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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

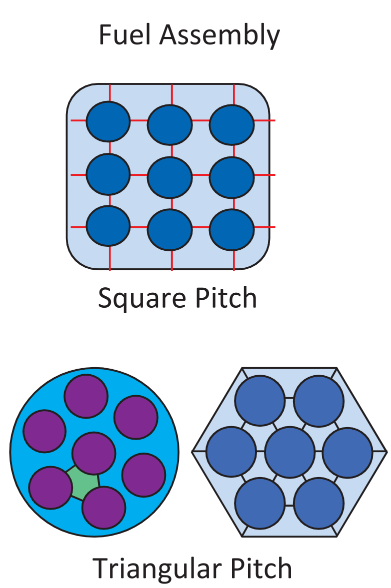

Fig. 1. Typical shapes of fuel assembly.

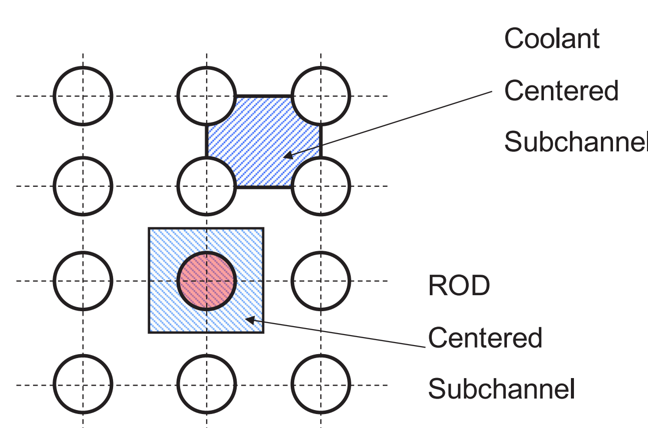

Fig. 2. Definition of fuel assembly sub-channels.

core or a single fuel assembly is very difficult even with the present-day computing resources. Hence, most of the nuclear reactor cores are designed based on the sub-channel analysis codes. The detailed predictions of the sub-channel codes are greatly improved by taking into consideration effect of the inter channel mixing model to account for the flow exchange between the fuel assembly sub-channels. The relation between sub-channel flow rate which is the mass flow rate in an axial direction through sub-channel area and diversion cross-flow which is the mass flow in a transverse direction resulting from local pressure differences between two sub-channels is strongly governed by momentum balance in a transverse direction. A correct formulation of momentum equations and good knowledge of the mixing process between sub-channels are an absolute necessity, for obtaining reliable predictions of coolant local conditions using sub-channel calculations.

In the basic design of the fuel assembly, the spacers are provided at different locations of the fuel pin, to avoid hotspots, to maintain proper coolant flow area and to control flow induced vibrations (FIV). The spacers also offer more pressure drop as compared to the bare assembly which improves the coolant redistribution inside the fuel assembly. Most of the improvements of the spacers were considered to reduce the pressure drop and improve the mixing between sub-channels by providing mixing vanes in the spacer. Due to the geometrical complexity of the spacer and the complexity of flow distribution downstream of the spacer, the spacer models are derived from both experiments and CFD analysis to account for the proper flow diversion in the lateral momentum balance equation. These models are implemented in the sub-channel analysis codes to include the effect of different type of spacers on flow distribution. In this way, sub-channel type of thermal hydraulics analysis codes are applied to the practical design of the nuclear reactor core even today. In case of transients and at higher power levels near the critical heat flux conditions the mixing within the assemblies are greatly affected by the local conditions due to two phase flow formations. Hence, the proper predictions of the local conditions are important in the estimation of the critical power of the assembly and hence, the temperature of the fuel pin surface.

2.1. Literature on inter sub-channel mixing

The literature reviews carried out on the topic of turbulent inter sub-channel mixing is shown in Fig. 3. It indicates that a vast amount of literature is available in the form of laboratory reports and journal articles.

A recent review on critical heat flux (Cheng and Muller, 2003) in a water cooled reactor shows that, the empirical models describing interchannel mixing affect the sub-channel conditions and a strong effect of the turbulent mixing coefficient on the calculated sub-channel conditions is observed. The accuracy of the CHF prediction method in the rod bundle geometry is coupled with the accuracy of determining sub-channel flow conditions. Also, the use of a constant mixing factor assigned without regard to differences in power distributions is improper as the effect of mixing on enthalpy rise depends on the radial thermal gradient between the channels as well as flow conditions (Chelemer et al., 1972, 1977).

Study on CHF and turbulent mixing (Cheng and Muller, 2003) suggest that despite large amount of theoretical and experimental studies on CHF and turbulent mixing, the knowledge of the precise nature of these phenomena is incomplete and that for each specific design, experimental investigations should be performed and validated prediction methods must be derived. Further, in their conclusion, detailed literature survey has indicated that the experimental and theoretical works on CHF and turbulent mixing in tight hexagonal rod bundles under high pressure and high mass fluxes are very limited.

During the initial period (1960’s to 1970’s) of reactor designs, the reactor core conditions are predicted based on the sub-channel analysis with suitable inter-channel mixing models derived from the experimental data of their respective geometries or simulated tests. The survey further shows that recent research is towards the phenomenological understanding of the turbulent flow structure in the rod bundles with the help of computational fluid dynamic techniques (Biemüller et al., 1996; Chang and Tavoularis, 2008; Tsutomu et al., 2010; Yan et al., 2012).

The topic of research articles in the early stages of sub-channel thermal hydraulic analysis were concentrated mainly on experimental and analytical works devoted to the development of the sub-channel analysis codes. Further, experimental and research works were oriented towards development of various correlations of mixing models and verification/validation of sub-channel analysis codes with experiments. In the recent past due to increased computational resources with advancement in CFD techniques and measuring instrumentations, the current research is devoted to simulation of experiments of rod bundle geometry using CFD techniques. These efforts were towards gaining confidence in CFD predictions and also deriving the models from CFD

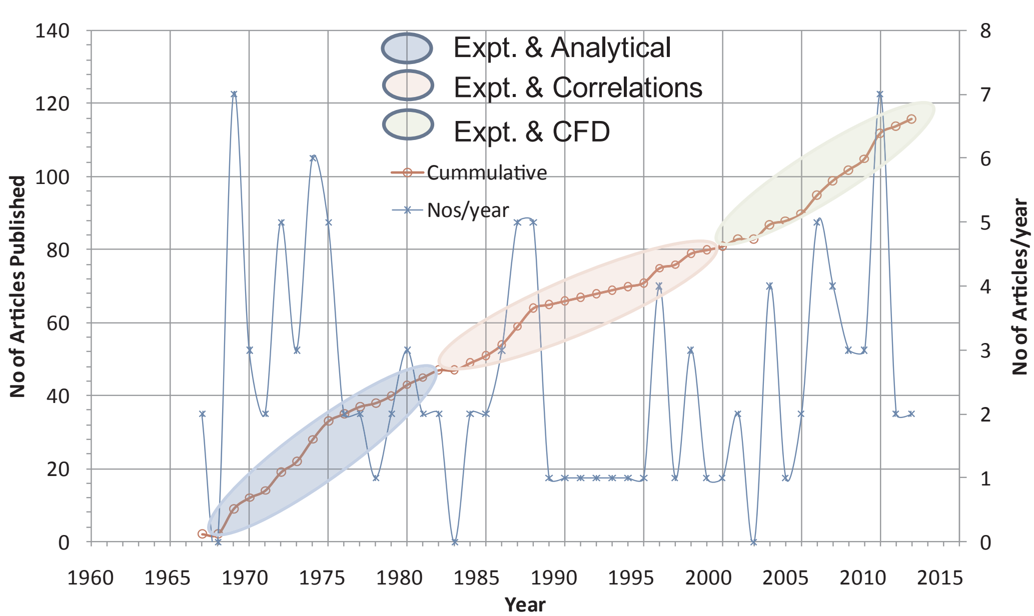

Fig. 3. Research articles published in turbulent Inter sub-channel mixing.

Table 1
Different mixing phenomena in rod bundles.

Phase	Natural Mixing	Diversion Cross Flow
Single Phase	Molecular Diffusion Turbulent Mixing	Pressure Difference Forced Diversion
Two Phase	Void Drift	Pressure Difference Forced Diversion

for new fuel geometries and thereby, reducing the time required for design and optimization of fuel assembly (Hu and Fanning, 2011) and towards the improvement of safety and conducting the minimal number of experiments to save the resources and time. Also, CFD techniques help to get the detailed flow and temperature distributions within assembly as in experiments.

2.2. Inter sub-channel mixing in rod bundles

In order to improve thermal hydraulic characteristics of the nuclear reactor core, a considerable amount of research has been carried out to obtain improved understanding of coolant flow and enthalpy distributions in rod bundle geometries. One form of fundamental research has been the study of the mixing process between complex geometries of sub-channels.

In analyzing the effect of mixing on rod bundle temperature and pressure gradient, it has generally been assumed that the mixing process is the result of several components as shown in Table 1. Further, the natural mixing describes turbulence in bundles of smooth bare rods (without protuberances) and which include both turbulent and diversion cross flow mixing.

Turbulent mixing which results from the oscillatory component of flow in a transverse direction between two sub-channels and it can be characterized by the eddy diffusivity of momentum. On the other hand, the diversion cross flow mixing is the rate of mass flow in a transverse direction though the gap between two sub-channels caused by radial pressure gradients. This flow contributes to the variation of sub-channel flow rate in an axial direction. In the case of forced mixing, the sub-channel mass exchange is induced by the presence of spacers in the rod bundle. The flow scattering due to the non-directional mixing effect is generally described in literature associated with grid spacers which break up streamlines; the turbulence intensity increases immediately downstream from the device and the flow sweeping is referred as directed cross flow effect associated with wire wrap spacers or grid spacers with mixing vanes which give a net cross flow in a preferred direction.

2.2.1. Mixing under single-phase flow conditions

Much research has been devoted to the study of turbulent flow processes and improved understanding of mixing phenomena. Most of the experimental works have been focused on quantifying the mechanisms individually and their dependency. First, the relative importance of these processes in rod bundle performance varies significantly with bundle geometrical characteristics, particularly, gap spacing and flow parameters. Secondly, though turbulence interchange is present in all situations, major emphasis has been laid on determining turbulent cross flow (diversion and sweeping) because of its most inherent property of momentum and energy transfer. It improves the thermal hydraulic performance of rod bundles and serves as an important mechanism for equalizing temperatures throughout the bundle. An excellent review of the related aspects of turbulent mixing and diversion cross flow is given by Rogers and Rosehart (1972); the reviewers describe the circumstances in which crossflow mixing is developing, particularly, in the entrance region of bundles, region of beginning and development of boiling crisis in the various sub-channels and region of physical distortion of the bundle elements.

2.2.2. Mixing under two-phase flow conditions

Experimental data available in the literature about the mixing rate under boiling flow conditions are limited. Some investigations have been made with air-water mixtures to explain mechanisms of mixing in two-phase flow and to indicate the effect of various parameters, such as, quality and mass flux on turbulent interchange rates. However, the principal information has been developed by Rowe (1967) through the use of thermal hydraulic code COBRA which constitutes a significant contribution to the knowledge of boiling flow behavior in rod bundles. It has been observed that boiling turbulent interchange appears to be a function of the channel geometry, quality and flow regime, with a maximum at low qualities in the transition region from bubbly to annular flow.

2.3. Inter channel mixing models

2.3.1. Equal mass exchange turbulent mixing model (EM)

In the EM model (Hwang et al., 2000), it is assumed that, the fluctuating mass flow rates between the interacting sub channels are identical (i.e. $\rho\nu_{ij}' = \rho\nu_{ji}'$). Hence, there is no net transfer of mass due to the turbulent mixing and the diversion cross flow is the only mechanism transferring mass between the sub channels.

2.3.2. Equal volume exchange with void drift (EVVD)

In two phase flow conditions, a substantial amount of net mass transfer has been experimentally observed in addition to the energy and momentum transfer. In order to explain this behavior, a model is devised by considering the equal-volume exchange of two-phase mixture between sub-channels ($\nu_{ij}' = \nu_{ji}'$), and assuming that two-phase turbulent mixing is proportional to the non-equilibrium void fraction gradient. The EVVD models (Hwang et al., 2000) consider the transverse mixing attributed to fluctuating flow at the interface and void drift between neighboring sub-channels. The void drift phenomenon is characterized by a strong trend towards the equilibrium void distribution, i.e., the vapor has a strong affinity for the sub-channel with the larger cross-sectional area and higher velocity (Sadatomi et al., 1994; Kawahara et al., 1997). Further, the effect of buoyancy on horizontal bundle under two-phase turbulent conditions and buoyancy drift is studied by Carlucci and Rowe in 2004.

3. Development of sub-channel analysis codes

An extensive literature review on the development and use of the sub-channel analysis codes applied to the nuclear reactor core design has been carried out from the existing literature which includes journals, books, laboratory reports of different research organizations and thesis of other investigator on the similar research topics. Table 2 gives the comprehensive list of sub-channel analysis codes like HECTIC, ENERGY, SUPERENERGY, COBRA (-I, II, IIIC, IV), CANAL, HAMBO, FLICA, THINC, VIPRE and COBRA-FLX developed at different times. These codes are still under development and undergoing improvements. They are used for design calculations and thermal hydraulic analysis of different reactors since 1960. Among all these codes, COBRA type of sub-channel codes were applied largely to many type of reactors. It has been modified continuously (COBRA-WC (George et al., 1980), COBRA-SC, COBRA-LM, COBRA-EN (Basile et al., 1999), COBRA-TF, COBRA-FLX (AREVA NP Inc, 2010) and many more (DOE/ET-0009, 1977)) to improve the predictions of local conditions under single phase as well as two phase and hence, the thermal hydraulic margins of core. Traditionally sub-channel analysis was applied in addition with modeling the local effects from the experimental data to account the local flow variations. Recently, with a significant increase in the computing power and the numerical techniques, CFD kind of approach is also made possible but still applying CFD to the whole core is cumbersome task. To some extent, CFD analysis of sub-channel can be used for deriving the modeling parameters instead of experimental setup. The typical

Table 2
Comprehensive list of sub-channel analysis codes.

