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A review of sub-channel thermal hydraulic codes for nuclear reactor core and future directions

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ABSTRACT

Thermal hydraulic analysis of nuclear reactor core is mainly performed using the sub-channel analysis codes to estimate different thermal hydraulic safety margins. The safety margins and the operating power limits of nuclear reactor core under different conditions of primary system i.e. system pressure, coolant inlet temperature, coolant flow rate and thermal power and its distributions are considered as the key parameters for sub-channel analysis. Considering the complexity of rod bundle geometry, different turbulent scales and due to their limitations of computational resources, performing the full scale computational fluid dynamic (CFD) analysis of nuclear reactor core is a cumbersome and time consuming task. Hence, the thermal hydraulic safety margins of most of the reactors operating in the world are carried out using the sub-channel analysis codes. In these codes, the governing equations of mass, momentum and energy are solved in control volumes which are connected in both radial and axial directions. The flow distributions in the rod bundle geometry are estimated by considering lateral momentum balance and the inter channel mixing models to account for the cross flow between the adjacent sub-channels. The accurate estimations of the local conditions of the sub-channels are required to predict fuel temperature, critical heat flux ratio (CHFR) and critical power ratio (CPR). In this paper, various experiments conducted on the different geometries of sub-channels related inter sub-channel mixing in rod bundles are identified. A comprehensive review of sub-channel thermal hydraulic codes used for the analysis of nuclear reactor core is presented. This review covers various aspects of experimental, analytical and computational works related to rod bundles carried out in the past and brings out future directions derived from earlier research works.

1. Introduction

The contribution of global energy requirement by the nuclear power is going to increase in future, keeping in view on the effect of greenhouse gas on our environment. Today about 440 nuclear power plants are operating around the world and 120 nuclear reactors are under construction stage. Many of the reactors are already operated for quite a long time, and ∼100 reactors are in shutdown state (IAEA-RDS-2/36). The capacity of the nuclear power plants is also increasing to reduce the capital cost with enhanced safety features. Following the nuclear emergencies caused by Three Mile Island (1979), Chernobyl (1986) and Fukushima (2011), more importance is laid on passive core cooling features. The safety of nuclear reactors is to be ensured under normal operation, operational transients, anticipated operational occurrences, design basis accidents (DBA) and under extreme emergency situations by incorporating the engineered safety systems by passive means. The lessons learnt from the previous experiences from analysis, experiments and accidents of nuclear systems are to be considered in the new design of the reactors, especially, the reactor core and its associated systems. The safety of the nuclear reactor system under all conditions of core can be ensured mainly by the thermal hydraulic analysis (Sha, 1980). Experiments and numerical simulations of nuclear reactor core are required to ensure reactor safety as briefed by Yadigaroglu et al. (2003). Thermal Hydraulic analysis of nuclear reactor core is carried out using the sub-channel analysis code to estimate the various thermal hydraulic safety parameters like critical heat flux (CHF) ratio, critical power ratio (CPR), fuel center line temperature, fuel surface temperature, sub-channel maximum temperature and bulk coolant outlet temperature (Chelemer et al., 1972). Critical heat flux ratio (CHFR) and hence, the critical power ratio and fuel center line temperature are the main parameters limiting the maximum operating power of the reactor (Cheng and Muller, 1998). Accurate calculations of CHFR, CPR and maximum fuel temperature are of prime importance to ensure the safety of the reactor under different states of the core (Chang et al.,

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2003). The estimation of CHF is done based on either local conditions or based on energy balance of the critical sub-channel (Weisman and Ying, 1984). The CHF values are very much dependent on the local fluid conditions such as pressure (Chun et al., 2001), temperature, mass flux, quality and geometrical parameters (Seung et al., 2001; De Crecy, 1994; Cheng and Muller, 2003; Groeneveld et al., 1996; Hejzlar and Todreas, 1996). There is a natural mixing due to turbulence, pressure gradient across the sub-channels and forced mixing between sub-channels owing to diversion cross flow or flow sweeping by spacer geometry/wire wraps (Rogers and Rosehart, 1972; Rogers and Tahir, 1975; Roidt et al., 1973, 1974; Roidt et al., 1980). Hence, there is an exchange of the mass and enthalpy between the neighboring sub-channels which significantly alters the local fluid conditions of the sub-channels (Rowe and Angle, 1967; Rowe and Angle, 1969; Lewis and Buettiker, 1974) that are used to estimate the CHF and temperature of the fuel pins (Khabensky et al., 1998; Kim et al., 2000; Lee, 2000). A survey of different sub-channel codes used for reactor design and related experimental studies are presented (Rowe, 1967, 1970; Webb, 1988; Burns and Aumiller, 2007).

The paper presents, a detailed literature survey related to sub channel analysis methods, codes used for sub-channel analysis and coolant inter sub-channel mixing. A comprehensive review of sub-channel thermal hydraulic codes for nuclear reactor core is also presented on various aspects of experimental, analytical and computational studies related to the rod bundle sub-channel analysis, carried out in the past. Further, perspectives for the future directions are derived from the earlier research. The various experimental and analytical works conducted by researchers on different geometries of the sub-channels are identified. The application of CFD technique to rod bundle flow distributions and heat transfer is also brought out from open literature and briefly discussed in different subsections with major findings and the gap areas.

