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Technical Note

Enhancement of COBRA-EN capability for VVER reactors calculations

M. Aghaie *, A. Zolfaghari, M. Minuchehr, A. Norouzi

Engineering Department, Shahid Beheshti University, GC, P.O. Box 1983963113, Tehran, Iran

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ABSTRACT

Thermal–hydraulic subchannel treatment of a typical hexagonal fuel assembly of VVER nuclear reactor core in steady-state or transient conditions needs consideration of detail geometry of fuel rods and subchannels. The COBRA-EN code could not generate the hexagonal subchannel and fuel-subchannel connections automatically and user must enter all required data for neighbor of each subchannel manually. Furthermore, treatment of the flow and heat transfer equations in the code needs to consider exact linear fission heat flux throughout the core and then one require assigning the heat flux manually for every discretized cell in the domain of problem. Therefore, user must prepare exact geometry of hexagonal fuel assembly which is time consuming. To improve the code capability and above adversity, the automatic hexagonal subchannel generation and power distribution schemes are developed and implemented within COBRA-EN source subroutines.

1. Introduction

The design of light water reactor cores requires accurate prediction of the peak temperatures of the rods and coolant to ensure that certain safety considerations will be met. To achieve a safe and economical design, it is necessary to use reasonably conservative limits. Many of these depend to fuel, cladding and coolant outlet temperature in the steady state or transient conditions.

A typical light water core is comprised of several thousands of fuel pins clustered in groups of fuel assemblies. Each fuel assembly consists of several hundred pins. A complete thermal–hydraulic analysis requires the knowledge of coolant distributions and pressure losses throughout the core. Heat transfer and flow field in the rod bundles are complex phenomena and the basic understanding of these phenomena is essential to achieving optimum design performance during steady state operation and keeping structural integrity during off-normal operations. In order to ensure that the design bases are satisfied, in past years much efforts has been made to develop bundle thermal hydraulic subchannel treatment codes that yield detailed coolant temperatures for all the subchannels in the bundle.

Furthermore, the water density reduction associated with the heat-up results in a pronounced coupling between neutronic and thermal–hydraulic analyses, which takes into account the strong natural influence of the incore distribution of power generation and water properties. The power density gradient within the fuel assemblies, together with the strong dependence of water properties on the temperature, and feed back effects on neutronic cross sections require us to consider a fine spatial resolution in which the individual fuel pins are resolved to provide precise evaluation of the clad, and coolant temperature, which are currently considered as one of the crucial design criteria. These goals have been achieved considering an advanced analysis method based on the usage of existing codes which have been coupled together. In the past years, there were many researches for coupling the neutronic with thermal–hydraulic codes. The WIMS–CITATION and COBRA-EN codes were coupled (Safaei Arshi et al., 2010) to investigate thermal–hydraulic analysis of VVER-1000 core with core analysis mode of the COBRA-EN.

Currently, COBRA-EN code (Basile et al., 1999) is used as one of the subchannel analysis code throughout the world. This code is an updated version of the COBRAIIIC (Rowe, 1973) and COBRA-IV-I (Wheeler et al., 1976) codes. Most of the subchannel codes currently used in the design of reactor cores were developed long ago and their applicable ranges and modeling are limited. Therefore, requests for the development of a subchannel code for improved estimate designs are raised based on various experimental results that have been recorded to date.

The hexagonal subchannels are characteristic of VVER reactors. It is hard and time consuming to define hexagonal subchannels and power distribution manually in COBRA-EN input deck. To relax these drawbacks, in this work capability of COBRA-EN code is enhanced by developing source code to handle VVER fuel rods configuration and accept arbitrary power distribution function easily.

* Corresponding author. Fax: +98 21 29902546.
E-mail address: M_Aghaie@sbu.ac.ir (M. Aghaie).

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2. Thermal–hydraulic flow field control volumes in COBRA-EN

The COBRA-EN thermal–hydraulic analysis is carried out in an array of parallel channels delimited by cylindrical fuel rods and open gaps (Basile et al., 1999). The axial direction (z axis) is assumed parallel to the channels and oriented from the flow inlet to outlet. To approximate the flow differential equations, the channels are divided into axial intervals by planes normal to the z axis and not necessarily equispaced. The volumes bounded by axial planes and channel lateral borders make up the three-dimensional grid of computational cells (control volumes) for mass, energy, axial and lateral momentum balance equations (Francisco et al., 2005). Figs. 1 and 2 illustrate lateral and top view of control volume for mass, energy, and momentum balance equations respectively. The parameters m, P, T, α, h, w and q'' are mass flow, pressure, temperature, void fraction, enthalpy, cross flow and linear heat flux in the control volume respectively. The notations i and j are used for axial and radial channel level indexes. Obtaining thermal–hydraulic parameters such cross flow, axial–lateral pressure gradient and fuel rod temperature in each control volume, one needs to couple continuity, energy and pressure equations.

3. Automatic hexagonal subchannel generation

In COBRA-EN the CHAN subroutine reads the configuration of the fuel rods and subchannels from input deck, and generates the necessary data for the next steps. The CHAN generates coupling parameters from the data which are defined in the input file of COBRA-EN. In card 6 of input file, the configuration parameters are defined for square subchannels and then the code generate automatically subchannel connections but in card 7 subchannel connection addressed manually.