Sub-Channel Code	Version/Year	Organisation/Country	Application	Reference
HECTIC	I/1961,II/1962	Aerojet Nucleonics/USA	GAS COOLED, PWR	Reynolds and Kattchee, 1961; Kattchee and Reynolds, 1962
SCEPTIC	1971	EIR/Switzerland	GAS / PWR	Eriksson, 1971
COBRA	I/1967, II/1970, III/1971, IIIC/1973, IV-I/1976, IV/1977	BNWL, USA	PWR, LMFBR	Rowe, 1967, 1970, 1971, 1973; Wheeler et al., 1976; Stewart et al., 1977
VIPRE	1/1993, 2/1994	PNL/EPRI/USA	PWR	Stewart et al., 1993; VIPRE-02, 1994
SCRIMP	1977	EIR/Switzerland	GAS / PWR	Huggenberger, 1977
MATTEO	1973	MIT/USA	BWR	Forti and Gonzalez-Santalo, 1973
WOSUB	1978	MIT/USA	BWR	Wolf et al., 1978
CANAL	1979	MIT/USA	BWR/PWR	Faya et al., 1979
ASSERT	4/1984	AECL/CA, USA	CANDU	Tye et al., 1994
ENERGY	I,II,IV/1975	MIT/USA	LMFBR	Chen and Todreas, 1975
SUPERENERGY	I, II/1980	MIT/USA	LMFBR	Basehore and Todreas, 1980
MISTRAL	II/1972	KFK/Germany	PWR	Baumann, 1972
DIANA	1974	Hitachi/Japan	PWR	Hirao and Nakao, 1974
THERMIT	II/1981	MIT/US	LMFBR	Kelly et al., 1981
MATRA	1999	KAERI/S. Korea	PWR	Yoo et al., 1999
MATRA-LMR	2002	KAERI/S. Korea	LMFBR	Kim et al., 2002
HAMBO	1967	AEEW/UK	PWR	Bowring, 1967
FLICA	III, IV/2000	CEA/France	LMFBR	Toumi et al., 2000
THINC	I/1968,II/1972,IV/1973	USA	PWR	Chu et al., 1973
SASS	I/1966	Sweden	CANDU	Pierre, 1966
SABRE	1, 1984, 4, 1992	France	LMFBR	Macdougall and Lillington, 1984; Dobson and O'Neill, 1992
SACoS	2012	Jiaotong University, China	SCWR	Khurrum et al., 2012
LYNXT	1 & 2/1976	B & W/US	PWR	BAW-10156-A
TORC	1975	CE/USA	PWR	CENPD-206-NP
SABENA	1985	Japan	LMFBR	Ninokata and Okano, 1985
NASCA	2001	Japan	PWR/LMFBR	Ninokata et al., 2001
MONA	2003	KOREA	LMFBR	Nordsveen et al., 2003
SUBCHANFLOW	2010	KIT/Germany	PWR	Imke et al., 2010
ATHAS/OE	2016	Jiaotong University, China	PWR	Pan Wu et al., 2016

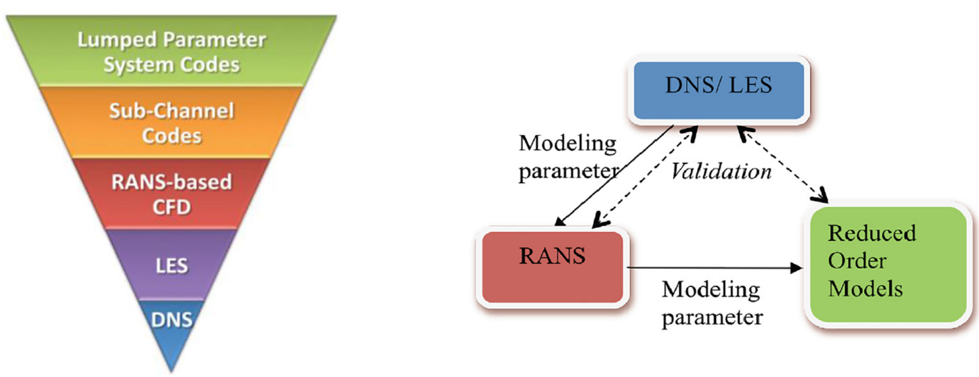

Fig. 4. Thermal Hydraulic Modeling of approaches of Nuclear Reactor systems and components (Hu and Fanning, 2011, ANL/NE-11/35).

approach used in the analysis from sub-channel analysis to CFD for obtaining the modeling parameters, followed by other researchers are systematically depicted in Fig. 4 (Hu and Fanning, 2011). Further, the experimental, analytical and computational works carried out for the development of sub-channel codes to predict the thermal margins are discussed in this section.

3.1. Experimental works of rod bundles

The detailed literature survey indicates that vast amount of research works were taken up on flow and heat transfer characteristics of rod bundle geometries in the form of experimental and analytical estimations. These studies were oriented towards the improved understanding of the coolant mixing inside the fuel assembly sub-channel and hence, the safety of nuclear reactor core under the different operating conditions. This section describes the experimental investigations carried out by different researchers from simple sub-channel geometries to rod bundle geometries. The list of different experimental studies of nuclear fuel bundle sub-channels is given in Table 3. The extensive experimental works of rod bundle sub-channels were aimed at the measurement of pressure drop across the rod bundle as well as in sub-channels (Novendstern, 1972; Maubach and Rehme, 1973; Rehme, 1980a, 1989; Grover and Venkat Raj, 1980; Hudina and Huggenberger, 1986). The major studies in rod bundles includes: measurement of flow velocity and its distribution (Ibragimov et al., 1967; Chieng and Lin, 1979; Neti et al., 1983), secondary flow vortices inside the sub-channel (Vonka, 1988), measurement of wall shear stress (Eichhorn et al., 1980) and turbulence quantities, the structure of turbulence and its implications on natural mixing. The pressure drop and flow distributions were measured on bare rod bundles, bundles with grid spacers, wire wrap spacers and mixing vanes (Collingam et al., 1971). Experiments were also conducted for determination of turbulent exchange coefficients in triangular, square pitched and asymmetric rod bundles by measurement of sub-channel cross flow due to pressure gradient and diversion cross flow (Roidt et al., 1974; Yue et al., 1991). The enhancements in mixing in the rod bundles are also investigated in detail. The increase in

Table 3
List of experimental studies of nuclear fuel bundle sub-channels.

Investigator	Geometry	P/D	Re	Measured Data	Methodology
Walton (1969)	Tri., array, T-T	1.052	1900–90000	Concentration	Water, Air tests, Dye injection
Tachibana et al. (1969)	7 Pin bundle	1.125	8000–45000	Heat Transfer Factors	Naphthalene sublimation
Skinner et al. (1969)	6 equally spaced rod	1.42	20000–80000	Velocity, Pressure	Nitrous oxide tracer, Pitot static probe.
Van Der Ros and Bogaart (1970)	Compound rect. channel	1.05	5000–30000	Pressure, velocity, Temperature	Pitot tubes, thermocouple, thermal tracer.
Rowe et al. (1974)	Sq. 3 × 3	1.125 1.25	50000–200000	TI auto.corr. functions	LDV
Trupp and Azad (1975)	Tri., array	1.2–1.5	12000–84000	Axial velocity, wall shear stress, Reynolds stress, eddy diffusivity	Pitot static probe, hot wire anemometry.
Carajilescov and Todreas (1976)	Sq. Channel	1.12	27,000	Axial velocity, Turbulent Kinetic energy, Reynolds stress	LDA
Rehme (1978)	Four Parallel rods	1.12	87000	V,TauW, k	Pitot tubes, preston tubes, HWA
Seale (1979)	Row of 3, 4, 5 rods	1.833 1.375 1.1	34369–299603	T and velocity distributions	Pitot temperature probe, manometer
Hooper (1980)	Sq. array	1.15	48000–156000	TauW, V,K	Pitot tube, Preston tube, HWA
Renksizbulut and Hadaller (1986)	Sq. array	1.15	500000	TauW, V,TI	Preston Tube, LDA
Aly et al. (1978)	Eq. triangular array	1.2–1.5	53000–107000	V, TauW, sec. flow, friction factor	Pitot tube, X-wire probe, HWA
Vonka (1988)	Tri. Array	1.3	60000 175000	Sec. flow vortices	LDA
Hooper and Rehme (1984)	Four par. Rods 6 sq. array	1.026–1.118 1.107	54600–105000 22600–207600	V, TauW, Rey. stress, TI	Pitot tube, Preston tube, HWA
Rehme (1987)	Four.par. rods	1.036–1.40	76000–119000	Rey. Stress, TI, TauW	HWA, Preston tubes.
Moller (1991)	Four. Par.	1.036–1.148	85000	V, Tauw,TI,pressure fluctuations	Pitot tube, HWA, microphone
Moller(1992)	Four. Par.	1.036–1.223	60880–148100	V, Tauw,TI	Pitot tube, HWA, microphone
Wu and Trupp (1994)	Single. Rod Trapezoidal	1.039–1.240	21300–52700	Fluctuating V, PSD	X-Wire probe.
Meyer and Rehme (1994)	Comp. rect.chn.	1.06–1.25	250000	V, TauW, Rey. Stress, TI, TKE	Pitot tube, preston tube, HWA
Wu (1995)	Asymmetric rod in trapiezoidal duct.	1.22, 1.039–1.079	26300	V, V', Energy spectra	Pitot tube, preston tube, HWA
Krauss and Meyer (1996)	Wall. Sub-channel in 37 rod bundle	1.12	65000	V,T,TauW, Turbulent velocity, heat flux.	Pitot tube, preston tube, HWA, Sheathed thermocouple.
Krauss and Meyer (1998)	Tri. Sub-channel in 37 rod bundle	1.12	65000	V,T,TauW, Turbulent velocity, heat flux.	Pitot tube, preston tube, HWA, Sheathed thermocouple.
Guellouz and Tavoularis (1992)	Rect. channel	1.025–1.35	108000	V, TauW, Reynolds stress, flow visualization	2D PIV, Acetic acid tracer.
Gosset and Tavoularis (2006)	Rect. Ch. With rods	g/d = 0.025–0.7	388–2223	Flow visualization	Dye injection
Baratto et al. (2006)	5 rods	1.149	42000	V, V', PSD, Turbulent stress	HWA, Preston tube.
Silin and Juanico (2006)	Three rods in circular channel	1.20, 1.12	1400, 130000	Temperature	Thermocouple

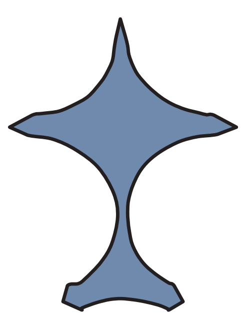

Fig. 5. Cross section of test section (1967).

turbulent mixing is attributed to the large scale periodic flow pulsation between the sub-channels (Leonhard Meyer, 2010). The measurement of frequencies and special correlations of coherent structures are carried out in recent times (Baratto et al., 2006). The averaged characteristics of the transport of momentum and energy in a heated rod bundles (Meyer, 1994) are also carried out to measure the turbulent mixing coefficient and also the cross flow mixing during boiling conditions. In single phase coolant mixing experiments, either temperature (Silin et al., 2004), dye concentration, conductivity or the radioactive tracers (Carelli et al., 1969; Castellani et al., 1975) are used to measure the coolant mixing between sub-channels. The single phase convective heat transfer characteristics of rod bundles of PWR (Holloway et al., 2008; Hochreiter et al., 2010), LMFBR (Chiu et al., 1980; Dwyer et al., 1972; Engle et al., 1980) and GCR (Hudina and Markbczy, 1977; Dalle Donne et al., 1977, 1979) are carried out to measure the temperature distribution (Nijsing et al., 1966). For the cases of two phase flow conditions in rod bundle, investigations on cross flow mixing during boiling, void distribution and two phase model for light water reactor sub-channel analysis were carried out by many researchers (Beus, 1970; Kelly, 1980; Anklam and White, 1981; Wolf et al., 1987). The core coolability (Grandjean, 2007) under LOCA conditions such as reflooding and rewetting experiments (Ihle and Rust, 1987; Duffey and Porthouse, 1973; Thompson, 1974; Muto et al., 1990; Tuzla et al., 1991; Abdul-Razzak et al., 1992; Sahu et al., 2010) were conducted.

The measurement of flow and enthalpy, in case of the cross flow mixing between parallel channels during boiling is carried out by Rowe (1967, 1970). The geometry of the interconnected square and triangular channel considered for study is shown in Fig. 5. The electrically heated test section made of SS-321 is used for these experiments. The amount of natural mixing occurring between two inter connected parallel channels during boiling as well as under the single phase conditions are investigated. The findings from the experimental works are used to validate COBRA code. The results of the calculations (Rowe, 1967) showed that turbulent mixing during non-boiling conditions is nearly independent of rod spacing. In the case of two phase flow, boiling can cause an increase in mixing only for certain conditions. The experiments indicated insignificant mixing in tightly packed rods with

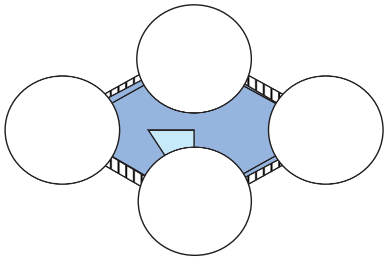

Fig. 6. Cross section of test section used by Eiffler and Nijsing for velocity field measurement (1967).

incremental influence on mixing at very high quality in two phase flow.

Fig. 6 shows the geometry considered (Eifler and Nijsing, 1973) to study the effect of secondary flow in rod bundle geometry (Eifler and Nijsing, 1967). Turbulent diffusion of momentum around the rod periphery is neglected in the model of Deisler and Taylor (Carajilescov, 1975). Their finding indicates that the secondary flow must transport turbulent rich fluid away from regions where turbulence production exceeds dissipation. The experimental works revealed that lateral variation of wall shear stress must be considered the prime cause for existence of secondary flow. Local wall shear stress gradient will induce circular motion tending to transport of high momentum fluid through main flow in the direction of decreasing wall shear stress. Nijsing et al. (1967) conducted the experiment for various p/d ratios (1.05, 1.1, 1.15) and for a range of Reynolds number from 15,000 to 50,000. Pitot tube is used for the measurement of flow velocities in the sub-channel and also observed the fluid mixing between sub-channels in a bundle of finned tube.

The measurement of turbulent intensities and wall shear stress in rod bundle geometry [Fig. 7] was carried out by Kjellstrom (1972) for a p/d ratio of 1.217 using air at a fixed Reynolds number of 2.74×10^5. The measurement of flow velocity and turbulence intensity distribution in axial, tangential and radial directions were performed by using hot wire anemometer, while the Reynolds stress distributions were obtained using Preston tube.

Many experiments on rod bundle hydrodynamics were carried out by Trupp and Azad (1975) using the hot wire anemometer for different p/d ratios varied from 1.20 to 1.50 and Reynolds number in the range of 1.2×10^4 to 8.4×10^4. The cross section of test section used in their study is shown in Fig. 8. The measurements were carried out for axial velocity, turbulent intensity, shear stress and power spectra of axial turbulence. Wall shear stress distribution was presented and discussed in detail.

Experimental and analytical study of axial turbulent flow in an interior sub-channel of a bare rod bundle is investigated by many other researchers. Experimental measurement of the distribution of the axial velocity, turbulent kinetic energy and Reynolds stress were performed using Laser Doppler anemometer (LDA) by Carajilescov (1975) for p/d ratio of 1.123 and L/d_h of 77 for the geometry depicted in Fig. 9. The comparison of HYBBAC code predictions with experimental data at the

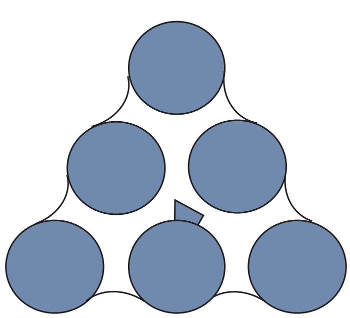

Fig. 7. Cross section of test section used by Kjellstrom, 1972, (P/D = 1.217).

