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Reduction of Critical Heat Flux due to steep power transients on PFCs

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Recent experiments carried out at the High heat flux test Facility (HHFTF) (1)(2) to study Critical heat Flux (CHF) margins for Plasma Facing components (PFCs) suggest a possible dependence of the CHF on the temporal power profile, leading to a reduction of CHF limits that could impact design margins for Plasma Facing components. This paper reports possibly for the first time, experimental results dedicated towards assessment of CHF limits of PFCs that indicate a transient dependence of CHF for a fixed set of thermal hydraulic conditions. The calculated ratio of the transient CHF to the steady state CHF values is ˜1.13 which could be significant, considering the safety factor of 1.4 required for the design of Plasma facing components (PFCs). Two different values of CHF have been obtained during the experiments –one corresponding to the 'steady state'CHF where the beam power is increased stepwise, till steady state conditions are achieved during each step upto the point where a sudden temperature excursion is obtained, and the other corresponding to a single shot at a fixed power. The Assessment of Critical Heat Flux (CHF) for water cooled PFCs is crucial in design of Divertors (3) in tokamaks, including the ITER Divertor to ensure safe operation during normal and off normal events of the machine. Several correlations have been derived based on experiments (4)(5) to accurately estimate the CHF for a range of thermal hydraulic parameters and are used in the design of PFCs. In most of the experiments performed, the CHF is obtained by gradually increasing the power stepwise till a sudden temperature transient is observed. However sudden changes in the thermal hydraulic conditions could also induce CHF to occur prematurely, which could impact the design of the component. While most of the studies have focussed on micro and mini channels and startup for BWRs, there is no reported study at thermal hydraulic conditions relevant to ITER.
The HHFTF at IPR is designed for high heat flux testing and qualification of Plasma Facing Components (PFCs) for the Indian Fusion programme. A copper test mockup of dimensions 41 x 26 x 400 mm with coolant channel diameter of 12mm was used for the study. Water at pressure of 9 bars, bulk temperature of 90° C and flowrate of 45 and 37 lpm was provided by the High pressure high temperature water cooling system (HPHTWCS). For experiments performed at 45 lpm, the CHF value in the 'steady state'condition was 8.45 MW/m2, while a value of 7.53 MW/m2 was obtained for the same thermal hydraulic parameters for a power transient of 16 s. The estimated CHF from Tongs-CEA correlation is 7.8 MW/m2. Several diagnostics were used for the detection of CHF –Single colour Pyrometer, two colour Pyrometer, Infrared Camera (IR), Differential pressure transmitter (DPT), acoustic sensor (AS) and thermocouples (TC). The pyrometer and IR camera were calibrated using a cavity blackbody source for accurate temperature measurements. Two Resistance Temperature Detectors (RTDs) were used to measure the absorbed power. DPT and acoustic sensor outputs indicate large fluctuations indicative of sudden formation of vapour bubbles during sudden power transients, as compared to gradual increase of power, leading to CHF at lower powers.

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The occurrence can be attributed to two phase flow instabilities (6) which is most likely due to the onset of Flow instability (OFI) (7). This was accompanied initially by high frequency noises followed by large vibrations at the test section corresponding to the sudden fluctuation in the DPT readings. Similar observations of noise have also been reported earlier under certain conditions, without occurrence of any instabilities (8). The exact mechanism is not understood well and further experiments will be performed to understand this. Experiments are also planned at ITER relevant conditions to check if a similar phenomenon is observed as this could have an impact on the design of the ITER Divertor and the CHF safety margin to be maintained for the divertor design (9).

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