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Thermal-Hydraulics analysis of pressurized water reactor core by using single heated channel model

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Abstract

Thermal hydraulics of nuclear reactor as a basis of reactor safety has a very important role in reactor design and control. The thermal-hydraulic analysis provides input data to the reactor-physics analysis, whereas the latter gives information about the distribution of heat sources, which is needed to perform the thermal-hydraulic analysis. In this study single heated channel model as a very fast model for predicting thermal hydraulics behavior of pressurized water reactor core has been developed. For verifying the results of this model, we used RELAP5 code as US nuclear regulatory approved thermal hydraulics code. The results of developed single heated channel model have been checked with RELAP5 results for WWER-1000. This comparison shows the capability of single heated channel model for predicting thermal hydraulics behavior of reactor core.

Keywords: Nuclear Reactor, Thermal hydraulics, RELAP5, Single Heated Channel Model.

1. Introduction

The heat must be removed from the reactor core structure in the same rate as it is generated to avoid the core damage. Usually the cooling of the reactor core is provided by forcing a working fluid – so called coolant – through it. The heat accumulated in the coolant is then used for various goals, according to the purpose of the nuclear reactor. In nuclear power reactors, the heat is transformed into electricity using the standard steam cycles. In nuclear propulsion reactors the heat is used to create the thrust. Whatever the purpose of the reactor is, various aspects of heat transfer and fluid flow are present. The branch of nuclear engineering which is dealing with these aspects is called the nuclear reactor thermal-hydraulics.

One of the major objectives of the reactor thermal-hydraulics is to predict the temperature distributions in various parts of the reactor. The most important part of the nuclear reactor is the reactor core, where heat is released and the highest temperatures are present. Such temperatures must be predicted for various reactor operation conditions. To guarantee a safe reactor operation it is necessary that the temperatures are below specific limit values for various construction and fuel materials.

Another important objective of the reactor thermal-hydraulics is to predict forces exerted by the flowing coolant onto internal structures of the reactor. Too high or persistent oscillatory forces may cause mechanical failures of the structures. Thus, the thermal-hydraulic analysis provides information about the mechanical loads in the nuclear reactor, which in turn is used in the structure-mechanics analysis to investigate the system integrity. Very often such analyses have to include the effect of temperature distributions as well, especially when thermal stresses are significant.

The thermal-hydraulic analysis provides input data (the moderator density and other nuclear data not specified here) to the reactor-physics analysis, whereas the latter gives information about the distribution of heat sources, which is needed to perform the thermal-hydraulic analysis. The strong coupling between the two types of analyses causes that iterative approaches have to be used. There is also a coupling between the thermal-hydraulic and structure-mechanics analyses; however, it is less strong. The mechanical and thermal loads obtained from the thermal-hydraulics analysis are used to predict the structure displacements and thus the actual geometry under consideration. Usually the displacements are small and can be neglected. Thus, the design geometry can be used while performing the thermal hydraulics analysis.

According to the importance of thermal hydraulics calculations, several codes likes COBRA and RELAP have been developed [1]. Most of reactor thermal hydraulic calculation performed by these codes [2-5], but in reactor control and dynamic models we need very fast thermal hydraulic model with enough accuracy to be coupled with other calculations like neutronic and control parameters. In this study, we present single heated channel model as a very fast model that can be used in reactor control and dynamic models. The results of this model has been compared with RELAP code as an approved code by the US nuclear regulatory.

2. Material and methods

2.1 RELAP5

The Reactor Excursion and Leak Analysis Program (RELAP) is a tool for analysing loss of coolant accidents (LOCA) and systems transients in LWR. RELAP nuclear code has been developed at Idaho National Laboratory (INL) in the last 30 years. The USA Department of Energy (DOE) and the NRC, among other institutions, have been supporting its development. R5 (i.e. version 5 of RELAP) has been used for licensing and regulation, evaluation of operator guidelines and plant accident analysis. The code was developed to provide the first order effect needed for accurate prediction of system transients and to be cost effective enough to allow sensitivity and parametric studies. Although it was initially designed for nuclear applications, the last version can be used for nuclear and nonnuclear applications involving vapour, liquid, no condensable gases and non-volatile solutions.