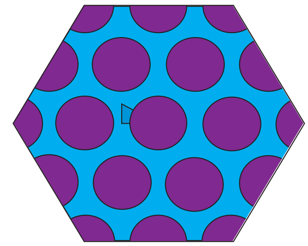

Fig. 8. Cross section of test section used by Trupp and Azad (1975).

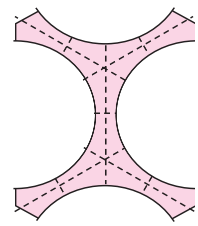

Fig. 9. Cross section of test section used by Carajilescov.

symmetry locations were made in his dissertation, work.

Experimental investigations on the structure of turbulent flow in rod bundles (Rehme, 1978, 1979, 1980a, 1981, 1987a,b) wall sub channels and the fluid flow through an asymmetric rod bundles were studied extensively by (Rehme, 1976, 1980, 1981, 1986, 1987, 1992) and Hooper and Rehme (1983). Measurements of velocity, turbulence structure using hot wire anemometer (Rehme, 1976, 1984a,b) and wall shear stress are carried out in a rod arrays for different p/d (1.036, 1.07, 1.148) and wide range of w/d (1.045, 1.048, 1.072, 1.074, 1.096, 1.118, 1.222, 1.252) ratios (Rehme, 1977, 1980b,c, 1982a,b,c, 1983a,b, 1984, 1985, 1986). A rectangular channel with four parallel rods is used for the experiments as shown in Fig. 10. The experimental findings reveal that structure of turbulent flow through rod bundles is different from that of circular tubes, at least for p/d and w/d ratios less than 1.2. They concluded that energetic and almost periodic azimuthal turbulent velocity component directed through the gap is responsible for the increased mixing process. The periodic pulsation is apparently generated by incompressible-flow parallel-channel instability. The long length scales of the axial and azimuthal turbulent-velocity components, relative to the gap width, emphasize the anisotropy of the turbulent transport processes in the gaps of rod bundles. The anisotropy of turbulent transport not only depends on the p/d and w/d ratios but also on its geometrical location inside the flow channels of rod bundles.

Experimental investigation of mass exchange under two phase flows between two laterally interconnected sub-channel is carried out by Tapucu et al. (1994). The exchange of air/water mixtures flowing in two parallel, square, communicating channels is used for the initial testing and validation of ASSERT code. Channel geometries used in experimental setup are given in Fig. 11. The experiments were conducted at the same initial nominal mass flux in both sub-channels, but for different initial voids and different orientations. The key

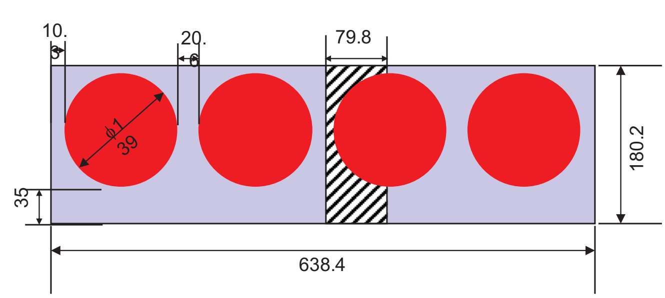

Fig. 10. Cross Section of the test channel used by Rehme.

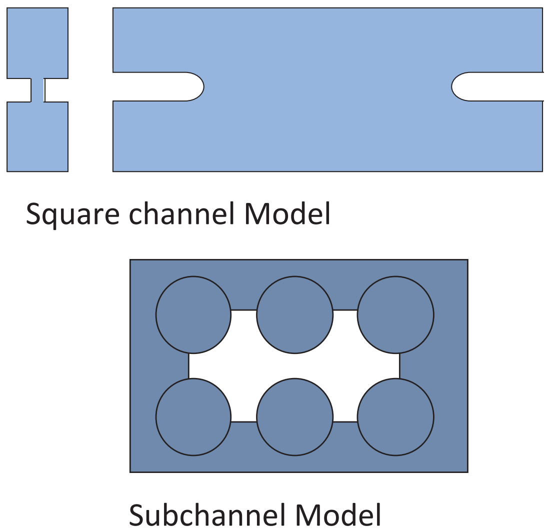

Fig. 11. Cross section of test Section used by Tapucu.

parameters, such as, pressure, void fractions, and liquid and gas flow rates in both channels, were measured at several axial locations and the developed code was validated. The code is capable of computing distribution of flow and phases in horizontal channels and fuel bundles, and the onset of subcooled boiling in bundles. However further, investigations are required to improve the constitutive relationships describing void generation, condensation and drift.

Nicolas Silin and Luis Juanico (2006) carried out an experimental study for dependency of Reynolds number with turbulent mixing between fuel-bundle sub-channels is studied. The physical configuration used in experimental setup is depicted in Fig. 12. The measurements were done on a triangular array bundle with a p/d = 1.20 and 10 mm rod diameter, in a low-pressure water loop, for a range of Reynolds numbers between 1.4×10^3 to 1.3×10^5. The high accuracy of the results is obtained by a recently developed thermal tracing technique. It is also observed that there is a marked increase in the mixing rate for lower Reynolds numbers. The weak theoretical base of accepted Reynolds number dependence is pointed out in light of the later findings and its ambiguous supporting experimental data. The recent results also provide indirect information about dominant large scale flow pulsations at different flow regimes.

Experimental works of turbulent diffusion of heat transfer between connected flow passages in the geometries shown in Fig. 13 are carried out by Seale (1979). The measurement of turbulent quantities in rod bundles is performed to assess the effect of turbulent inter channel mixing due to structure of turbulence, effect of secondary flow (0.5% of axial flow velocity) and effective diffusivity in circumferential direction. This study is concerned with mixing due to turbulent diffusion and

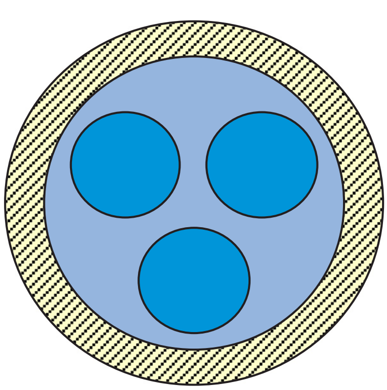

Fig. 12. Test section used in mixing experiments.

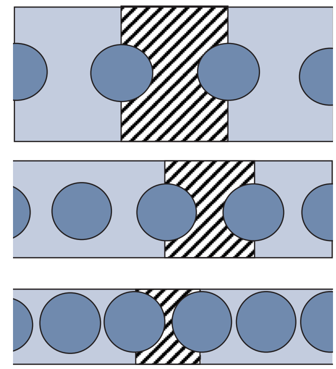

Fig. 13. Cross section of Seale's facility (Seale, 1979).

secondary flow between sub-channels formed by bare smooth rods.

The experimental results confirm that inter sub-channel mixing is considerably higher than that predicted by a simple turbulent diffusion theory. Further, effective diffusivity through the gap is also observed to be strongly anisotropic in nature.

The contours of axial velocity and temperature do not suggest any disturbances which could be attributed to secondary flow. The enhanced mixing rate through the gap is diffusive in nature and is due to a marked anisotropy in the gap region. Secondary flow appears to be much less significant in the ducts of the current type compared to that believed earlier. The source of marked anisotropic diffusivity remains a major unanswered question.

Fully developed turbulent flow of air in five rod sector of 37 rod bundle CANDU type fuel assembly (Fig. 14.) is conducted to measure the mean velocity, Reynolds stress and turbulent scales at the wall and inner sub-channel (Ouma and Tavoularis, 1991). The study was conducted by moving the central rod inside wise as well in rotated conditions, thereby, creating the rod-to-rod and rod-to-wall contacts. Aim of the study was oriented towards understanding of heat transfer characteristics under distorted geometrical conditions. The measurements show that local friction factor varied appreciably than the average friction factor based on sub-channel bulk velocity.

Measurements of enthalpy and flow distributions at the outlet of square-lattice bundle (Fig. 15.) at 69 bar (BWR Operating conditions) are carried out in the General Electric (GE) 9-rod bundle (Lahey et al., 1970). Similarly, two phase flow distribution tests were conducted in ISPRA-16 rod bundle under PWR operating conditions. These tests were used to assess the inter channel mixing models used in the COBRA-IV codes (2000).

Experimental and theoretical investigations of critical heat flux and turbulent mixing in hexagonal tight lattice 7-rod bundles (Fig. 16) were investigated by Cheng and Muller (1998) using Freon-12 as a working fluid. It has been found that the two-phase mixing coefficient increases with decreasing mass flux for the test conditions considered. Similar experiments in two-phase flow regime were conducted in a 7 rod bundle with circular channel by many other researchers to study the effect of channel wall flow distribution (Carver et al., 1984).

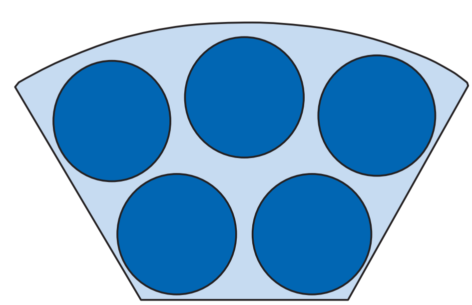

Fig. 14. Cross section of Five rod sector of 37 rod bundle test section.

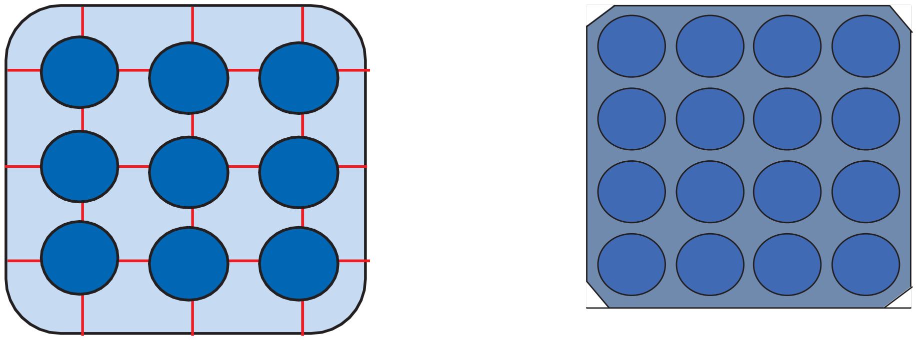

Fig. 15. GE-9 rod and ISPRA-16 rod two phase flow distribution test bundles.

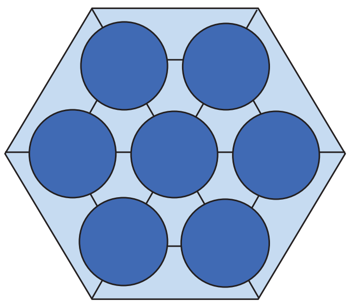

Fig. 16. Test section used in CHF and mixing experiment.

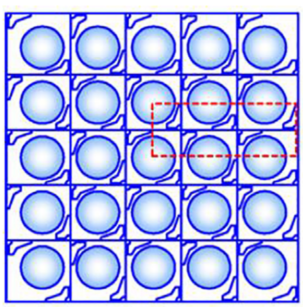

Fig. 17. Rod bundle with mixing vanes.

Phenomenological investigations on the turbulent flow structures in a 5 × 5 rod bundle array (Fig. 17) with mixing devices is carried out by Chang et al. (2008), in MATIS-H facility. This experiment was conducted in the cold test loop at Korea Atomic energy research institute (KAERI) using Laser Doppler Anemometry (LDA) to measure the velocities and the turbulent quantities. The measurement and CFD predictions indicated that split type spacer grid is more effective for mixing between neighboring sub-channels rather than mixing within a sub-channel. Swirl type mixing vane has better performance for mixing within a sub-channel.

3.2. Theoretical studies of fuel assembly sub-channels

The analytical works related to the turbulent inter channel mixing, sub-channel analysis, fluid flow and heat transfer characteristics of rod bundles were surveyed from the literature and some of the literature relevant to inter channel mixing and sub-channel analysis is discussed in this section.

The analytical findings on rod bundles were aimed at predicting rod bundle's hydraulic and heat transfer characteristics under single and two phase flow conditions in the fuel bundles of different types of reactors. The sub-channel analysis is aimed at predicting flow, velocity and temperature distributions in bare as well as wire wrapped rod bundle (Bishop and Todreas, 1979; Rehme, 1982; Hooper and Rehme, 1984; Hooper and Wood, 1984; Rapley and Gosman, 1986; Kaiser and Zeggel, 1987; Lee and Jang, 1997; Kriventsev and Ninokata, 1999; Misawa et al., 2003; Chang and Tavoularis, 2007; Kazuyuki and Ninokata, 2007; Ninokata et al., 2009) and also for blocked sub-channels (Ang et al., 1987; Hooper and Wood, 1983). The analysis included the prediction of pressure drop across fuel assemblies and the friction factors (Rehme, 1972, 1973a,b; Marek et al., 1973; Marek and Rehme, 1979; Lee, 1995; Jian et al., 2002; Su and Freire, 2002; Schikorr et al., 2010) of different types of sub-channels (Cheng and Todreas, 1986) such as central, wall and corner in rod bundles. The evaluation of flow distribution, coolant mixing within the bundle due to inter-channel mixing under single phase (Castellana et al., 1974; Jeong et al., 2006) and two phase flow conditions (Lahey and Schruab, 1969; Carlucci et al., 2004; Kawahara et al., 2006) with different geometrical parameters (Jeong et al., 2007) of bare rod assembles (Hee, 1970), assemblies with grid spacers, wire wrapped spacers (Ginsberg, 1971) and mixing vanes (Lee and Choi, 2007) were carried out using different mixing models in sub-channels analysis codes (Tapucu et al., 1994; Dae-Hyun et al., 2000). Different mixing models and correlations were studied to estimate the turbulent mixing rate and turbulent energy exchange between sub-channels for water (Petrunik, 1973; Ramm et al., 1974; Ninokata and Todreas, 1975; Rehme, 1979; Rehme, 1980; Eiff and Lightstone, 1996; Xiaohua Wu, 1995) and low Prandtl number fluids (Rehme, 1992; Kim and Chung, 1999, 2001). The influence of wall sub-channels on turbulent flow and temperature distributions are also studied analytically (Rehme, 1978a,b; Mohanty and Sahoo, 1986). The coolant mixing within the reactor pressure vessel and outlet plenum were examined to assess the flow through the fuel bundle (Todreas, 1979, 1985) and also the influence of wall and upper tie plate on flooding in rod bundle (Okhawa and Lahey, 1980; Spatz and Mewes, 1987). Apart from flow distributions, turbulent heat transfer characteristics is also studied numerically using various turbulent models in bare rod and wire wrapped bundles (Axford, 1967; Baglietto and Ninokata, 2003; Khan et al., 1975; Lewis and Buettiker, 1974; Matiur and Ahmad, 1982; Tae Sun and Todreas, 1988; Thompson and Holy, 1975).

