Recent challenges in Reactor Core Thermal-Hydraulics

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Abstract

Reactor core thermal hydraulics has always been one of the key components for the safety of a nuclear reactor. Continuous efforts have been devoted to the investigation and understanding of its basic physical phenomena and safety analysis. Complicated and unique features of non-uniform heating (both axial and radial), the open but narrow subchannel geometry (especially for PWR), the axially periodic mixing promoted by MVG at a set span g_{sp} (grid span), and most importantly, the potential of non-local effects and flow regime transition resulting from strong dynamic interaction among neighboring subchannels' with drastic differences in thermal-hydraulic conditions it is rather challenging to obtain accurate computations of local conditions under these various geometries and broad range of operating conditions. Recently, the subjects of reactor core thermal-hydraulics have gained more attention in the international community through three international seminars (ISACC 2013 Xian China, ISACC 2015 Shenzhen China, and IS-ReCTHA 2018 Lake Lecco, Italy) on reactor core thermal-hydraulics with special focuses on Computational Fluid Dynamics (CFD) modeling, subchannel analysis, and rod bundle Critical Heat Flux (CHF) experiments.

This paper presents a brief summary of some ongoing key challenging issues concerning various aspects of reactor core thermal-hydraulics including CFD modeling, subchannel analysis, rod bundle experiments, rod bundle CHF modeling predictions that are critical for understanding the underlying fundamental physical phenomena, the advancement in reactor core thermal-hydraulics, and its safety applications for commercial reactors and nuclear power plants. The main challenges mainly involve:

1) Challenges in CHF Modeling and Safety Analysis(Local vs Non-local CHF phenomena, Non-uniform heating Homogeneous Nucleation Model)
2) Challenges in Subchannel Analysis (Non-representative mixing vane grid model, prototype and ocean conditions)
3) Challenges in Rod Bundle Experiments (Non-representative high uncertainty uniform rod bundle CHF, exit reflood quenching effect, bowed rod CHF test, non-representative prototype test grid, non-conservative indirect heater, heat loss, Freon scaling issues)
4) Benchmarking of rod bundle thermal-hydraulic facilities (An open independent international benchmarking CHF test facility)
5) Challenges in CHF Correlations Development and Safety Analysis (BO/BO vs mDNBR/mDNBR approach)
6) Challenges in CFD modeling (Multiphase CFD modeling including CHF simulations)

In the above issues, for the first time after over 30 years of practices, the hidden non-conservative non-typical CHF physical phenomena due to non-representative exit quenching effects for both

uniformly heated and potentially exit peaking power profile rod bundle CHF tests are identified. Recently , another topicbrought to light is the issue of non-representation in using Freon as a surrogate for water in rod bundle CHF tests due to non-compatible scaling under two phase conditions. In particular consideration for the non-uniform high axial power peaking characteristics associated with PWR fuel, especially in short bundle fuel assemblies, a new CHF mechanistic model, the Non-uniform heater Homogeneous Nucleation Model (NHNM), was introduced along with a proposal for new or modified boiling curve due to a jump from a point between Onset of Nucleation Boiling and Onset of Significant Void to transition boiling or boiling crisis. Meanwhile, a new application of Distributed Resistance Model (DRM) in subchannel code proposed for rod bundle with mixing vane grids to account for the contribution of diversional cross flow mixing in spacer grids is introduced in detail.