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Investigation of cross flow mixing on rod bundle safety margins using sub-channel analysis framework

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ABSTRACT

An investigation on effect of cross flow mixing on rod bundle thermal hydraulic safety margins is carried out using a sub-channel analysis framework. In the present study, general purpose automated sub-channel analysis framework is developed to carry out the thermal hydraulic analysis of nuclear reactor core. The framework involves the preprocessor to generate the sub-channel layout of the fuel assemblies of different sizes and shapes such as square and triangular pitched pin bundle. The analyses are carried out for different simulated rod bundles of square pitched array of typical nuclear fuel assembly. The parameters considered in this study are the size of the bundle, p/d ratio, intensity of turbulent mixing, mass flux, uniform and non-uniform axial power distribution. The coolant local conditions are significantly affected due to the inter channel mixing between sub-channels. The sub-channel layouts for fuel bundles of different sizes varying from 2 × 2 to 10 × 10 and 17 × 17 for same p/d and w/d ratios of 1.05 to 1.3 are generated. The analyses are carried out by varying the intensity of turbulent mixing parameter (c) from 0 (no mixing) to 0.1 (high mixing) using the developed framework. It is found that hot channel temperature decreased by 26% in smaller bundle due to strong interaction between wall and corner channel. In case of large sized assembly above 5 × 5, hot channel temperature is less affected. The estimation of coolant temperature, fuel pin surface temperature and local quality are also carried out without considering the inter-channel mixing. The results are compared with a change in the local conditions for different degree of mixing among the fuel bundle sub-channels. The developed Framework is used for performing the sensitivity-studies during the design and analysis phase of nuclear reactor core for estimation of thermal hydraulic safety margins in an efficient way with minimum human intervention.

1. Introduction

The safety of nuclear reactor core is ensured based on accurate estimation of local conditions of the coolant in the fuel assembly/sub-channels. The coolant flow distribution among bundles is affected due to non-uniformity of entry conditions at the inlet plenum that is also measured by hydraulic models and experiments (Hetsroni, 1967) or predicted numerically by computational fluid dynamic (CFD) codes. The coolant local conditions are also affected very much due to the inter channel mixing between sub-channels. Cross-flow and mixing of water between semi-open channels is studied by Hetsroni et al. (1968) by the measurement of heat transfer between hot and cold water and overall effective diffusivity of heat is derived as a function of Reynolds number. A two-dimensional turbulent flow mixing model for parallel-flow rod bundles in terms of fluid enthalpy rise is formulated by Lowe Phillip (1968). He found that mixing was a strong function of non-dimensional geometry factor and Peclet number. A theoretical analysis was performed to study molecular and turbulent transport phenomena between sub-channels by Ramm et al. (1974). They studied the single phase transport within bare rod arrays at laminar, transition and turbulent flow conditions. Lateral turbulent diffusion of mass for longitudinal flow in a rectangular channel for a range of Reynolds number 30,000 to 86,000 was carried out by Nijsing et al. (1975). They correlated the turbulent diffusivity of mass by eddy diffusion of momentum. Tapucu (1977) carried out studies on pressure induced diversion cross flow between two parallel channels communicating by a lateral slot based on transverse flow resistance coefficient. Rowe and Angle (1967a,b) conducted experiments of flow mixing between rod bundle fuel element flow channels during boiling. The turbulent diffusion of mass in circular pipe is studied by Puri et al. (1983). They have compared three different distributions for radial mass diffusivity with experimental data. The turbulent heat transfer characteristics in compound channels with

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Nomenclature

A	area of subchannel, m^2
C	loss coefficient for transverse momentum,
c	coefficient in mixing correlation.
d	hydraulic diameter, m
f_T, f_D	correction factors of turbulent and diversion momentum respectively,
g_c	gravitational constant, kg-m/(N-s^2)
$\overline{G}_{ij}$	average of mass flux through sub-channel i and j, kg/(m^2s)
h	enthalpy of Sub-channel, J/kg
L	length of the fuel pin active region, m
m	mass flow rate, kg/s
N	number of subchannel
p	pitch of fuel rod array, m
q''	heat flux, W/m^2
Re	Reynolds number
s	gap between pins, m
T	temperature, °C
T^*	non-dimensional temperature,
V'	effective specific volume, m^3/kg
v	effective velocity, m/s
w	diversion cross flow between adjacent subchannels, kg/(m-s)
w'	turbulent cross flow, kg/(m-s)
x	quality of coolant in sub-channel,
z	axial distance along the pin, m

