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Modified COBRA-EN code to investigate thermal-hydraulic analysis of the Iranian VVER-1000 core

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A B S T R A C T

Thermal-hydraulic analysis of a typical VVER-1000 core at steady-state condition, using COBRA-EN code, is presented herein. Required power distribution was computed by the WIMS-D4 and CITATION codes based on the neutronic calculations. Maximum and average fuel temperature, enthalpy, void fraction, coolant temperature and density, coolant mass flow rate and pressure drop are calculated using EPRI model. Thermal-hydraulic calculations of the most rated channel which is determined based on neutronic calculations results in temperature, enthalpy, critical heat flux and minimum DNBR (MDNBR) of the core hottest channel are investigated. The COBRA-EN code is modified in order to make a thermal-hydraulic analysis for the VVER reactor, and this is the main objective of the present article. Our results are compared with analytical approaches and the reactor FSAR.

1. Introduction

An accurate prediction of thermal hydraulic performance of a nuclear reactor is a major concept in its design for both economic and safety reasons. Thermal-hydraulic analysis includes heat transfer and the hydrodynamic characteristics of the reactor core: the distribution of coolant parameters and temperature fields and conditions for crisis occurrence of heat transfer both in steady and transition conditions, related to reactor startup and shutdown. This article provides the result of thermal-hydraulic analysis of the typical VVER-1000 reactor core at steady-state condition. In recent years many digital computer programs have been written to solve the set of fluid conservation equations which characterize the steady state and/or transient thermal hydraulic performance of nuclear reactors. In this study, the COBRA-EN computer code (Basile et al., 1999) is used as the main thermal hydraulic code for our calculations. The analytical approach is performed for verification of the results of modified COBRA-EN code. The required power distribution is computed by the WIMS-D4 (United Kingdom Atomic Energy Authority, 1998) and CITATION (Fowler et al., 1971) codes which are determined based on the neutronic calculations. Maximum and average fuel temperature, enthalpy, void fraction, coolant temperature and density, coolant mass flow rate and pressure drop are calculated using EPRI model (Todreas and Kazimi, 1990). Thermal-hydraulic calculations of the most rated channel, which was determined based on the neutronic calculation, is carried out and therefore result in temperature, enthalpy, critical heat flux (CHF) and minimum DNBR (MDNBR) of the hottest channel are obtained. The COBRA-EN code is modified in order to make a thermal-hydraulic analysis of a VVER nuclear reactor, to be capable for modeling the annular fuel rods of this type of reactor.

2. Material and methods

2.1. COBRA-EN code

COBRA-EN code (Basile et al., 1999) is an upgraded version of the COBRA-3C (Rowe, 1973) and COBRA-IV-I (Wheeler et al., 1976) codes for thermal-hydraulic analysis of reactor cores such as PWRs or BWRs. COBRA-EN code is as the thermal-hydraulic module for core kinetics and long-term reactivity simulators that use nodal coarse-mesh approximations to the neutron diffusion equations, and allows two kinds of analysis to be performed, i.e., "core analysis" and "sub-channel analysis". The former allows the analysis of an assembly of open or separated coolant channels each containing a bundle of fuel rods and represented by lumped thermal-hydraulic parameters and by an average fuel pin and the latter is the analysis of an array of

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Nomenclature

Symbol	Description (Unit)
q	power of the hot rod (W)
q'	linear power (W/m)
W	mass flow rate (kg/s)
C_p	specific heat at constant pressure (J/kg °C)
T_{bi}	bulk coolant temperature at the channel inlet (°C)
T_{bo}	bulk coolant temperature at the channel outlet (°C)
T_{ci}	clad inside surface temperature (°C)
T_{co}	clad outside surface temperature (°C)
T_b	bulk coolant temperature (°C)
T_{fi}	fuel inner surface temperature (°C)
T_{fo}	fuel outer surface temperature (°C)
h_{gap}	effective gap conductance (W/m^2 °C)
h	single phase heat transfer coefficient (W/m^2 °C)
K_c	clad thermal conductivity (W/m °C)
K_f	fuel thermal conductivity (W/m °C)
R_g	mean radius in the gap (m)
R_v	internal cavity radius (m)
R_{fo}	fuel pellet radius (m)
R_{ci}	clad inside radius (m)
R_{co}	clad outside radius (m)
Re	Reynolds number at average coolant temperature and hydraulic diameter
Pr	Prandtl number at average coolant temperature
$Nu_{c.t}$	Nusselt number for a circular tube (hD/K_{fluid})
P	array pitch (m)
D	rod diameter (m)
$B.A$	boric acid
h_{in}	inlet enthalpy (Btu/lb_m)
h_{fg}	vaporization enthalpy (Btu/lb_m)
P_r	critical pressure ratio (=system reference pressure/critical pressure)
G	coolant mass flux (lb_m/s ft^2)
X_{in}	inlet flowing vapor quality
q''	local heat flux (Btu/s ft^2)

