F4E R&D PROGRAMME and RESULTS ON IN-VESSEL DUST and TRITIUM

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In a tokamak vacuum vessel, plasma-wall interactions can result in production of radioactive dust and H isotopes (including tritium) can be trapped both in in-vessel material and in dust. The vacuum vessel represents the most important confinement barrier to this radioactive material.

In the event of an accident involving ingress of steam to the vacuum vessel, hydrogen could be produced by chemical reactions with hot metal and dust. Hydrogen isotopes could also be desorbed from in-vessel components, e.g. cryopumps. In events where an ingress of air to the vacuum vessel occurs, reaction of the air with hydrogen and/or dust therefore cannot be completely excluded.

Due to the radiological risks highlighted by the safety evaluation studies for ITER in normal conditions (e.g. in-vessel maintenance chronic release) and accidental ones (e.g. challenge of vacuum vessel tightness in the event of a hydrogen/dust explosion with air), limitations on the accumulation of dust and tritium in the vacuum vessel are imposed as well as controls over the maximum extent of the quantity of accidental air ingress

ITER IO has defined a strategy for the control of in-vessel dust and tritium inventories below the safety limits based primarily on the measurement and removal of dust and tritium.

In this context, this paper will report on the efforts under F4E responsibility aimed at developing a number of the new ITER baseline systems.

In particular the paper will provide the status of:1) Tasks launched on diagnostics (“Divertor Erosion Monitor”, “Capacitive Diaphragm Monitor” and “Hot Dust Measurement Using Steam Injection”) 2) On-going R&D programme (experimental and numerical simulation) at FZK, CEA and ENEA on in-vacuum vessel dust-H explosion and dust mobilization 3) Experiments to enrich the data about the effectiveness of desorption of tritium from Be at 350°C (divertor baking aiming to release significant amount of tritium trapped in Be co-deposit).