2. Sub-channel thermal hydraulic analysis

Reactor core fuel assemblies are of different shapes typically, like, square, hexagonal or circular in cross section as shown in Fig. 1 (Ginox, 1978; Todreas and Kazimi, 2001). Typical Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR) and Liquid Metal Fast Breeder Reactor (LMFBR) fuel pins are assembled in the form of regular arrangements/patterns and are called fuel assemblies/bundles. Inside the fuel assembly, fuel pins are arranged either in square or triangular lattice configuration. In reactors like pressure tube type reactors (PHWR), fuel is in the form of small bundles without fuel channel cover but the coolant flow is restricted within the coolant channel. Old PWR designs as well as the recent large commercial PWR designs do not contain fuel assembly channel cover or might contain the openings along the channel in the flow direction between the assemblies. This kind of arrangement causes the inter channel mixing of coolant between assemblies along the axial direction. Deliberately the mixing is allowed between assemblies to reduce the hot channel temperatures. These fuel assemblies constituting the nuclear reactor cores are generally called open channel core. In some configurations, the fuel assemblies are fully covered by the zircalloy channels which prevent the inter channel mixing of the coolant among the assemblies. This kind of fuel assemblies constituting the nuclear reactor cores are called closed channel core. Thermal hydraulic analysis of nuclear reactor core is carried out to estimate the detailed flow and temperature distribution within the fuel assembly. In case of open channel assemblies, the entire core or symmetric sector of the core is to be analyzed to estimate the flow and temperature distribution within the core. In the case of closed channel assemblies, individual assemblies can be analyzed separately after getting the assembly flow distribution.

The thermal hydraulic safety analyses of nuclear reactor are performed in two ways. First, the system level thermal hydraulic analysis codes like RELAP, RETRAN, ATHLET are used to get the system behaviors under different steady state and transient operating conditions. The results of this analysis give the boundary conditions for the core level/component analysis. The detailed analysis of the reactor core is performed using the sub-channel thermal hydraulic codes like COBRA (Rowe, 1967), VIPRE (Stewart et al., 1993) to estimate the thermal hydraulic safety margins of nuclear reactor core under steady state and transient conditions. A sub-channel is defined as a flow passage formed between number of rods or some rods and wall of channel/shroud tube. The sub-channels can be formed by either coolant centered sub channels or the rod centered sub-channels as shown in Fig. 2. It can be either square or triangular in shape depending on the type of fuel pin arrangements either in square or in triangular pitch arrangements of the fuel assembly. The concept of sub-channel analysis method is an important tool for predicting the thermal hydraulic performance of rod bundle nuclear fuel element. It considers a rod bundle to be a continuously interconnected set of parallel flow sub-channels which are assumed to contain one dimensional flow coupled to each other by cross flow mixing. The axial length is divided into a number of increments such that the whole flow space of a rod bundle is divided into a number of nodes.

The sub-channel thermal hydraulic analysis basically solves the conservation equations of mass, momentum and energy on the specified control volumes. The one dimensional control volumes are connected in both axial and radial directions to get the three dimensional effect of the core. Performing the detailed CFD kind of analysis for the entire

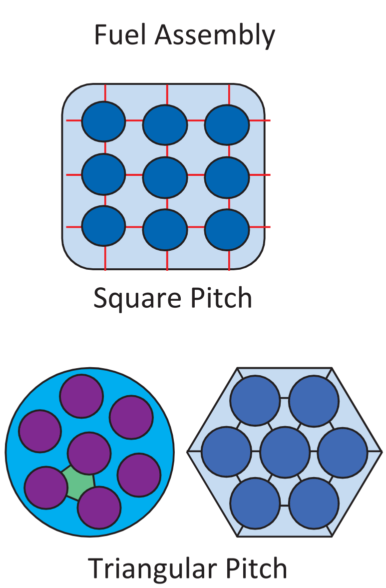

Fig. 1. Typical shapes of fuel assembly.

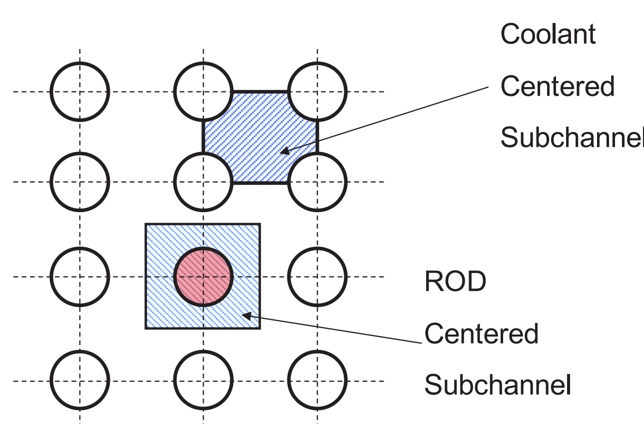

Fig. 2. Definition of fuel assembly sub-channels.