Greek symbols

β	mixing coefficient or mixing Stanton number
ϕ	two phase friction multiplier,
ρ	density of fluid, kg/m^3
θ	two phase mixing multiplier

Subscripts

avgo	average outlet
fmax	fuel miximum
fmin	fuel minimum
in	average inlet
i, j	subchannel identification number
smax	sub-channel maximum
smin	sub-channel minimum
SP	single phase
subo	sub-channel outlet
TP	two phase

gap are investigated by Hong (2011). The study was focused on the cause of pulsating flow by investigating the turbulent flow in compound channels through transient analysis. They found that the pulsating flow generates a strong turbulent flow mixing. Unsteady numerical simulations of the single-phase turbulent mixing between two channels connected by a narrow gap was carried out by Liu and Ishiwatari (2011). They studied the influence of strong anisotropic turbulent flow in the gap region and found that mixing rate enhanced significantly by periodic flow pulsation. Effect of gap width on turbulent mixing for parallel flow in a square channel with a cylindrical rod was carried by Chi Young Lee et al. (2013). They carried out experimental study on the effect of gap width on flow pulsation and turbulent mixing using PIV (Particle Image Velocimetry). They also suggested an empirical correlation of turbulent mixing coefficient.

However, in the practical application of these data in reactor fuel bundles, coolant mixing among sub-channels is a combined effect of diversion cross flow due to pressure gradient, turbulent mixing due to eddy diffusion, large scale flow pulsations and forced diversions due to spacer geometries, wire wraps or the shape of fuel pins. There are various uncertainties involved in manufacturing of fuel components, critical power correlations and measurements of parameters such as reactor power, reactor pressure, coolant flow through the assembly and coolant inlet temperature etc. which, in turn, affects the estimated thermal hydraulic safety margins and fuel temperatures. Due to these uncertainties, both the local conditions and critical heat flux will differ from their nominal values which, in turn, affects critical power of bundle. Apart from the nominal core thermal hydraulic analysis, these uncertainties are accounted in the hot-spot and hot channel analysis. At the design and operating stage, large numbers of thermal hydraulic analysis are required to be carried out for different operating regimes of the reactor core. Traditionally COBRA kind of sub-channel analysis is performed which involves preparation of inputs for different operational conditions of the core and power distribution. In the original COBRA codes, difficulties in convergence due to cross flow mixing and flow splitting existed and were also improved in the later versions (Masterson and Wolf, 1977). Further, in recent times, advanced sub-channel analysis solvers based on COBRA were developed for performance, speed and handling of large number of sub-channel (Burtak et al., 2006; Gluck, 2007). The large increase in the number of applications of sub-channel analysis codes on reactor safety, requires a continued development of software and improvements on sub-channel analysis for both user and convenience features as well as analysis options (Swindlehust, 1995). These types of development of sub-channel analysis utility software helps in, fuel assembly design optimization, fuel cycle design optimization studies and core power level uprating studies. These improvements are aimed to reduce cost and improved quality, repeatability of analysis for both the vendor as well as regulatory purpose and overall improvement of safety of nuclear reactor core (Swindlehust, 1995).

Other than sub-channel analysis codes, nowadays, various advanced simulation packages are being written, like 'Fuel Rod Analysis Toolbox' for reducing the large amount of input data preparation and simplify the setup of input for complex data sets and reducing the analysis time (Lassmann et al., 2015). The efforts on coupled fine mesh neutronics and thermal hydraulics of PWR fuel assemblies with pin cell and sub-channel level calculations were modeled and implemented in a software framework (Jareteg et al., 2015) are available in literature. Also, a domain specific analysis system for examining nuclear reactor simulation data for light water and sodium cooled fast reactor were built to handle the large thermal hydraulic analysis of nuclear reactor core as well as coupling to the reactor system levels codes (Billings et al., 2015) were taken up considering the complexity of core thermal hydraulic analysis and hence the safety of the nuclear reactor.

From the above literature review, it is clear that many studies are focused on in the field of inter-channel mixing in turbulent flows in rod bundles. These research work indicate that the cross flow mixing is a key parameter and plays a vital role in the accurate prediction of flow distribution in a rod bundle subchannels. The computation of fuel and rod surface temperatures, critical heat flux and critical power of rod bundles are essential to estimate the limits on safe operating power of core under different states of reactor. The estimation of safety margins are usually carried out using sub-channel analysis codes. During the design and optimization phases, a large number of cases are to be analyzed. It takes considerable amount of time in using the bench-marked codes for analyzing all the fuel assemblies for entire range of operating power and also at different flow regimes and transients to ensure the safety of reactor. In recent past many software tools are being developed for the analysis of core. These frameworks are to be