The analytical works on the temperature and heat flux distribution in the tightly packed liquid metal cooled pin bundle are carried out by many researchers (Dwyer, 1966; Nijsing et al., 1964, 1966, 1975; Subbotin et al., 1961; Subbotin et al., 1971; Levchenko et al., 1967; Hofmann, 1970; Zaiyong Ma et al., 2012).

Analysis of liquid metal heat transfer in closely spaced fuel rods assemblies is carried out by Nijsing and Eifler (1969). A theoretical evaluation of two-dimensional temperature and heat flux distribution for turbulent axial flow at relatively low Peclet numbers is reported. In the analysis, turbulent, fully developed axial flow and heat transfer conditions are considered for heat transfer in infinite triangular or rectangular array of fuel rods cooled by liquid metal. It was assumed that, the coolant turbulent heat transport is negligible with respect to heat conduction due to liquid metal and close rod spacing. The results were presented for dimensionless rod spacing in the range p/

d = 1–1.15 and various rod material combinations of practical interest. Theoretical predictions were compared with experimental data and found to be in good agreement.

A review of heat transfers to liquid metals flowing in-line through un-baffled rod bundles is presented by Dwyer (1969) for fully developed heat transfer to liquid metals flowing turbulently and longitudinally through closely spaced un-baffled rod bundles (Dwyer et al., 1972). The pins were arranged in triangular pitch, and rod spacings, rod design, and ranges of independent variables covered were chosen with reference to liquid-metal-cooled nuclear reactor applications. Three different sets of thermal boundary conditions were considered i.e. (i) uniform heat flux in the axial direction with uniform temperature in the circumferential direction, on the outer surface of the cladding, (ii) uniform heat flux in both directions on the outer surface of the cladding, and (iii) uniform heat flux in both directions on the inner surface of the cladding. The rod-average heat-transfer coefficients and circumferential variations of temperature and heat flux on the rod surface were presented as functions of p/d and Pe. Different methods of estimating eddy diffusivity of momentum in the circumferential direction were employed, and the reference method i.e. equality of the eddy diffusivities of momentum in both the circumferential and radial directions appears to give satisfactory predictions.

A numerical analysis of heat transfer in turbulent longitudinal flow through assemblies of un-baffled fuel rods is presented by Pfann (1972). The solution applied to triangular or rectangular arrays of fuel rods with fully developed velocity and temperature profiles, for fluids $Pr \leq 1$. In the case of liquid metals, the thermal resistance of the cladding and bond were considered. However, the turbulent heat transport component is neglected. For common liquids the circumferential turbulent heat transfer is considered. Results are compared in the range of dimensionless rod spacing of 1.0–1.6. The experimental results of other authors were found to be in good agreement with these calculations.

An analytical method of evaluating the circumferential variations of temperature and heat flux fields inside and around a displaced fuel rod in triangular rod bundles in turbulent flow were presented with examples by Hishida (1974). The analysis mainly consists of the derivation of simultaneous solutions of a set of heat conduction equations for fuel, cladding and coolant for fully developed flow and heat transfer conditions. The local coolant velocity distribution, which is necessary for deriving the temperature field in coolant, is determined by solving the Navier-Stokes equation and the turbulent mixing of coolant is taken into consideration. The results show how the circumferential variations in the temperature and heat flux fields on the outer surface of the cladding increases with lowering the p/d ratio and the larger the fuel rod displacement due to thermal conduction and peripheral coolant flow velocity distribution.

The calculation of heat transfer developments in fuel bundle is carried out by Vonka and Boonstra (1974). Two elementary heat transports in rod bundle geometry were investigated, namely, the transport from a fuel rod surface into the adjacent sub-channel and from one sub-channel into the next one. The first transport is characterized in terms of the Nusselt or Stanton numbers, while the later in terms of the Stanton gap number or mixing factor. Hydraulically developed flow is assumed, with no feedback of heat transport on the flow condition. A three-dimensional numerical calculation by means of the finite difference method is applied. The comparison of available experimental data and with theoretical predictions, it is found that the heat transfer between sub-channels is developed considerably slower in comparison with the development of the heat transport from fuel rod to sub-channel.

Solving the conservation equations in fuel rod bundles exposed to parallel flow by means of curvilinear-orthogonal coordinates is carried out by Meyder (1975). A unique method is described for constructing a curvilinear-orthogonal mesh grid. A method to perform calculations in such mesh grid is illustrated and demonstrated for the problem of laminar flow in rod bundle geometries. Further, for prediction of turbulent velocity and temperature distribution in the central sub-channel of rod bundles (Meyder, 1975), the method was presented to calculate local heat transfer coefficients, temperatures, mixing rates, friction factors and wall shear stress distribution. This model led to satisfactory agreement between experimental and theoretical results. The methods outlined here for the central sub-channel in a fuel element could now be applied in an analogous way to corner and wall channels. By coupling the different channel types, it would be possible to give a complete and economic description of the geometry in a fuel element or in a test section.

Some recent computations of rod bundle thermal hydraulics using boundary-fitted coordinates were presented by Chen et al. (1980). In this paper, the status of development of two computer codes for benchmark rod bundle thermal hydraulic analysis using the boundary-fitted coordinate method are described. A classification of the flow in rod bundles is made and details of computer codes are given. The codes solved essentially, identical sets of conservation of mass, momentum and energy. However, one of them treated the fully-elliptic set of equations, and the other neglects the axial diffusion of momentum and heat. The ignorance of stream wise diffusion made the flow partially-elliptic, and brought significant reductions in computer storage and modest savings in required computer time. Results of some validation calculations were presented with applications to possible reactor geometries.

Three dimensional distributed resistance modeling of wire-wrapped rod bundles were developed by Ninokata et al. (1987). The model is developed mainly for replacing the forced crossflow models that are currently in wide use in sub-channel analysis codes. The models specifically account for the presence of the wire-wrap spacer and might be used for any lumped parameter thermo-hydraulic analysis. Validation studies of the hydrodynamic resistance models were also performed using a sub-channel code ASFRE (Yoshika et al., 1981; Ninokata et al., 1985). The models were tested against sub-channel velocity and temperature data taken from bundles of triangular rod array configurations with wire-spacers. Overall the models performed satisfactorily, predicting the most important qualitative trends for flows in wire-wrapped rod bundles.

Turbulent flow and heat transfer in rod bundle sub-channels is analyzed using two-dimensional eddy diffusivity and a higher order K-l model proposed by Mohanty and Sahoo (1988). The eddy diffusivity was a generalization of the Reichardt's model for a concentric annulus. The K-I model is used to examine the effects of anisotropic turbulent diffusion. Fully developed flow and thermal conditions were assumed with the rod surface subjected to uniform heat flux in triangular and square arrangements of rods were investigated. The pitch to diameter ratio (p/d) is varied from 1.1 to 1.5. The algebraic eddy diffusivity was found to reproduce literature information, including those attributed to anisotropy or secondary flow effects. Thermal calculations have been carried out both for constant and variable values of turbulent Prandtl number.

3.3. CFD analysis of rod bundles

In recent times, as discussed earlier due to advancement of computational resources and CFD techniques, the efforts are taken up by researchers to reduce the experimental works and apply CFD for the simulation of the flow characteristics in a rod bundles. The literature on application of CFD technique for the study of rod bundle flow distributions and heat transfer characteristics were surveyed. The application of different CFD techniques and turbulence models were discussed in this section.

The CFD simulations were carried out by various researches, in PWR fuel assemblies, for the single phase turbulent flow simulations to estimate the velocity and temperature fields (Ninokata and Merzari, 2007), secondary flow in rod bundles (Baglietto and Ninokata, 2004), turbulent flow structures and mixing across the narrow gap between rod bundle sub-channels (Biemüller et al., 1996; Chang and Tavoularis, 2008). In these simulations, various models of CFD techniques were studied. These include RANS (Chandra et al., 2009) or URANS (Merzari et al., 2007, 2008) with different turbulent models k-ε (Baglietto and Ninokata, 2005; Lee and Jang, 1997), LES (Biemüller et al., 1996) and DNS (Ninokata et al., 2004; Baglietto et al., 2006). The applications of CFD for the prediction of two phase flow velocity, sub-cooled boiling (Krepper et al., 2007) and for void distribution. Further these analyses were extended in the rod bundle (Carver et al., 1984; Anglart and Nylund, 1996). The CFD modeling was attempted for all types of reactors such as PWR (Conner et al., 2010; Cinosi and Walker, 2016), LMFBR (Cheng and Tak, 2006; Gajapathy et al., 2009; Chandra et al., 2009) and SCWR (Gu et al., 2008; Zhi, 2009; Zhi and Simon, 2010). The applications using computational fluid dynamics for the simulation rod bundles of next generation nuclear systems are being developed world wide (Angel et al., 2015).

On the discussions of turbulence models for rod bundle flow computations (Gabor Hazi, 2005) using commercial computational fluid dynamics codes, there are more than one turbulence model built in. It is the user responsibility to choose one of those models, suitable for the problem being studied. A common feature in a number of those simulations is that they were performed using the standard k–ε turbulence model without justifying the choice of the model. The authors have highlighted that, the flow in a fuel rod bundles as a case study and discuss why the application of the standard k–ε model fails to give reasonable results in this situation. It was shown that the k–ε model could not reproduce the basic characteristics of such flows and demonstrated that the Reynolds-stress model predicts, accurate modeling of rod bundles flows.

A numerical study using CFD is carried out to estimate pressure loss in strap and mixing vane structures and fluid hot spot locations in 5 × 5 fuel rod bundle (Ikeda et al., 2006). A CFD simulation under single-phase flow condition is conducted for one specific condition in a water departure from nucleate boiling (DNB) test to examine the applicability of the CFD model for predicting the CHF rod position. Energy flux around the rod surface in a water DNB test is the sum of the intrinsic energy flux from a rod and the extrinsic energy flux from other rods, and increments of the enthalpy and decrements of flow velocity near the rod surface were assumed to affect CHF performance. CFD makes it possible to model the complicated flow field consisting of a spacer grid and a rod bundle and evaluate the local velocity and enthalpy distribution around the rod surface, which are assumed to determine the initial conditions for the two-phase structure. The results of this study indicated that single-phase CFD can play a significant role in designing PWR spacer grids for improved CHF performance.

CFD analysis for thermal–hydraulic behavior of heavy liquid metal flows, especially lead–bismuth eutectic, in sub-channels of both triangular and square lattices is carried out by Cheng and Tak (2006). Sensitivity of different turbulence models and the effect of pitch-to-diameter ratio, on the thermal–hydraulic behavior is investigated. Among the turbulence models selected, only the second order closure turbulence models reproduce the secondary flow. For the entire parameter range studied in this paper, the amplitude of the secondary flow is reported less than 1% of the mean flow. A strong anisotropic behavior of turbulence is observed. The turbulence behavior is similar in both triangular and square lattices. The average amplitude of the turbulent velocity fluctuation across the gap is about half of the shear velocity. It is only weakly dependent on Reynolds number and p/d ratio. A strong circumferential non-uniformity of heat transfer is observed in tight rod bundles, especially in square lattices. Related to the overall average Nusselt number, CFD codes found to reproduce results for both triangular and square rod bundles. Comparison of the CFD results with bundle test data in mercury indicated that the turbulent Prandtl number for heavy liquid metal (HLM) flows in rod bundles is close to 1.0 at high Peclet number conditions, and increases by decreasing Peclet number. Based on the present results, the SSG Reynolds stress model with semi-fine mesh structures ($y_1^+ < 15$) is recommended for the application of HLM flows in rod bundle geometries.

CHF experiments and CFD analysis in a 2 × 3 rod bundle with mixing vane is performed by (Shin and Chang, 2005, 2009) and In et al., 2008. The CHF enhancement using various mixing vanes is evaluated and the flow characteristics are investigated through the CHF experiments and CFD analysis. The detailed flow characteristics were also investigated by CFD analysis using the same conditions as the CHF tests. To calculate the subcooled boiling flow, the wall partitioning model is applied to the wall boundary and various two-phase parameters were also considered. The reliability of the CFD analysis in the boiling is confirmed by comparing the average void fractions of the analysis and the experiments, the results agreed well. From the CFD analysis, the void fraction flattening as a result of the lateral velocity induced by the mixing vane is observed. By the lateral motion of the liquid, the void fraction in the near wall is decreased and that of the core region is increased resulting in the void fraction flattening. The decrease of the void fraction in the near wall region promoted liquid supply to the wall and consequently the CHF increased. For the quantification of the void flatness, an index is developed and the applicability of the index in the CHF assessment is confirmed.

CFD analysis of flow field in a triangular rod bundle is investigated in sub-channels of VVER-440 pressurized water reactor fuel assemblies for a triangular lattice of p/d 1.35 by Toth and Aszodi (2010). The effect of mesh resolution and turbulence model is studied to obtain guidelines for CFD calculations of VVER-440 rod bundles. Results were compared to measurement data published by Trupp and Azad (1975). The study pointed out that, RANS method with BSL Reynolds stress model using a sufficient fine grid can provide an accurate prediction for the turbulence quantities for such types of lattice. Applying the experiences of the sensitivity study thermal hydraulic processes were investigated in VVER-440 rod bundle sections. Based on the examinations they reported that the spacer grids have significant effects on the cross flows, axial velocity and outlet temperature distribution of sub-channels. Therefore, they have to be modeled satisfactorily in CFD calculations. Some deviations were reported between them. The differences probably were caused by the deficiencies of the COBRA model (inaccurate flow resistance and turbulent mixing coefficients, etc.) and CFD model gave a more accurate prediction. However, it must be mentioned that the suitability of the applied mesh and turbulence models were uncertain therefore, further investigations were suggested.

CFD modeling methodology and validation for steady state single-phase flow in PWR fuel assemblies for normal operation was investigated by Conner et al. (2010). This work was part of a program that is developing a CFD modeling methodology and predicting single-phase and two-phase flow conditions downstream of structural grids.

The three-dimensional CFD simulations of turbulent flow in a wire-wrapped 19-pin rod bundle of an LMFBR for the pressure drop and heat transfer characteristics have been investigated by Natesan et al. (2010). The predicted results of eddy viscosity based turbulence models (k–ε & k–ω) and the Reynolds stress model were compared with those of experimental correlations for friction factor and Nusselt number for different range of Re and for the ratio of helical pitch of wire wrap to the rod diameter. All the three turbulence models considered yield similar results. The friction factor increases with reduction in the wire-wrap pitch while the heat transfer coefficient remains almost unaltered.

