Annals of Nuclear Energy 50 (2012) 263–267

Contents lists available at SciVerse ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Technical Note

Optimizing a gap conductance model applicable to VVER-1000 thermal–hydraulic model

M. Rahgoshay *, M. Hashemi-Tilehnoee

Department of Nuclear Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran

ARTICLE INFO

Article history:
Received 20 June 2012
Received in revised form 12 August 2012
Accepted 15 August 2012
Available online 6 October 2012

Keywords:
Gap conductance model
CALZA-BINI
RELAP5
COBRA-EN
VVER-1000

ABSTRACT

The modeling of gap conductance for application in VVER-1000 thermal–hydraulic codes is addressed. Two known models, namely CALZA-BINI and RELAP5 gap conductance models, are examined. By externally linking of gap conductance models and COBRA-EN thermal hydraulic code, the acceptable range of each model is specified. The result of each gap conductance model versus linear heat rate has been compared with FSAR data. A linear heat rate of about 9 kW/m is the boundary for optimization process. Since each gap conductance model has its advantages and limitation, the optimized gap conductance model can predict the gap conductance better than each of the two other models individually.

1. Introduction

Nuclear industry and licensing authorities need to rely on the good performance of methods and computer programs. An accurate prediction of thermal hydraulic performance of a nuclear reactor is crucial in its design and operation for both economic and safety reasons. This justifies the recent significant exchange of information between western and eastern countries in the nuclear technology area. Due to the specific design of VVER-1000 fuel pins and fuel assemblies, some attempts were made to verify neutronics and thermal–hydraulics codes for this type of reactor. VVER-1000 has been theoretically investigated in relation to some accidents at various levels in its first and second loop and core. Utilizing codes like COSTANZA, WIMS, CITATION and RELAP, researchers have modeled group-10 control rod scram, control rod ejection accident and loss of heat sink transients in the secondary circuit of VVER-1000 (Rahgoshay and Rahmani, 2007; Tabadar et al., 2012; Abbasi and Hadad, 2012). However, some code modifications are proposed for this type of reactor. Safaei Arshi et al. (2010) have modified COBRA-EN code to investigate thermal–hydraulic analysis of the Iranian VVER-1000 core. The main changes were applied to the subroutine "TEMP" to enable the code to analyze the hollow fuel type. Since the results were consistent with FSAR, the authors recommended this modified version of COBRA-EN code for further studies in sub-channel analysis and as a computational tool for transient analysis of the VVERs. In a new research, the automatic hexagonal sub-channel generation and power distribution schemes are developed and implemented within COBRA-EN source subroutines (Aghaie et al., 2012). The authors proposed that their pre-program reduce huge time used for input preparation of VVER-1000 fuel assemblies. However, not-licensed users need the source code of COBRA-EN for these modifications. In this study, the COBRA-EN thermal–hydraulic capability has been improved without source modification, and the modification has focused on the gap conductance model.

2. Materials and methods

2.1. COBRA-EN

COBRA-EN code (Basile et al., 1999) is an upgraded version of the COBRA-3C and COBRA-IV-I code for thermal–hydraulic analysis of reactor cores such as PWRs or BWRs. COBRA-EN code acts as the thermal–hydraulic module for core kinetics and long-term reactivity simulators that use nodal coarse mesh approximations in the neutron diffusion equations, and allow 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 represented by lumped thermal–hydraulic parameters and an average fuel pin but the latter is the analysis of an array of individual fuel rods, which divide the coolant flow area into small sub-channels. A "channel" can represent an individual fuel assembly (FA) 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

* Corresponding author. Tel.: +98 912 821 9564; fax: +98 21 44868421.
E-mail address: m.rahgoshay@gmail.com (M. Rahgoshay).