In subchannel analysis, GENDAT subroutine of COBRA-EN is only responsible for automatic generation of the coupling variables for PWR fuel assembly types, square subchannels, as shown in Fig. 3.

In this work, an adapted algorithm for automatic generation of coupling parameters for arbitrary hexagonal fuel rod configurations, VVER fuel assembly types, is developed. Hexagonal fuel assemblies are used in VVER reactors; subchannels in a hexagonal fuel assembly are classified into three types. Fig. 4 shows a typical hexagonal rod bundle consists of several types of rods and subchannels. These consist of central, edge, and corner subchannels and rods (Todreas and Kazimi, 1990).

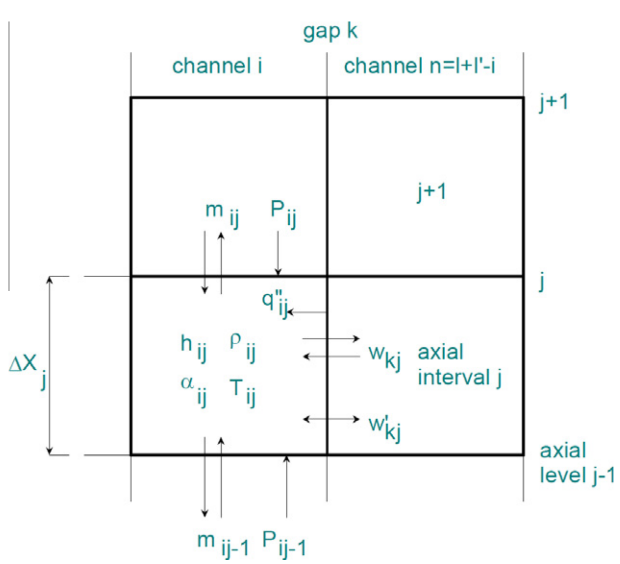

Fig. 1. Control volume for mass, energy and axial momentum balance finite-difference equations (lateral view).

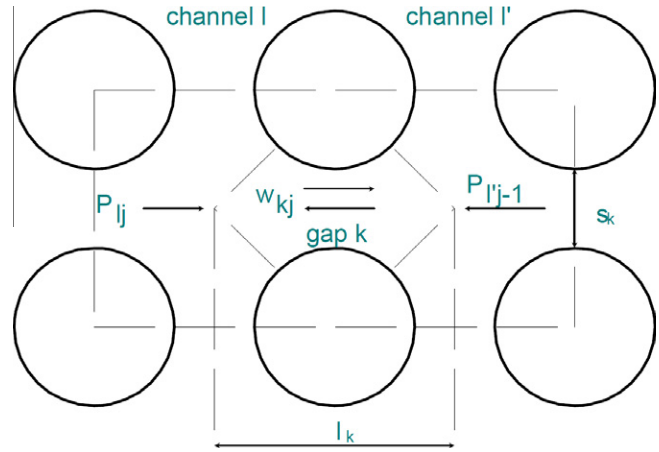

Fig. 2. Control volume for lateral momentum balance finite-difference equation (top view).

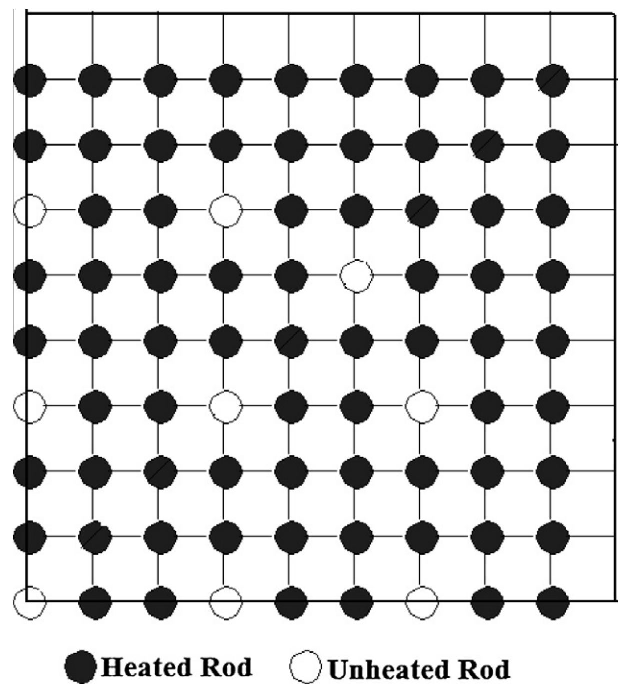

Fig. 3. PWR fuel rod, square configuration (1/4 fuel assembly top view).

Implementing the presented development and definition of the necessary parameters in card 6, one may eliminate using the complicated card 7.

In the first step for implementing automatic hexagonal subchannel generation, one needs to specify $NRod_{min}$, $NRod_{max}$, p, and D which are the minimum and maximum number of fuel rods in arrays of an assembly, fuel rods pitch and rods diameter in a desired hexagonal fuel assembly respectively; for instance in a typical fuel assembly which is shown in Fig. 5 $NRod_{min} = 3$, $NRod_{max} = 5$ and the total rows, total rods and total subchannels are obtained by:

$$\text{Total Rows} = 2(NRod_{max} - NRod_{min}) \tag{1}$$

$$\text{Total Rods} = NRod_{max} + 2 \sum_{i=0}^{\frac{\text{Total NRow}}{2}-1} (NRod_{min} + i) \tag{2}$$