individual fuel rods which partition the coolant flow area into small sub-channels. A “channel” can represent an individual fuel rod assembly or a half- or a quarter-fuel assembly, or even a cluster of fuel assemblies (and the implied coolant channels) and also an associated number of equal fuel rods which are assigned to an equal share of the specified power input to the channel. Starting from a steady-state condition in an LWR core or fuel element, COBRA-EN code allows simulating the thermal-hydraulic transient response to user-supplied changes of the total power of the outlet pressure and of the inlet enthalpy and mass flow rate.

2.1.1. Code modification

In order to make use of COBRA-EN code for thermal-hydraulic analysis of a VVER-1000 nuclear reactor, which contains hollow fuel pellets, the subroutine “TEMP” has been modified to be capable of thermal-hydraulic analysis of this fuel type. For heat flow out of the inner surface of a hollow fuel element the following relations are applied (El-Wakil, 1993):

$$\frac{dT}{dr} = 0 \quad \text{at} \quad r = r_i \tag{1}$$

where r is the radius of the rod and r_i is the radius of the fuel pellet hole. In this modification, instead of the rod center, the adiabatic or symmetry boundary condition is applied to the inner surface of the pellet and the first radial node in fuel rod heat transfer model is assumed on the inner surface of the fuel pellet.

2.2. Reactor description and modeling

VVER-1000 reactor is a Russian-type pressurized water reactor. The major difference between the VVER and a Western PWR, in the present study, is the fuel assembly design and the core geometry. The specifications of reactor which are studied in this article are presented in Table 1. Specifically, the reactor under study is made up of 163 hexagonal fuel assemblies of three different enrichments, i.e., 1.6%, 2.4% & 3.6%. Fig. 1 shows the fuel assemblies arrangement in the core which is used for “core thermal-hydraulic analysis” using COBRA-EN code. In core analysis the individual sub-channels are lumped together to give an equivalent flow area and the fuel rods are modeled using a single rod to represent the average behavior of all rods in each channel. In this analysis, one sixth of VVER-1000 core has been modeled. The channels are axially divided into 10 equal intervals and the fuel pellets are divided into 5 radial intervals. Three coupled differential equation is solved by COBRA-EN in which the conservation equations for mass, energy, axial and transverse momentum conservation can be solved. A thermal-hydraulic grid of 280 control volume, each assembly-size of 23.6 cm hexagonal horizontal cross-section and axial height of 35.5 cm is considered.

The required linear power distribution was computed by the WIMS-D4 (United Kingdom Atomic Energy Authority, 1998) and CITATION (Fowler et al., 1971) codes, using the procedures applied in neutronic modeling of Bushehr nuclear reactor, a VVER-1000 Russian reactor, by Faghihi et al. (2007) (Faghihi and Mirvakili, 2009). According to neutronic calculation of this reactor, radial and axial power distribution for each fuel assembly of the core, radial and axial power peaking factors (c.f., Fig. 2) and also the hot channel, where maximum power density occurs, are determined. Thermal-hydraulic calculations of this channel contains temperature, enthalpy and critical heat flux (CHF) and also the minimum DNBR (MDNBR) of the hottest channel of the core. In this simulation a full boiling curve comprising five heat-transfer regimes, i.e., single-phase forced-on, sub-cooled nucleate boiling, saturated nucleate boiling, transition and film boiling (or post-CHF boiling) is considered. In this case, the following correlations are applied:

- Weisman correlation (El-Wakil, 1993) for heat transfer coefficient in single-phase forced convection.

Table 1
Reactor specifications.

Reactor core operating conditions	Value
Reference pressure (MPa)	15.7
Reactor thermal power (MWt)	3120
Inlet coolant flow rate (m^3/h)	84 800
Inlet coolant enthalpy (kJ/kg)	1290
Coolant temperature at the core inlet (K)	564.15
Fuel assembly	
Fuel assembly form	Hexagonal
Number of fuel assembly in the core	163
Pitch between the assemblies	23.6
Number of fuel rod in the fuel assembly	311
Fuel rod	
Hole diameter in the fuel pellet	1.5 mm
Fuel pellet outside diameter	7.57mm
Cladding outside diameter	9.1 mm
Fuel pellet material	UO_2
Cladding material	Alloy Zr + 1% Nb
Fuel rod pitch	12.75mm