associated modelling.

Fig 1. EPP representation with temperatures and alarms in CCS HMI

Stray radiation estimate

- The injected Stray heat Radiation in the plasmas due to power injection can propagate to the ECH window through optical paths and may cause damage. Some thermocouples are used as part of the Electron Cyclotron Heating (ECH) sensors monitoring the ECH stray radiation going through the diagnostic windows.
- A specific coated ring is mounted in the front of window disc. The coating of this ring is specifically selected to react to ECH frequencies (60 & 170 GHz). When power is injected through the windows, the ring' temperature changes. Thermocouple mounted on the ring collects the temperature and with specific algorithms, it is possible to estimate live the ECH Stray Radiation and results are displayed on the HMI
- To synchronize all events, internal TCN (Time Communication Network) is used.
- Different tests were performed to estimate the potential losses in the thermocouple's wires (length and number of junction's impact) and the interactions with high energy cables running beside them (inducted current effect / high frequency perturbation).

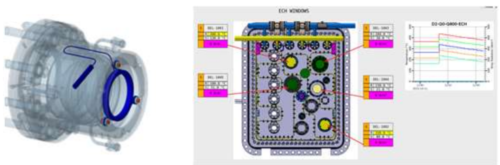

Fig 2. ECH window ring (left) and CCS HMI with stray radiation estimations (right)

Magnetic sensors calibration and tests

ITER in vessel Mirnov Coils calibration

- Accurate measurements are mandatory to ensure precise magnetic reconstruction of the plasma and, as a result, to perform plasma control and wall protection.
- On ITER tokamak, 450 mirnov coils (inductive magnetic sensors) will be installed on the inner skin of the vacuum vessel [1], [2].
- The full set of sensors was calibrated with a dedicated testbed in the frequency range of 20 Hz up to 2 MHz by measuring their response in terms of amplitude and phase shift.

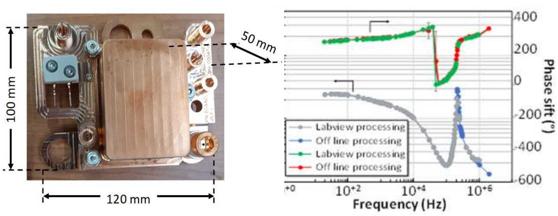

Fig 3 An integrated In Vessel Coil with key dimensions (left).Typical frequency response for an ITER tangential (variant 55.AA) IVC (right)

ITER magnetic diagnostics integrator tests on WEST

- The integrator [3] associated to any inductive magnetic sensor are crucial element in the magnetic signal measurement chain because, being prone to drift, they may compromise the accuracy of the plasma reconstruction accuracy.
- One prototype of ITER integrator was qualified on WEST during plasma experiments (800 pulses with duration up to 6'04'') and compared to WEST measurements in order to identify any drift,

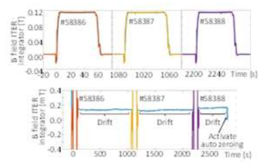

Fig 4. Measurement of the drift of ITER integrator during 3000 s (3 consecutive pulses). Top: zoom on pulse performed during the test. Bottom: zoom on integrator signal during and between pulses.

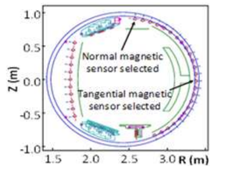

Fig 5. Cross section of WEST tokamak showing the location of the normal sensors (φ=300° and φ=60°) and the tangential sensors (φ=240° and φ=0°) connected respectively to ITER and WEST integrators

Leak detection and localization

Very encouraging preliminary results

- ITER requires also a high vacuum (1×10^{-5} Pa) to create and sustain a high-performance plasma. [3].
- A leak as small as $10^{-6}\ Pa \cdot m^3 \cdot s^{-1}$ of either air or water compromise the high-vacuum condition and could impede the operation of a fusion device [5].
- Detecting the existence of a leak can be done using pressure gauges, but identifying which component is leaking could take several weeks in present day tokamaks like ITER.
- In order to address this issue, an approach has been devised for the detection and localization of leaks using glow discharge cleaning (it is a method of conditioning vacuum vessel walls using a low temperature unmagnetized plasma discharge [6]) together with visible spectroscopy and visible imaging.
- First encouraging results of the experiments are presented as seen in Figure 6, showing the measured visible spectrum when the calibrated water leak valve is closed and when it is set to the smallest possible leak achievable by the experimental setup, which is about $\Phi_{H2O} = 6 \times 10^{-4} Pa \cdot m^3 \cdot s^{-1}$ (which corresponds to 15 mg/h)
- During the smallest water leak achievable by the experimental setup (i.e., 15mg/hr): Strong growth in the amplitude of the entire spectrum (blue line) is observed. The same conclusion is reached for air leak.

Fig 6. Experimental set up (left) and Intensity versus wavelength recorded by visible spectrometer during water leak (right)

Thermavip

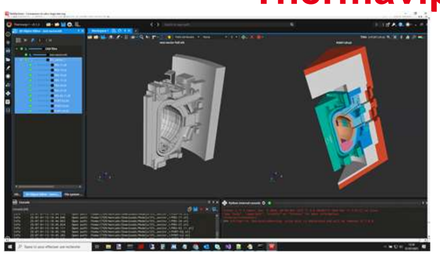

Fig 7. Experimental set up (left) and Intensity versus wavelength recorded by visible spectrometer during water leak (right)

The goal to deploy the Thermavip software in a server located in IT/XPOZ domain and adapt it to read the data required for Tokamak System Monitor (TSM) data analysis activities.
This work follows two phases:

- Make the necessary developments in a Thermavip plugin to read back UDA data from the CODAC XPOZ archive, stream live data and ensure that ITER 3D CAD models can be open, visualized and manipulated
- Demonstrate the use of Thermavip interface for the Tokamak Systems Status and warning panels as implemented in the existing TSM PDR prototype.

PPTF functional tests I&C design

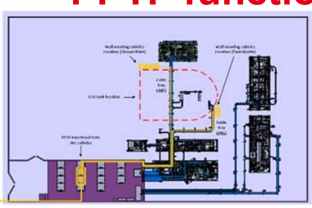

Fig. 8 PPTF test stand #4
I&C functional test equipment integration

- Port Plug Test Facility (PPTF) consists of 4 Test Stands which have been designed for performing environmental tests and providing the environmental conditions and services necessary for the functional tests of the clients (Port Plugs and In-Vessel Components).
- The conception of a PPTF system capable of acquiring and checking the basic correct behavior of any sensors inside the PP shipped to IO before its installation to the pit

Conclusions

- Many different activities conducted at CEA WIDIA Laboratory and WEST Tokamak
- Standard calibration and I&C developments
- Innovative research for detecting and localizing water/air leaks, designing PPTF I&C

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The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.