The Numerical simulation of the thermal hydraulic characteristics within the fuel rod bundle using CFD methodology is carried out by Liu and Ferng (2010). These localized information, including flow, turbulence, and heat transfer characteristics, etc., assist in the design and the improvement of rod bundles for nuclear power plants which is difficult to measure in actual geometry. In this paper, a three-dimensional CFD analysis with the Reynolds stresses turbulence model is proposed to simulate these characteristics within the rod bundle and subsequently to investigate the effects of different types of grids on the turbulent mixing. Heat transfer enhancement is studied in detail. Based on the CFD simulations, the secondary flow could be reasonably captured in the rod bundle with the grid. The CFD simulation results clearly showed that split-vane pair grid would enhance both the flow mixing and the heat transfer capability more than the standard grid. In addition to the comparison of the results of experiment and correlation, the present predicted result for the Nusselt (Nu) number distribution downstream the grid shows reasonable agreement for the standard grid design. However, there is discrepancy in the decay trend of Nu number between the prediction and measurement for the split-vane pair gird. This would be significantly improved by adopting the finer mesh ($y^+ < 1$) simulation and Low-Reynolds number turbulence model.

Large Eddy Simulation (LES) and unsteady Reynolds averaged Navier-Stokes (URANS) simulations of turbulent flow in 4 rod bundles in ocean environment to simulate the rolling motion is investigated by Yan et al. (2009, 2011). The effect of rolling motion consists of two parts, the axial additional force which causes velocity oscillation and the radial additional force. The analysis indicate that in the ocean environment, effect of rolling motion on the flowing similarity is considerable and cannot be neglected for proper prediction of flow and heat transfer.

Numerical simulation of the coherent structures and turbulent mixing in tight Lattice is investigated by Yan et al. (2012). The calculation results were first validated with experimental data and then the effects of Reynolds number, Prandtl number and pitch to diameter ratio on the coherent structure and turbulent mixing were analyzed. They reported that the effect of Prandtl number on the coherent structure and turbulent mixing is more limited than that of Reynolds number. The effect of pitch to diameter ratio on the coherent structure is most significant. The critical pitch to diameter ratio for this lattice is 1.03. As the pitch to diameter ratio decrease from a large number to the critical value, the coherent structure becomes more significant which is beneficial to the local flow and heat transfer. But as the pitch to diameter ratio continues to decrease from this critical value to a very small value, the coherent structure becomes weaker and weaker, and the local flow and heat transfer also become more and more terrible.

Performing the full bundle CFD analysis is computationally intensive and more time consuming, hence, simulating the fuel assemblies with low resolution CFD approaches is investigated by Roelofs et al. (2012). In addition to the traditional fuel assembly simulations using system codes, sub-channel codes or porous medium approaches, as well as detailed CFD simulations to analyze single sub-channels, a Low Resolution Geometry Resolving (LRGR) CFD approach and a Coarse-Grid-CFD (CGCFD) approach are considered in their work. Both methods were based on a low resolution mesh that allows the capture of large and medium scale flow features such as recirculation zones, which are difficult to be reproduced by the system codes, sub-channel codes and porous media approaches.

4. Summary and future directions of research

Thermal hydraulic investigations on the reactor core for different geometries of the fuel assemblies are being carried out since the beginning of the nuclear power program worldwide. Researchers have used different sub-channel analysis codes, experimental techniques, analytical and computational tools for the estimation of key parameters on rod bundle mixing to determine reactor safety margins. In this paper, the works carried out by different groups of researchers in rod bundle sub-channel analysis is reviewed extensively. More than 250 research papers on this topic were referred. The major findings and the limitations were critically reviewed. Based on the understanding from the vast literature available in open sources, it was found that the most of the studies focused on the effect of key mixing parameters such as beta, gaps, p/d ratio and other parameters characterizing the single phase mixing between sub-channels. These studies emphasize the importance of predicting the accurate local conditions such as flow rate, velocities and temperature distribution in the rod bundles. Further, these studies are oriented towards establishing reliable thermal hydraulic safety parameters and thermal margins on CHF and CPR as discussed above.

Based on the literature survey of the subject, it is also noted that various experimental tests on turbulent mixing under different fluid conditions, sub-channel and CFD analysis can be carried out to simulate those experiments. Thus reliable predictions of the mixing by CFD can be established. Further, design exploration studies of various parameters affecting the inter-channel mixing using the CFD techniques and to extend the mixing models for wide range of operating and geometrical conditions can be approached. From the investigations on turbulent inter-channel mixing, improved understanding on estimation of mixing parameters to characterize the key thermal hydraulic safety margins at different states of the operational and transient conditions of the core can be evaluated. All these sub-channel codes require large amount of data processing of sub-channel geometry, rod to sub-channel connectivity details, fluid properties, power distribution and operating conditions. Therefore, the research shall be directed in the following gap areas of sub-channel analysis for reliable estimation of the safety margins of nuclear reactor core.

a. Improved thermal hydraulic models of heat transfer and mixing correlations,
b. Reliable sub-channel analysis tools,
c. limitations for the calculation speed,
d. Improved code to handle large number of sub-channels and fuel rods,
e. Code to include the verification and validation of earlier research works,
f. Non-linear variation of the properties of coolant, fuel and clad materials.

The development of software framework to automate the entire core thermal hydraulic analysis procedure and the presentation of thermal hydraulic results and hence the safe, reliable and accurate predictions of the thermal hydraulic safety margins is suggested. This will certainly improve the speed, accuracy and the repeatability of the sub-channel analysis and hence, the reliable estimation of the safety margins of nuclear reactor core.

ReferencesA. Sub channel thermal hydraulic analysis and codes

Abdul-Razzak, A., Shoukri, M., Chan, A.M.C., 1992. Rewetting of hot horizontal tubes. Nucl. Eng. Des. 138, 375–388.

AREVA NP Inc, ANP-10311P, Revision 0, 'COBRA-FLX: A Core Thermal-Hydraulic Analysis Code Topical Report', March 31, 2010, ADAMS Accession No. ML101550172.

Basehore, K.L., Todreas, N.E., 1980, SUPERENERGY-2: A Multi assembly, Steady-State Computer Code For LMFBR Core Thermal-Hydraulic Analysis, PNL-3379, C00-2245-57TR, August 1980.

Basile D., Chierici R., Beghi M., Salina E., Brega E., COBRA-EN, an Updated Version of the COBRA-3C/MIT Code for Thermal-Hydraulic Transient Analysis of Light Water Reactor Fuel Assemblies and Cores, Report 1010/1 (revised 1.9.99), ENEL-CRTN Compartimento di Milano.

Baumann, W., MISTRAL-II, Thermohydraulische Mischströmungsalgorithmen für Stabbündel. KfK 1605, Juni 1972 (in German).

BAW-10156-A, Rev. 1, LYNXT Core Transient Thermal-Hydraulic Program, B&W Fuel Company, August 1993.

Bowring, R.W., Hambo, a computer programme for the subchannel analysis of the hydraulic and burnout characteristics of rodclusters, part 1: general description, AEEW-R524, 1967, and part 2: the equations, AEEW-R582, 1968.

Burns, C.J., Aumiller, D.L., 2007. COBRA-IE evaluation by simulation of the NUPEC BWR full-size fine-mesh bundle tests (BFBT), B-T-3653.

CENPD-206-NP, Combustion Engineering Inc., TORC Code, Verification and Simplification Methods, CENPD-206-NP, 1977.

Chu, P.T., Chelemer, H., Hochreiter, L.E., THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores, WCAP-7956, 1973.

Chang Soon Heung, Won-Pil Baek, 2003. Understanding, predicting, and enhancing critical heat flux, NURETH-10.

Chelemer, H., Weisman, J., Tong, L.S., 1972. Sub-channel thermal analysis of rod bundle cores. Nucl. Eng. Des. 21, 35–45.

Chelemer, H., Hochreiter, L.E., Boman, L.H., Chu, P.T., 1977. An improved thermal-hydraulic analysis method for rod bundle cores. Nucl. Eng. Des. 41, 219–229.

Chen, Todreas, N.E., 1975. Prediction of the coolant temperature field in a breeder reactor including inter assembly heat transfer. Nucl. Eng. Des. 35 (3), 423–440.
Cheng, X., Muller, U., 1998. Critical Heat flux and turbulent mixing in hexagonal tight rod bundles. Int. J. Multiphase Flow 24 (1998), 1245–1263.
Cheng, X., Muller, U., 2003, Review on Critical Heat Flux in water cooled reactors. FZKA-6825, Forschngszentrum Karlsruhe GmbH, Karlsruhe.
Chun, S.-Y., Chung, H.-J., Moon, S.-K., Yang, S.-K., Chung, M.-K., Schoesse, T., Aritomi, M., 2001. Effect of pressure on critical heat flux in uniformly heated vertical annulus under low flow conditions. Nucl. Eng. Des. 203, 159–174.
De Crecy, F., 1994. The effect of grid assembly mixing vanes on critical heat flux values and azimuthal location in fuel assemblies. Nucl. Eng. Des. 149, 233–241.
Dobson, G.P., O'Neill, J.M., 1992. SABRE, User Guide for Version 4, RSSD 261, AEA Technology.
DOE/ET-0009, 1977. A Compendium of Computer Codes for the Safety Analysis of Fast Breeder Reactors.
Duffey, R.B., Porthouse, D.T.C., 1973. The physics of rewetting in water reactor emergency core cooling. Nucl. Eng. Des. 25, 379–394.
Dwyer, O.E., 1969. Heat transfer to liquid metals flowing in line through unbaffled rod bundles: a review. Nucl. Eng. Des. 10 (1969), 3–20.
Eifler, W., Nijsing, R., 1973. VELASCO – Velocity field in asymmetric rod configurations, Ispra Report, EUR 4950e.
Eriksson, S.O., 1971, SCEPTIC. A Fortran IV Computer Program for Temperature Analysis of Gas or Liquid Cooled Flow Passages with Heated Smooth or Roughened Surfaces. A user's Guide. EIR, TM-IN-H79, 16.7.1971.
Faya, A., Wolft, L., Todreas, N., 1979, CANAL USER'S MANUAL, Energy Laboratory Report No. MIT-EL 79-028, November 1979.
Forti, G., Gonzalez-Santalo, J.M., 1973. A model for subchannel analysis of BWR rod bundles in steady-state and transient, Int. Conf. Reactor Heat Transfer, Karlsruhe, Germany.
George, T.L., Basehore, K.L., Wheeler, C.L., Masterson, R.E., 1980 COBRA-WC: A Version of COBRA for Single-Phase Multi-assembly Thermal Hydraulic Transient Analysis, PNL-3259, Pacific Northwest Laboratory, Richland, WA.
Ginox, J.J., 1978. Two-Phase Flows and Heat Transfer With Application to Nuclear Reactor Design Problems, Hemisphere publishing corporation.
Grandjean, C., 2007. Coolability of blocked regions in a rod bundle after ballooning under LOCA conditions. Main findings from a review of past experimental programmes. Nucl. Eng. Des. 237, 1872–1886.
Groeneveld, D.C., Leung, L.K.H., Kirillov, P.L., Bobkov, V.P., Smogalev, I.P., Vinogradov, V.N., Huang, X.C., Royer, E., 1996. The 1995 look-up table for critical heat flux in tubes. Nucl. Eng. Des. 163, 1–23.
Hejzlar, P., Todreas, N.E., 1996. Consideration of critical heat flux margin prediction by subcooled or low quality critical heat flux correlations. Nucl. Eng. Des. 163, 215–223.
Hirao, S., Nakao, N., 1974. DIANA – a fast and high capacity computer code for interchannel coolant mixing in rod arrays. Nucl. Sci. Des. 30, N3.
Huggenberger, M., 1977. SCRIMP A Thermal-Hydraulic Subchannel Analysis Computer Code., Eidg. Institut für Reaktorforschung Würenlingen (Switzerland), EIR-Bericht Nr. 322, Würenlingen, June 1977.
Ihle, P., Rust, K., 1987. PWR reflood experiments using full length bundles of rods with zircaloy claddings and alumina pellets. Nucl. Eng. Des. 99, 223–237.
Imke, U., Sanchez, V., Gomez, R., SUBCHANFLOW: an empirical knowledge based subchannel code. In: Proceedings of the Annual Meeting on Nuclear Technology, pp. 4–6, Berlin, Germany, May 2010.
Kattchee, N., Reynolds, W.C., 1962. HECTIC II. An IBM 7090 FORTRAN Computer Program for Heat Transfer Analysis of Gas or Liquid Cooled Reactor Passages. ID0-28595, AGN, California, December 1962.
Kelly, J.E., Kao, S.P., Kazimi, M.S., 1981. THERMIT-2: A two-fluid model for light water reactor sub-channel transient analysis, MIT-EL-81-014, MIT.
Khabensky, V.B., Malkin, S.D., Shalia, V.V., Ilukhin, Y.N., Nigmatulin, B.I., 1998. Critical heat flux prediction in vertical bottom-closed rod bundles. Nucl. Eng. Des. 182, 203–224.
Chaudri, Khurrum Saleem, Su, Yali, Chen, Ronghua, Tian, Wenxi, Su, Guanghui, Qiu, Suizheng, 2012. Development of sub-channel code SACoS and its application in coupled neutronics/thermal hydraulics system for SCWR. Ann. Nucl. Energy 45, 37–45.
Kim, H.C., Baek, W.-P., Chang, S.H., 2000. Critical heat flux water in vertical round tubes at low pressure and low flow conditions. Nucl. Eng. Des. 199, 49–73.
Kim, W.S., Kim, Y.G., Kim, Y.J., 2002. A subchannel analysis code MATRA-LMR for wire-wrapped LMR subassembly. Ann. Nucl. Energy 29 (2002), 303–321.
Lee, M., 2000. A critical heat flux approach for square rod bundles using the 1995 Groeneveld CHF table and bundle data of heat transfer research facility. Nucl. Eng. Des. 197, 357–374.
Lewis, M.J., Buettiker, P., 1974. Momentum losses and convective heat transfer in rod bundles – an overview. Nucl. Eng. Des. 31, 351.
Macdougall, J.D., Lillington, J.N., 1984. The SABRE code for fuel rod cluster thermo-hydraulics. Nucl. Eng. Des. 82, 171–190.
Muto, S., Anegawa, T., Morooka, S., Yokobori, S., Ebata, S., Yoshimoto, Y., Suzuki, S., 1990. An experimental study on rewetting phenomena in transient conditions of BWRs. Nucl. Eng. Des. 120, 311–321.
Ninokata, H., Deguchi, A., Baba, K., 1985. ASFRE-III: A computer program for triangular rod array thermo hydraulic analysis of fast breeder reactors, PNC Report, PNC-N941 85–106 (July 1985).
Ninokata, H., Okano, T., 1985. SABENA: An Advanced sub-channel Code for Sodium Boiling Analysis. In: Proc. 3rd Int. Topical Meeting on Reactor Thermal Hydraulics, Newport, Rhode Island.
Ninokata, H., Aritomi, M., Anegawa, T., Hara, T., Kamo, H., Kusuno, S., Mishima, K., Morooka, S., Yamamoto, Y., Nishida, K., Sadatomi, M., Sou, A., Yabushita, Y., 2001. Development of the NASCA Code for Predicting Transient BT Phenomena in BWR Rod Bundles. NEA/CSNI/R (2001) 2, vol. 2.
Nordsveen, M., Hoyer, N., Adamsson, C., Anglart, H., 2003. The MONA Subchannel Analysis Code – Part A: Model and Description, Proceedings: Tenth International Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-10), Seoul, Korea.
Wu, Pan, Shan, Jianqiang, Xiang, Xiong, Zhang, Bo, Gou, Junli, Zhang, Bin, 2016. The development and application of a sub-channel code in ocean environment. Ann. Nucl. Energy 95, 12–22.
Pierre St. C.C., 1966. SASS code 1, Subchannel Analysis for the Steady State. AECL-APPE-41 (1966).
Reynolds, W.C., Kattchee, N., 1961. HECTIC, An IBM 704 Computer Program for Heat Transfer Analysis of Gas Cooled Reactors. Aerojet-General Nucleonics, AGN-TM381, May 1961.
Rehme, K., 1987. Convective heat transfer over rod bundles. In: Kakac, S., Shah, R.K., Aung, W., (Eds.), Handbook of Single phase Convective Heat Transfer, Wiley-Interscience, New York, 1987, pp. 7.1–7.62.
Rowe D.S., 1970. Cobra-II: A digital computer program for thermal hydraulic subchannel analysis of rod bundle nuclear fuel elements, BNWL-1229 VC-80.
Rowe, D.S., 1971. COBRA-III: A Digital computer program for steady state and transient thermal hydraulic analysis of rod bundle nuclear fuel elements. BNWL-B-82.
Rowe, D.S., 1973. COBRA IIIC: A Digital Computer Program for Steady-State and Transient Thermal Analysis of Rod bundle Nuclear Fuel Elements, BNWL-1695, Pacific Northwest Laboratory.
Sahu, S.K., Das, P.K., Bhattacharyya, S., 2010. An experimental investigation on the quenching of a thin vertical tube by spray at high flow rate. Nucl. Eng. Des. 240, 1558–1568.
Hirao, Seizo, Nakao, Noboru, 1974. DIANA – a fast and high capacity computer code for interchannel coolant mixing in rod arrays. Nucl. Eng. Des. 30, 214–222.
Seung Hoon Ahn, Gyoo-Dong Jeun, 2001. Effect of Spacer grids on CHF at PWR Operating Conditions, JKNS-33-3-283-297.
Sha, W.T., 1980. An overview of rod-bundle thermal-hydraulic analysis. Nucl. Eng. Des. 62, 1–24.
Stewart, C.W., Wheeler, C.L., Cena, R.J., 1977. COBRA-IV: the model and the method. In: Battelle Pacific Northwest Laboratories Richland, Washington, USA.
Stewart, C.W., Cuta, J.M., Montgomery, S.D., Kelly, J.M., Basehore, K.L., George, T.L., Rowe. D.S., 1993. VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores, Volume 1: Mathematical Modeling, EPRI NP-2511-CCM, Electric Power Institute.
Thompson, T.S., 1974. On the process of rewetting of hot surface by a falling liquid film. Nucl. Eng. Des. 31, 234–245.
Todreas, N.E., Kazimi, M.S., 2001. Nuclear Systems II-Elements of Thermal hydraulic Design. Taylor and Francis.
Toumi, I., Bergeron, A., Gallo, D., Royer, E., Caruge, D., 2000. FLICA-4: a 3D two-phase flow computer code with advanced numerical methods for nuclear applications. Nucl. Eng. Des. 200, 139–155.
Tuzla, K., Unal, C., Chen, J.C., 1991. An experimental study of bottom rewetting in a rod bundle geometry. Nucl. Eng. Des. 125, 189–200.
VIPRE-02: A Two-Fluid Thermal-Hydraulic Code for Reactor Core and Vessel Analysis, Volume 1: Mathematical Modeling and Solution Methods, TR-103931-V1 Research Project, Final Report, June 1994.
Webb, B.J., 1988. COBRA-IV PC: A Personal Computer Version of COBRA-IV-I for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores, PNL-6476.
Weisman, J., Ying, S.H., 1984. A theoretically based critical heat flux prediction for rod bundles at PWR conditions. Nucl. Eng. Des. 85, 239–250.
Wheeler, C.L., Stewart, C.W., Cena, R.J., Rowe, D.S., Sutey, A.M., 1976. COBRA-IV-I: An Interim Version of COBRA for Thermal Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores, BNWL-1962, Pacific Northwest Laboratory.
Wolf, L. Guillebaud, Louis Jean Marie, Faya A., 1978. WOSUB – A Subchannel Code for Steady-State and Transient Thermal Hydraulic Analysis of BWR Fuel Pin Bundles. Volume I – Model Description, MIT-EL-78-023 (1978).
Yadigaroglu, G., Anderani, M., Dreier, J., Coddington, P., 2003. Trends and needs in experimentation and numerical simulation for LWR safety. Nucl. Eng. Des. 221, 205–223.
Yan, B.H., Yu, L., Li, Y.Q., 2009. Research on operational characteristics of passive residual heat removal system under rolling motion. Nucl. Eng. Des. 239, 2302–2310.
Yoo, Y.J., Hwang, Dae Hyun, Sohn, Dong Seong, 1999. Development of a subchannel analysis code MATRA applicable to PWRs and ALWRs. J. Korean Nucl. Soc. 31 (3), 314–327.
Yoshikawa, N., Ishimaru, J., Tamura, S., Ninokata, H., Hirata, N., 1981. ASFRE: A Computer code for single-phase subchannel thermal hydraulic analysis of LMFBR single subassembly, PNC Report, PNC-N941 81–74 (April 1981).