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http://dx.doi.org/10.1016/j.anucene.2012.08.011

Nomenclature

C	constant in contacted gap conductance formula
H	Meyer's hardness number of the softer material
h_g	total gap heat transfer coefficient
$h_{contact}$	contacted gap heat transfer coefficient
$h_{g,open}$	open gap heat transfer coefficient
k	thermal conductivity
k_{gas}	thermal conductivity of the gas
L_{ActF}	active height of fuel rod
N_{FA}	number of fuel assemblies in the core
N_{FR}	number of fuel rods in the core
P_i	surface contact pressure
P_{th}	core thermal power
PPF	power peaking factor
$\dot{q}$	linear heat rate
Ro	surface roughness
RP	relative power
$\dot{q}'$	cladding inner surface temperature
T_{fo}	fuel surface temperature
t_o	as-fabricated fuel-cladding gap width

Greek symbols

δ_{eff}	effective gap width
δ_g	mean thickness of the gas space
δ_{jump}	temperature jump distance
ε	surface emissivity of the fuel
σ	Stefan–Boltzmann constant

Subscript

c	clad
f	fuel

Abbreviations

BWR	boiling water reactor
FA	fuel assembly
FSAR	final safety analysis report
LB-LOCA	large break loss of coolant accident
LWR	light water reactor
NPP	nuclear power plant
PWR	pressurized water reactor
RELAP	reactor excursion and leak analysis program
UO_2	uranium dioxide
VVER	vodno-vodyanoi energetichesky reactor

equal fuel rods, which are assigned to an equal share of the specified power input to the channel. COBRA-EN uses the VIPRE code gap conductance model. The gap between the fuel pellet outside surface and cladding inside surface considers two heat-transfer mechanisms; thermal radiation and conduction in the fill gas.

The required thermal properties (emissivity of the clad and fuel pellet surfaces, conductivities of the fill gases, generally as a function of temperature) have been taken from MATPRO code. Due to the physical contact between fuel pellets and clad, however, neither a fuel rod deformation model nor the related conduction is included.

Thus, the gap conductance model should be used with caution or, at least, with values of the clad thickness and pellet diameter, which is deemed to be representative of the conditions being dealt with (excluding, in any case, physical contact). Otherwise, an input constant value is preferred (Basile et al., 1999).

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 is the fuel assembly design and the core geometry. The characteristics of the studied reactor are presented in Table 1.

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 study, one-sixth of VVER-1000 is modeled. The required radial power peaking factor (PPF) is shown in Fig. 1 (FSAR, 2005).

Since radial and axial power profile affect gap conductance, the linear heat rate of each fuel assembly in axial intervals can be obtained as:

$$\dot{q}' = \frac{P_{th}}{N_{FA}N_{FR}L_{ActF}(\mathrm{PPF} \cdot RP)} \tag{1}$$

Fig. 2 shows the graph of relative power; thermal power in each axial node to nominal thermal power of the core.

2.3. Gap conductance model

The gap is supposed to consist of an annular space occupied by gases. The gas composition is initially the fill gas, which should be an inert gas such as helium, but is gradually altered with burn-up by adding gaseous fission product such as xenon and krypton. However, this simple picture does not reflect the real conditions of the fuel pin after some irradiation. The fuel pellets usually crack upon irradiation. For typical LWR fuel rods, the resistance of the UO_2 fuel is by far the largest. The next largest resistance is that of the gap. Therefore, models for the gap conductance with increasing sophistication have been developed over the years (Todreas and Kazimi, 1999).

In addition, thermal expansions of the fuel and cladding are often different resulting in a substantial pellet cladding contact at the

Table 1
Reactor specifications.

Reactor core and components characteristics	Value
Operation conditions	
Reference pressure (MPa)	15.7
Reactor thermal power (MWt)	3120
Inlet coolant flow rate (m^3/h)	84800
Inlet coolant enthalpy (kJ/kg)	1290
Coolant temperature at the core inlet (°K)	564.15
Coolant temperature at the core outlet (°K)	594.15
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
Fresh fuel assembly enrichment	1.6%, 2.4%, 3.6%
Fuel rod	
Hole diameter in the fuel pellet (mm)	1.5
Fuel pellet outside diameter (mm)	7.57
Cladding inside diameter (mm)	7.73
Cladding outside diameter (mm)	9.1
Fuel rod pitch (mm)	12.75
Fuel pellet material	UO_2
Cladding material	Alloy Zr + 1% Nb