B. Rod bundle experimental works

Aly, A.M.M., Trupp, A.C., Gerrard, A.D., 1978. Measurements and prediction of fully developed turbulent flow in an equilateral triangular duct. J. Fluid Mech. 85, 57–83.
Anklam, T.M., White, M.D., 1981. Experimental investigations of two-phase mixture level swell and axial void fraction distribution under high pressure, low heat flux conditions in rod bundle geometry, conf-810806-8.
Baratto, F., Bailey, S.C.C., Tavoularis, S., 2006. Measurements of frequencies and spatial correlations of coherent structures in rod bundle flows. Nucl. Eng. Des. 236, 1830–1837.
Beus, S.G., 1970. A Two-Phase Turbulent Mixing Model for Flow in Rod Bundles, WAPD-T-2438.
Carajilescov, P., Experimental and analytical study of Axial Turbulent Flows in an interior

subchannel of a bare rod bundle, PhD Thesis, Massachusetts Institute of Technology. MIT, March-1975.
Carajilescov, P., Todreas, N.E., 1976. Turbulent diffusion of heat between connected flow passages. Nucl. Eng. Des. 54, 183–195.
Carelli, M., Curzio, G., Renieri, A., 1969. A radioactive tracer technique for measuring coolant mixing in nuclear reactor fuel sub-assembly. Nucl. Eng. Des. 11, 93–102.
Castellani, F., Curzio, G., Pieve, L., Grazzini, R., 1975. Experimental tests on the applicability of a radioactive tracer technique for measuring coolant mixing in nuclear reactor fuel subassemblies. Nucl. Eng. Des. 32, 105–109.
Chieng, C.C., Lin, C., 1979. Velocity distribution in the peripheral subchannels of the CANDU-type 19 rod bundle. Nucl. Eng. Des. 55, 389–394.
Chiu, R., Todreas, N., Morris, R., 1980. Experimental techniques for LMCFBR assembly thermal/hydraulic tests. Nucl. Eng. Des. 62, 253–270.
Collingam, R.E., Thorne, W.L., McCormack, J.D., 1971. 217-pin wire-wrapped bundle coolant mixing test. TME 71–146.
Donne, Dalle, Marek, J., Martelli, A., Rehme, K., 1977. BR2 bundle mockup heat transfer experiments. Nucl. Eng. Des. 40 (1), 143–156 Special Issue GCFR.
Dalle Donne, M., Martelli, A., Rehme, K., 1979. Thermo fluid dynamic experiments with gas-cooled bundles of rough rods and their evaluation with the computer code SAGAPO. Int. J. Heat Mass Transfer 22 (1979), 1355–1374.
Dwyer, O.E., Berry, H.C., Hlavac, P.J., 1972. Heat transfer to liquid metals flowing turbulently and longitudinally through closely spaced rod bundles (part ii uniform heat flux at the inner surface of the cladding). Nucl. Eng. Des. 23, 295–308.
Eichhorn, R., Kao, H.C., Neti, S., 1980. Measurements of shear stress in a square array rod bundle. Nucl. Eng. Des. 56, 385–391.
Eifler, W., Nijsing, R., 1967. Experimental investigation of velocity distribution and flow resistance in a triangular array of parallel rods. Nucl. Eng. Des. 5, 22–42.
Engle, F.C., Minushkin, B., Atkins, R.J., Markley, R.A., 1980. Characterization of heat transfer and temperature distributions in an electrically heated model of an LMFBR blanket assembly. Nucl. Eng. Des. 62 (1980), 335–347.
Gosset, A., Tavoularis, S., 2006. Laminar flow instability in a rectangular channel with a cylindrical core. Phys. Fluids 18.
Grover, R.B., Venkat Raj, V., 1980. Pressure drop along longitudinally-finned seven-rod cluster nuclear fuel elements. Nucl. Eng. Des. 58, 79.
Guellouz, M.S., Tavoularis, S., 1992. Heat transfer in rod bundle subchannels with varying rod-wall proximity. Nucl. Eng. Des. 132, 351–366.
Hochreiter, L.E., Cheung, F.B., Lin, T.F., Frepoli, C., Sridharan, A., Todd, D.R., Rosal, E.R., 2010. Rod Bundle Heat Transfer Test Facility Test Plan and Design, NUREG/CR-6975.
Holloway, M.V., Beasley, D.E., Conner, M.E., 2008. Single-phase convective heat transfer in rod bundles. Nucl. Eng. Des. 238, 848–858.
Hooper, J.D., Rehme, K., 1983. The structure of single-phase turbulent flows through closely spaced rod arrays, KfK Report 3467, Kernforschungszentrum Karlsruhe (1983).
Hudina, M., Markbczy, G., 1977. The hexagonal bundle heat transfer and fluid flow experiment AGATHE HEX. Nucl. Eng. Des. 40 (1), 121–131 Special Issue GCFR.
Hudina, M., Huggenberger, M., 1986. Pressure drop and heat transfer in gas-cooled rod bundles. Nucl. Eng. Des. 97, 347–360.
Ibragimov, M.C., Usupov, I.A., Subbotin, V.I., 1967. The calculation and experimental investigation of velocity distribution in noncircular channels of complex form (transl.), Liquid Metal, 234, Atomizdat (1967).
Kawahara, A., Sato, Y., Sadatomi, M., 1997. The turbulent mixing rate and the fluctuations of static pressure difference between adjacent subchannels in a two-phase subchannel flow. Nucl. Eng. Des. 175, 97–106.
Kelly, J.E., 1980. Development of Two-fluid Two-phase model for light water reactor subchannel Analysis, PhD Thesis.
Kjellstrom, B., 1972. Studies of Turbulent Flow Parallel to a Rod Bundle of Triangular Array. AB Atomenergie, Studsvik, Sweeden AE-RL-1344.
Krauss, T., Meyer, L., 1996. Characteristics of turbulent velocity and temperature in a wall channel of a heated rod bundle. Exp. Thermal Fluid Sci. 1996 (12), 75–86.
Krauss, T., Meyer, L., 1998. Experimental investigation of turbulent transport of momentum and energy in a heated rod bundle. Nucl. Eng. Des. 180, 185–206.
Lahey, R.T., Shiralkar, B.S., Radcliffe, O.W., Subchannel and Pressure Drop Measurements in a Nine-Rod Bundle for Diabatic and Adiabatic Conditions, March 1970, (GEAP-13049).
Meyer, Leonhard., 2010. From discovery to recognition of periodic large scale vortices in rod bundles as source of natural mixing between subchannels-a review. Nucl. Eng. Des. 240, 1575–1588.
Maubach, K., Rehme, K., 1973. Pressure drop for parallel flow through a roughened rod cluster. Nucl. Eng. Des. 25, 369–378.
Meyer, L., 1994. Measurements of turbulent velocity and temperature in axial flow through a heated rod bundle. Nucl. Eng. Des. 146, 71–82.
Moller, S.V., 1991. On phenomena of turbulent flow through rod bundles. Exp. Thermal Fluid Sci. 4, 25–35.
Moller, S.V., 1992. Single-phase turbulent mixing in rod bundles. Exp. Thermal Fluid Sci. 5, 26–33.
Silin, Nicolas, Juanico, Luis, 2006. Experimental study on the Reynolds number dependence of turbulent mixing in a rod bundle. Nucl. Eng. Des. 236, 1860–1866.
Neti, S., Eichhorn, R., Hahn, O.J., 1983. Laser Doppler measurements of flow in a rod bundle. Nucl. Eng. Des. 74, 105–116.
Nijsing, R., Eifler, W., Delfau, B., Camposilvan, J., 1967. Studies on fluid mixing between sub-channels in a bundle of finned tubes. Nucl. Eng. Des. 5, 229–253.
Nijsing, R., Gargantini, I., Eifler, W., 1966. Analysis of fluid flow and heat transfer in a triangular array of parallel heat generating rods. Nucl. Eng. Des. 4, 375–398.
Novendstern, E.H., 1972. Turbulent flow pressure drop model for fuel rod assemblies utilizing a helical wire wrap spacer system. Nucl. Eng. Des. 22 (1972), 19–27.
Ouma, B.H., Tavoularis, S., 1991. Flow measurements in rod bundle subchannels with varying rod-wall proximity. Nucl. Eng. Des. 131, 193–208.
Rehme, K., 1976. Hot-wire anemometry in subchannels of rod bundles, KTG-Fachtagung Experimentiertechnik in der Reaktor-Fluiddynamik, Berlin, 1976 (in German).
Rehme, K., 1977. Measurements of the velocity, turbulence and wall shear stress distributions in a corner channel of a rod bundle, KfK 2512, Kernforschungszentrum Karlsruhe (1977, in German).
Rehme, K., 1978. The structure of turbulent flow through a wall subchannel of a rod bundle. Nucl. Eng. Des. 45, 311–323.
Rehme, K., 1979. The Structure of turbulent flow through subchannels of rod bundles. In: Yao, S.C., Psand, T.A., Fluid Flow and Heat Transfer Over Rod or Tube Bundles, ASME, New York, 1979, pp. 67–76.
Rehme, K., 1980a. Experiments on the structure of turbulence in a wall subchannel of a rod bundle (P/D = 1.07), KfK 2983, Kernforschungszentrum Karlsruhe (1980, in German).
Rehme, K., 1981. Structure of turbulence in a wall subchannel of a rod bundle, KfK 3177, kernforschungszentrum Karlsruhe (1981, in German).
Rehme, K., 1984a. Computer-controlled rig for measurements of velocity and turbulence distributions by hot-wires, KfK 3744, Kernforsehungszentrum Karlsruhe (1984, in Germarn).
Rehme, K., 1984b. Comparison between results obtained by different hot-wire methods, KfK 3813, Kernforschungszentrum Karlsruhe (1984, in German).
Rehme, K., 1986. Turbulence measurements in a wall subchannel of a rod bundle (P/ D = W/D = 1.148), KfK-Report 4027 (1986).
Rehme, K., 1987a. The structure of turbulent flow through rod bundles. Nucl. Eng. Des. 99, 141–154.
Rehme, K., 1987b. The structure of turbulence in wall sub channels of a rod bundle. Atomkernenergie 49, 145–150.
Rehme, K., 1992. The structure of turbulence in rod bundles and the implications on natural mixing between subchannels. Int. J. Heat Mass Transfer 35, 567–581.
Rehme, K., 1980b. Experimental investigations on the fluid flow through an asymmetric rod bundle (W/D = 1.096), KfK 3047, Kernnforschungszentrum Karlsruhe (1980, in German).
Rehme, K., 1980c. Experimental investigations on the fluid flow through an asymmetric rod bundle (W/D = 1.048), KfK 3069, Kemforschungszentrum Karlsruhe (1980, in German).
Rehme, K., 1982a. Experimental investigations on the fluid flow through an asymmetric rod bundle (W/D = 1.118), KfK 3318, Kernforschungszentrum Karlsruhe (1982, in German).
Rehme, K., 1982b. Experimental investigations on the fluid flow through an asymmetric rod bundle, KfK 3324, Karlsruhe (in German) (1982).
Rehme, K., 1982c. Experimental investigations on the fluid flow through a wall subchannel of a rod bundle (P/D = 1.036, W/D = 1.072), KfK 3361, Kernforschungszentrum Karlsruhe (1982, in German).
Rehme, K., 1983a. Experimental investigations on the fluid flow through an asymmetric rod bundle (P/D = 1.148, W/D = 1.045), KfK 3597, Kemforschungszentrum Karlsruhe (1983, in German).
Rehme, K., 1983b. Experimental investigation on the fluid flow through an asymmetric rod bundle (P/D = 1.148, W/D = 1.252), KfK 3598, Kernforschungszentrum Karlsruhe (1983, in German).
Rehme, K., 1984. Experimental investigations on the fluid flow through an asymmetric rod bundle (P/D = 1.148, W/D = 1.074), KfK 3818, Kernforschungszentrum Karlsruhe (1984, in German).
Rehme, K., 1985. Experimental investigations on the fluid flow through an asymmetric rod bundle (P/D = 1.148, W/D = 1.222), KfK 3822, Kernforschungszentrum Karlsruhe (1985, in German).
Rehme, K., 1989. Experimental observations of turbulent flow through subchannels of rod bundles. Exp. Thermal Fluid Sci. 2, 341–349.
Renksizbulut, M., Hadaller, G.I., 1986. An experimental study of turbulent flow through a square-array rod bundle. Nucl. Eng. Des. 91, 41–55.
Roidt, M., Pechersky, M.J., Markley, R.A., Vegter, B.J., 1973. Experimental determination of turbulent exchange coefficients in a model reactor rod bundle. Conf-730803-2.
Roidt, R.M., Bartholet, T.G., Harper, L.J., 1974. Experimental determination of interior subchannel cross flow and axial flow in a model of the Clinch River breeder reactor fuel assembly rod bundle with wire wrap spacers., conf-741107-4.
Roidt, R.M., Carelli, M.D., Markley, R.A., 1980. Experimental investigations of the hydraulic field in wire-wrapped LMFBR core assemblies. Nucl. Eng. Des. 62, 295–321.
Rogers, J.T., Rosehart. R.G., 1972. Mixing by turbulent interchange in fuel bundles. Correlations and interfaces, 72-HT-53.
Rogers, J.T., Tahir, A.E.E., 1975. Turbulent interchange mixing in rod bundles and role of secondary flows, 75-HT-31.
Rowe, D.S., 1967, Cross-Flow mixing between parallel flow channels during boiling Part-I COBRA-Computer program for coolant boiling in rod arrays. BNWL-371-Pt-I.
Rowe, D.S., Angle, C.W., 1967. Cross flow mixing between parallel flow channels during boiling part ii measurement of flow and enthalpy in two parallel channels, BNWL-371-Pt-2.
Rowe, D.S., Angle, C.W., 1969. Cross Flow mixing between paralell flow channels during boiling Part III-Effect of spacers on mixing between two channels, BNWL-371-Pt-3.
Rowe, D.S., Johnson, B.M., Knudsen, J.G., 1974. Implications concerning rod bundle cross flow mixing based on measurements of turbulent flow structure. Int. J. Heat Mass Transfer 17, 407–419.
Sadatomi, M., Kawahara, A., Sato, Y., 1994. Flow redistribution due to void drift in two-phase flow in a multiple channel consisting of two subchannels. Nucl. Eng. Des. 148, 463–474.
Seale, W.J., 1979. Turbulent diffusion of heat between connected flow passages. Nucl. Eng. Des. 54, 183–195.
Seale, W.J., 1979. Turbulent diffusion of heat between connected flow passages; Part 2:

prediction using K – e turbulence model. Nucl. Eng. Des. 54, 197–209.
Shin, B.S., Chang, S.H., 2005. Experimental study on the effect of angles and positions of mixing vanes on CHF in a 2 × 2 rod bundle with working fluid R-134a. Nucl. Eng. Des. 235, 1749–1759.
Shin, B.S., Chang, S.H., 2009. CHF experiment and CFD analysis in a 2 × 3 rod bundle with mixing vane. Nucl. Eng. Des. 239, 899–912.
Silin, N., Juanico, L., Delmastro, D., 2004. Thermal mixing between subchannels: measurement method and applications. Nucl. Eng. Des. 227, 51–63.
Skinner, V.R., Freeman, A.R., Lyall, H.G., 1969. Gas mixing in rod clusters. Int. J. Heat Mass Transfer 12, 265–278.
Tachibana, F., Oyama, A., Akiyama, M., Kondo, S., 1969. Measurement of heat transfer coefficients for axial air flow through eccentric annulus and seven-rod cluster. J. Nucl. Sci. Technol. 6, 207–214.
Trupp, A.C., Azad, R.S., 1975. The structure of turbulent flow in triangular array rod bundles. Nucl. Eng. Des. 32, 47–84.
Van Der Ros, T., Bogaart, M., 1970. Mass and heat exchange between adjacent channels in liquid-cooled rod bundles. Nucl. Eng. Des. 12, 259.
Vonka, V., 1988. Measurement of secondary flow vortices in a rod bundle. Nucl. Eng. Des. 106, 191–207.
Walton, F.B., 1969. Turbulent mixing measurements for single phase air, single phase water and two phase air-water flows in adjacent triangular subchannels, MS-Thesis.
Wolf, L., Fischer, K., Herkenrath, H., Hufschmidt, W., 1987. Comprehensive assessment of the Ispra BWR and PWR subchannel experiments and code analysis with different two-phase models and solution schemes. Nucl. Eng. Des. 99, 329–350.
Wu, X., 1995. On the transport mechanisms in simulated heterogeneous rod bundle subchannels. Nucl. Eng. Des. 158, 125–134.
Wu, X., Trupp, A.C., 1994. Spectral measurements and mixing correlation in simulated rod bundle subchannels. Int. J. Heat Mass Transfer 37, 1277–1281.
Yue, F.S., Zi, D.C., Qing, G.L., 1991. An investigation of cross flow mixing effect caused by grid spacer with mixing blades in a rod bundle. Nucl. Eng. Des. 125, 111–119.

C. Rod bundle analytical works

Ang, M.L., Aytekin, A., Fox, A.H., 1987. Analysis of flow distribution in a PWR fuel rod bundle model containing a 90% blockage. Nucl. Eng. Des. 103, 165–188.
Axford, R.A., 1967. Two-dimensional multi region analysis of temperature fields in reactor tube bundles. Nucl. Eng. Des. 6, 25.
Baglietto, E., Ninokata, H., 2003. Turbulence models for heat transfer simulation in tight lattice fuel bundles. In: 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea, October-2003, 5–9.
Bishop, A.A., Todreas, N.E., 1979. Hydraulic characteristics of wire-wrapped rod bundles. Nucl. Eng. Des. 62, 271–293.
Carlucci, L.N., Hammouda, N., Rowe, D.S., 2004. Two-phase turbulent mixing and buoyancy drift in rod bundles. Nucl. Eng. Des. 227, 65–84.
Castellana, F.S., Adams, W.T., Casterline, J.E., 1974. Single phase sub-channel mixing in a simulated nuclear fuel assembly. Nucl. Eng. Des. 26, 242–249.
Chang, D., Tavoularis, S., 2007. Numerical simulation of turbulent flow in a 37-rod bundle. Nucl. Eng. Des. 237, 575–590.
Chen, B.C.J., Vanka, S.P., Sha, W.T., 1980. Some recent computations of rod bundle thermal hydraulics using boundary fitted coordinates. Nucl. Eng. Des. 62, 123–135.
Cheng, S.K., Todreas, N.E., 1986. Hydrodynamic models and correlation for bare, and wire-wrapped hexagonal rod bundles – bundle friction factor, subchannel friction factors and mixing parameters. Nucl. Eng. Des. 92, 227–251.
Hwang, Dae-Hyun, Yoo, Yeon-Jong, In, Wang-Kee, Zee, Sung-Quun, 2000. Assessment of the interchannel mixing model with a subchannel analysis code for BWR and PWR conditions. Nucl. Eng. Des. 199, 257–272.
Dwyer, O.E., 1966. Analytical study of heat transfer to liquid metals flowing in-line through closely packed rod bundies. Nucl. Sci. Eng. 25, 343.
Eiff, O.S., Lightstone, M.F., 1996. On the modelling of single-phase turbulent energy transport in sub-channels.
Hazi, Gabor, 2005. On turbulence models for rod bundle flow computations. Ann. Nucl. Energy 32, 755–761.
Ginsberg, T., 1971. Forced-flow interchannel mixing model for fuel rod assemblies utilizing a helical wire-wrap spacer system. Nucl. Eng. Des. 22, 28–42.
Jeong, Hae-Yong, Ha, Kwi-Seok, Kwon, Young-Min, Lee, Yong-Bum, Hahn, Dohee, 2007. A dominant geometrical parameter affecting the turbulent mixing rate in rod bundles. Int. J. Heat Mass Transfer 50, 908–918.
Jeong, Hae-Yong, Ha, Kwi-Seok, Kwon, Young-Min, Chang, Won-Pyo, Lee, Yong-Bum, 2006. A correlation for single phase turbulent mixing in square rod arrays under highly turbulent conditions. Nucl. Eng. Technol. 38, 809–818.
Jong, Hee Cha, Moon, Haeng Cho, 1970. A study on coolant mixing in multirod bundles subchannels. J. Korean Nucl. Soc. 2 (1970), 19–25.
Hisashi, Hishida, 1974. An analytical method of evaluating the effect of rod displacement on circumferential temperature and heat flux distributions in turbulent flow through unbaffled rod bundles. Nucl. Eng. Des. 26, 408–422.
Hofmann, F., 1970. Velocity and temperature distribution in turbulent flow in sodium cooled fuel elements with eccentric geometry. Nucl. Eng. Des. 14, 43.
Hooper, J.D., 1980. Developed single phase turbulent flow through a square pitch rod cluster. Nucl. Eng. Des. 60, 365–379.
Hooper, J.D., Wood, D.H., 1983. Flow recovery from a single subchannel blockage in a square-pitched array. Nucl. Eng. Des. 74, 91–103.
Hooper, J.D., Rehme, K., 1984. Large-scale structure effects in developed turbulent flow through closely spaced rod arrays. J. Fluid Mech. 145, 305–337.
Hooper, J.D., Wood, D.H., 1984. Fully developed rod bundle flow over a large range of Reynolds number. Nucl. Eng. Des. 83, 31–46.
Jian, S., Atila, P., Silva, F., 2002. Analytical prediction of friction factors and Nusselt numbers of turbulent forced convection in rod bundles with smooth and rough surfaces. Nucl. Eng. Des. 215, 111–127.
Kaiser, H.G., Zeggel, W., 1987. Turbulent flows in complex rod bundle geometries numerically predicted by the use of fem and a basic turbulence model. Nucl. Eng. Des. 99, 351–363.
Kawahara, A., Sadatomi, M., Kudo, H., Kano, K., 2006. Single and Two-Phase Turbulent mixing rate between subchannels in triangular Tight lattice rod bundle. JSME Int. J., Ser. B 49, 287–295.
Kazuyuki Takase, Hisashi Ninokata, 2007. Large-Scale Simulations on Thermal-Hydraulics in Fuel Bundles of Advanced Nuclear Reactors, CH.3-Epoch-Making Simulation.
Khan, E.U., Rohsenkow, W.M., Sonin, A.A., Todreas, N.E., 1975. A porous body model for predicting temperature distribution in wire-wrapped fuel rod assemblies. Nucl. Eng. Des. 35, 1–12.
Kriventsev, V., Ninokata, H., 1999. Calculation of detailed velocity and temperature distributions in a rod bundle of nuclear reactor. In: Proceedings of the Proceedings of the Ninth International Topical Meeting on Nuclear Reactor Thermal Hydraulics. NURETH-9, 3–8 October-1999, San Francisco, CA.
Lahey, R.T., Schruab, F.A., 1969. Mixing, Flow Regimes, and Void Fraction for Two-Phase Flow in Rod Bundles.
Lee, C.M., Choi, Y.D., 2007. Comparison of thermo-hydraulic performances of large scale vortex flow (LSVF) and small scale vortex flow (SSVF) mixing vanes in 17_17 nuclear rod bundle. Nucl. Eng. Des. 237, 2322–2331.
Lee, K.B., 1995. Analytical prediction of subchannel friction factor for infinite bare rod square and triangular arrays of low pitch to diameter ratio in turbulent flow. Nucl. Eng. Des. 157, 197–203.
Lee, K.B., Jang, H.C., 1997. A numerical prediction on the turbulent flow in closely spaced bare rod arrays by a nonlinear k–ε model. Nucl. Eng. Des. 172, 351–357.
Levchenko, J.D., Subbotin, V.I., Ushakow, P.A., 1967. Coolant velocity and wall shear stress distribution in closely packed rods. Atomnaja Energija 22 (1967), 218–225.
Lewis, M.J., Buettiker, P., 1974. Momentum losses and convective heat transfer in rod bundles - an overview. Nucl. Eng. Des. 31, 351–362.
Marek, J., Rehme, K., 1979. Heat transfer in smooth and roughened rod bundles near spacer grids. In: Fluid Flow and Heat Transfer Over rod or tube Bundles. ASME, Presented at the ASME Winter Annual Meeting, December 2–7, 1979, pp. 163–170.
Marek, J., Maubach, K., Rehme, K., 1973. Heat transfer and pressure drop performance of rod bundles arranged in square arrays. Int. J. Heat Mass Transfer 16, 2215–2228.
Matiur Rahman, Ahmad, S.Y., 1982. Finite element analysis of axial flow with heat transfer in a square duct, AMM-6-481-490.
Meyder, R., 1975. Solving the Conservation Equations in Fuel Rod Bundles Exposed to Parallel Flow by Means of Curvilinear-Orthogonal Coordinates. JCP-17-53-67.
Meyder, R., 1975. Turbulent velocity and temperature distribution in the central subchannel of rod bundles. Nucl. Eng. Des. 35, 181–189.
Meyer, L., Rehme, K., 1994. Large scale turbulence phenomena in compound rectangular channels. Exp. Thermal Fluid Sci. 8, 286–304.
Misawa, T., Ninokata, H., Maekawa, I., 2003. Calculation of detailed velocity and temperature distributions in a triangular pin bundle using pseudo direct numerical simulation. In: 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea, October 5–9.
Mohanty, A.K., Sahoo, K.M., 1986. Laminar convection in wall sub-channels and transport rates for finite rod bundle assemblies by superposition. Nucl. Eng. Des. 92, 169–180.
Mohanty, A.K., Sahoo, K.M., 1988. Turbulent flow and heat transfer in rod-bundle subchannels. Nucl. Eng. Des. 106, 327–344.
Nijsing, R., Eifler, W., 1969. Analysis of liquid metal heat transfer in assemblies of closed spaced fuel rods(A theoretical evaluation of two-dimensional temperature- and heat flux distribution for conditions of turbulent axial flow at relatively low Peclet numbers). Nucl. Eng. Des. 10, 21–54.
Nijsing, R., Eifler, W., Delfau, B., 1975. Lateral turbulent diffusion for longitudinal flow in a rectangular channel. Nucl. Eng. Des. 32, 221–238.
Nijsing, R., Eifler, W., Delfan, B., Camposilvan, J., 1967. Studies on fluid mixing between subchannels in a bundle of finned tubes. Nucl. Eng. Des. 5, 229–253.
Nijsing, R., Gargantini, I., Eifler, W., 1964. Fundamental studies of fluid flow and heat transfer in fuel element geometries, part I, EUR 2193.e, Joint Nuclear Research Centre, Ispra (1964).
Ninokata, H., Merzari, E., Khakim, A., 2009. Analysis of low Reynolds number turbulent flow phenomena in nuclear fuel pin subassemblies of tight lattice configuration. Nucl. Eng. Des. 239, 855–866.
Ninokata, H., Todreas, N.E., 1975. Turbulent momentum exchange coefficients for reactor fuel bundle analysis. Report 000-2245-22TR, Nucl. Eng. Dep., MIT (1975).
Ninokata, H., Efthimiadis, A., Todreas, N.E., 1987. Distributed resistance modeling of wire wrapped rod bundles. Nucl. Eng. Des. 104, 93–102.
Okhawa, K., Lahey, R.T., 1980. The analysis of CCFL using drift-flux models. Nucl. Eng. Des. 61.
Jaroslav, Pfann., 1972. Heat transfer in turbulent longitudinal flow through unbaffled assemblies of fuel rods. Nucl. Eng. Des. 25, 217–247.
Petrunik, 1973. Turbulent interchange in simulated rod bundle geometries for genetron-12., PhD Thesis.
Ramm, H., Johannsen, K., Todreas, N.E., 1974. Single phase transport within bare-rod arrays at laminar, transition and turbulent flow condition. Nucl. Eng. Des. 30, 186–204.
Rapley, C.W., Gosman, A.D., 1986. The prediction of fully developed axial turbulent flow in rod bundles. Nucl. Eng. Des. 97, 313–325.
Rehme, K., 1972. Pressure drop performance of rod bundles in hexagonal arrangements. Int. J. Heat Mass Transfer 15, 2499–2517.

Rehme, K., 1973a. Simple method of predicting friction factors of turbulent flow in non-circular channels. Int. J. Heat Mass Transfer 16 (1973), 933–950.
Rehme, K., 1973b. Pressure drop correlations for fuel element spacers. Nucl. Technol. 17, 15–23.
Rehme, K., 1978a. Turbulent flow through a wall subchannel of a rod bundle, KfK-2617, Kernforschungszentrum Karlsruhe (1978, in German).
Rehme, K., 1978b. Distributions of velocity and turbulence of turbulent flow through a wall subchannel of a rod bundle, KfK-2637, Kernforschungszentrum Kaflsruhe (1978, in German).
Rehme, K., 1979. Non-isotropic eddy viscosities in turbulent flow through rod bundles. In: Kakac, S., Spalding, D.B., (Eds.), Turbulent Forced Convection in Channels and Bundles, Vol. I, Hemisphere/McGraw-Hill, New York, 1979, pp. 505–519.
Rehme, K., 1980. Turbulent momentum transport in rod bundles. Nucl. Eng. Des. 62, 137–146.
Rehme, K., 1982. Distributions of velocity and turbulence in a parallel flow along an asymmetric rod bundle. Nucl. Technol. 59, 148–159.
Rehme, K., 1992. The prediction of turbulent Prandtl and Schmidt numbers. Int. J. Heat Mass Transfer 18, 1055–1069.
Schikorr, M., Bubelis, E., Mansani, L., Litfin, K., 2010. Proposal for pressure drop prediction for a fuel bundle with grid spacers using Rehme pressure drop correlations. Nucl. Eng. Des. 240, 1830–1842.
Kim, Sin, Chung, Bum-Jin, 1999. Estimation of the turbulent mixing rate for low Prandtl number flows through rod bundles. Int. Commun. Heat Mass Transfer 26 (7), 1009–1017.
Kim, Sin, Chung, Bum-Jin, 2001. A scale analysis of the turbulent mixing rate for various Prandtl number flow fields in rod bundles. Nucl. Eng. Des. 205, 281–294.
Spatz, R., Mewes, D., 1987. The influence of walls and upper tie plate slots on the flooding mechanism in fuel elements with and without heat transfer between steam and water. Nucl. Eng. Des. 110, 413–422.
Su, K., Freire, A.P.S., 2002. Analytical prediction of friction factors and Nusselt numbers of turbulent forced convection in rod bundles with smooth and rough surfaces. Nucl. Eng. Des. 217, 111–127.
Subbotin, V.I., Ushakov, P.A., Gabrianovich, B.N., Zhukov, A.B., 1961. Heat exchange during the flow of mercury and water in a tightly packed rod pile. Soviet J. Atom. Energy 9, 1001–1009.
Subbotin, V.I., Ushakov, P.A., Levchenko, Y.D., Aleksandrov, A.M., 1971. Velocity fields in turbulent flow past rod bundles. Heat Transfer, Soviet Res. 3 (2), 9–35.
Tae Sun, R.O., Todreas, N.E., 1988. Energy transfer mechanisms in LMFBR rod bundles under mixed convection conditions. Nucl. Eng. Des. 108, 343–357.
Tapucu, A., Teyssedou, A., Tye, P., Troche, N., 1994. The effect of turbulent mixing models on the prediction of subchannel codes. Nucl. Eng. Des. 149, 221–231.
Thompson, J.J., Holy, Z.J., 1975. Conjugate heat transfer and thermo elastic analysis of heat generating rods. Nucl. Eng. Des. 35, 247–268.
Todreas, N.E., 1979. Coolant mixing in LMFBR rod bundles and outlet plenum mixing transients, DOE/ET/37240-72.
Todreas, N.E., 1985. Coolant mixing in LMFBR rod bundles and outlet plenum mixing transients. DOE/ET/37240-109-PR.
Tye, P., Teyssedou, A., Tapucu, A., 1994. An investigation of the constitutive relations for intersubchannel transfer mechanisms in horizontal flows as applied in the ASSERT-4 subchannel code. Nucl. Eng. Des. 149, 207–220.
Vonka, V., Boonstra, B.H., 1974. Calculated heat transfer development in bundles. Nucl. Eng. Des. 31, 337–345.
Xiaohua, Wu., 1995. On the transport mechanisms in simulated heterogeneous rod bundle subchannels. Nucl. Eng. Des. 158, 125–134.
Zaiyong Ma, Yingwei Wu, Zicheng Qiu, Wenxi Tian, Guanghui Su, Suizheng Qiu, 2012. An innovative method for prediction of liquid metal heat transfer rate for rod bundles based on annuli, ANE-47-91-97.

D. Rod bundle CFD works

Anglart, H., Nylund, O., 1996. CFD application to prediction of void distribution in bubbly flows in rod bundles. Nucl. Eng. Des. 163 (1996), 81–98.
Papukchiev, Angel, Roelofs, Ferry, Shams, Afaque, Lecrivain, Gregory, Ambrosini, Walter, 2015. Development and application of computational fluid dynamics approaches within the European project THINS for the simulation of next generation nuclear power systems. Nucl. Eng. Des. 290, 13–26.
Baglietto, E., Ninokata, H., 2004. CFD Modeling of Secondary Flows in Fuel Rod Bundles. NUTHOS-6, Nara, Japan.
Baglietto, E., Ninokata, H., 2005. A turbulence model study for simulating flow inside tight lattice rod bundles. Nucl. Eng. Des. 235 (2005), 773–784.
Baglietto, E., Ninokata, H., Misawa, T., 2006. CFD and DNS methodologies development for fuel bundle simulations. Nucl. Eng. Des. 236, 1503–1510.
Biemüller, M., Meyer, L., Rehme, K., 1996. Large eddy simulation and measurement of the structure of turbulence in two rectangular channels connected by a gap. In: Heraklion, G.R., Rodi, W. (Publ.), Proc. 3rd Internat. Symp., Eng. Turbulence Modelling and Experiments, Elsevier, Amsterdam (u.a.), pp. 249–258.
Carver, M.B., Tahir, A., Rowe, D.S., Tapucu, A., Ahmad, S.Y., 1984. Computational analysis of two-phase flow in horizontal bundles. Nucl. Eng. Des.
Chandra, L., Roelofs, F., Houkema, M., Jonker, B., 2009. A stepwise development and validation of a RANS based CFD modelling approach for the hydraulic and thermal-hydraulic analyses of liquid metal flow in a fuel assembly. Nucl. Eng. Des. 239, 1988–2003.
Chang, D., Tavoularis, S., 2008. Simulations of turbulence, heat transfer and mixing across narrow gaps between rod-bundle subchannels. Nucl. Eng. Des. 238, 109–123.
Cheng, X., Tak, N.I., 2006. CFD analysis of thermal–hydraulic behavior of heavy liquid metals in sub-channels. Nucl. Eng. Des. 236, 1874–1885.
Cinosi, N., Walker, S.P., 2016. CFD analysis of localized crud effects on the flow of coolant in nuclear rod bundles. Nucl. Eng. Des. 305, 28–38.
Conner, M.E., Baglietto, E., Elmahdi, A.M., 2010. CFD methodology and validation for single-phase flow in PWR fuel assemblies. Nucl. Eng. Des. 240, 2088–2095.
Gajapathy, R., Velusamy, K., Selvaraj, P., Chellapandi, P., Chetal, S., 2009. A comparative CFD investigation of helical wire-wrapped 7, 19, 37 fuel pin bundles and its extendibility to 217 pin bundle. Nucl. Eng. Des. 239, 2279–2292.
Gu, H.Y., Cheng, X., Yang, Y.H., 2008. CFD analysis of thermal hydraulic behavior in SCWR typical flow channels. Nucl. Eng. Des. 238, 3348–3359.
Hu, R., Fanning, T.H., 2011. Progress Report on Development of Intermediate Fidelity Full Assembly Analysis Methods. September-2011. ANL/NE-11/35.
Ikeda, K., Makino, Y., Hoshi, M., 2006. Single-phase CFD applicability for estimating fluid hot-spot locations in a 5 × 5 fuel rod bundle. Nucl. Eng. Des. 236, 1149–1154.
In, W.K., Chun, T.H., Shin, C.H., Oh, D.S., 2008. Numerical computation of heat transfer enhancement of a PWR rod bundle with mixing vane spacers. Nucl. Technol. 161, 69–79.
Krepper, E., Končar, B., Egorov, Y., 2007. CFD modelling of subcooled boiling-Concept, validation and application to fuel assembly design. Nucl. Eng. Des. 237, 716–731.
Lee, K.B., Jang, H.G., 1997. A numerical prediction on the turbulent flow in closely spaced bare rod arrays by a nonlinear k-ε model. Nucl. Eng. Des. 172 (1997), 351–357.
Liu, C.C., Ferng, Y.M., 2010. Numerically simulating the thermal-hydraulic characteristics within the fuel rod bundle using CFD methodology. Nucl. Eng. Des. 240, 3078–3086.
Merzari, E., Ninokata, H., Baglietto, E., 2007. Unsteady Reynolds averaged Navier-Stokes simulation for an accurate prediction of the flow insight tight rod bundles. Proc. NURETH-12, Log number 213, Pennsylvania, USA.
Merzari, E., Ninokata, H., Baglietto, E., 2008. Numerical simulation of flows in tight lattice fuel bundles. Nucl. Eng. Des. 238, 1703–1719.
Natesan, K., Sundararajan, T., Narasimhan, A., Velusamy, K., 2010. Turbulent flow simulation in a wire-wrap rod bundle of an LMFBR. Nucl. Eng. Des. 240, 1063–1072.
Ninokata, H., Atake, N., Baglietto, E., Misawa, T., Kano, T., 2004. Direct numerical simulation of turbulence flows in a subchannel of tight lattice fuel pin bundles of nuclear reactors. http://www.Jamstec.go.jp/esc/publication /annual/annual2004/.
Ninokata, H., Merzari, E., 2007. Computational fluid dynamics and simulation based-design approach for tight lattice nuclear fuel pin subassemblies. In: Proc.NURETH-12, Log no: KN#6, Pennsylvania, USA.
Roelofs, F., Gopala, V.R., Chandra, L., Viellieber, M., Class, A., 2012. Simulating fuel assemblies with low resolution CFD approaches. Nucl. Eng. Des. 250, 548–559.
Chang, Seok Kyu, Moon, Sang Ki, Baek, Won Pil, Choi, Young Don, 2008. Phenomenological investigations on the turbulent flow structures in a rod bundle array with mixing devices. Nucl. Eng. Des. 238, 600–609.
Toth, S., Aszodi, A., 2010. CFD analysis of flow field in a triangular rod bundle. Nucl. Eng. Des. 240, 352–363.
Ikeno, Tsutomu, Kajishima, Takeo, 2010. Analysis of dynamical flow structure in a square arrayed rod bundle. Nucl. Eng. Des. 240, 305–312.
Yan, B.H., Yu, L., Gu, H.Y., Yang, Y.H., 2011. LES and URANS simulations of turbulent flow in 4 rod bundles in ocean environment. Prog. Nucl. Energy 53, 330–335.
Yan, B.H., Gu, H.Y., Yu, L., 2012. Numerical simulation of the coherent structure and turbulent mixing in tight lattice. Prog. Nucl. Energy 54 (2012), 81–95.
Shang, Zhi, 2009. CFD investigation of vertical rod bundles of supercritical water-cooled nuclear reactor. Nucl. Eng. Des. 239, 2562–2572.
Shang, Zhi, Lo, Simon, 2010. Numerical investigation of supercritical water-cooled nuclear reactor in horizontal rod bundles. Nucl. Eng. Des. 240, 776–782.