R5 calculates the expected evolution of TH properties during steady-state or transients, including the interaction among system components (R5 uses an internal temperature calculation scheme based on input power). It implements a two fluid and six-equations model to simulate the two phases coolant behaviour, and consequently the system's TH behaviour. The R5 numerical solution is based on semi-implicit finite-difference. Although it was originally designed to model TH phenomena in 1D volume, it can also be used to simulate more complex systems using multiple channels geometry – for example, the NPP core can be simulated by up to a few hundred parallel nodes connected at several axial positions. However, the PK model inherent to R5 cannot consider multidimensional effects caused by spatial variations of feedback parameters.

2.2 Single Heated Channel Model

Temperature profile calculations of fuel assemblies and fuel rod channels in desired time step of fuel burnup have a very important role in finding the location of hot rod and hot channel and also accuracy of final results. For this purpose, we used single heated channel model that is fast and also relatively accurate [6]. Single heated channel model is a one dimensional heat conduction/convection model that is used for each channel. The hydraulic calculations of this model is based on the laws of conservations of mass (eq. (1)), momentum (eq. (2)), and energy (eq. (3)).

$$\frac{\partial \rho}{\partial t}+\nabla.(\rho\vec{v})=0 \tag{1}$$

$$\frac{\partial}{\partial t}\rho\vec{v}+\nabla.(\rho\vec{v}\vec{v})=\nabla p+\nabla.\overline{\overline{\tau}}+\rho\vec{f} \tag{2}$$

$$\frac{\partial}{\partial t}\rho h+\nabla.(\rho h\vec{v})=-\nabla.\dot{q}''+\dot{q}'''+\frac{\partial p}{\partial t}+\nabla.(p\vec{v})+\overline{\overline{\tau}}:\nabla\vec{v} \tag{3}$$

In single heated channel model for a single phase and a steady state condition, fluid pressure, enthalpy and temperature along the length of each channel, finally will be attained from eqs. (4-6).

$$p_{in} - p_{out} = \left(\frac{G_m^{\ 2}}{\rho_m^+}\right)_{out} - \left(\frac{G_m^{\ 2}}{\rho_m^+}\right)_{in} + \int_{z_{in}}^{z_{out}} \frac{fG_m\left|G_m\right|}{2D_e\rho_m}dz + \int_{z_{in}}^{z_{out}} g\rho_m dz \tag{4}$$

$$dh_m = \left[\frac{q''P_h}{\dot{m}} + \frac{1}{\rho_m}\left(\frac{\partial p}{\partial z} + \frac{fG_m\left|G_m\right|}{2D_e\rho_m}\right)\right]dz \tag{5}$$

$$T(z) = \frac{1}{Cp}\int_{z_{in}}^{z}\left[\frac{\dot{q}''P_h}{\dot{m}} + \frac{1}{\rho_m}\left(\frac{\partial p}{\partial z} + \frac{fG_m\left|G_m\right|}{2D_e\rho_m}\right)\right]dz + T_{in} \tag{6}$$

For convective heat transfer coefficient calculation we used DittuseBoelter correlation [6]. According to the heat convective and conduction equations, for calculation of the outer and inner temperature of clad and the outer temperature of fuel, eqs. (7-9) have been used.

$$T_{co}^{z} = \frac{q'(z)}{2\pi R_{co}^{hot} h_{cool}} + T_{cool}(z) \tag{7}$$

$$T_{ci}(z) = q^{'\circ}(z)\frac{1}{2\pi k_c} ln\frac{R_{co}^{hot}}{R_{ci}^{hot}} + T_{co}(z) \tag{8}$$

$$T_{Fo}(z) = \frac{q''}{h_g} + T_{ci}(z) \tag{9}$$

The gap conductance of the fuel-clad gap that is shown in eq. (10) which in this equation conductance of the gap is the sum of heat transfer across the open gap and heat transfer coefficient for gap closure (occurs because of fuel swelling and thermal expansion).

$$h_g = h_{g,open} + h_{g,closure} \tag{10}$$

The heat transfer coefficient for open gap is obtained:

$$h_{g,open} = \frac{k_{gas}}{\delta_{eff}} + \frac{\sigma}{\frac{1}{\varepsilon_f} + \frac{1}{\varepsilon_c} - 1}\frac{T_{Fo}^4 - T_{Fi}^4}{T_{Fo} - T_{Fi}} \tag{11}$$

According to eq. (12) effective gap width can be calculated where δ_g is gap width and δ_{jump} is found to be 10 mm [6].

$$\delta_{eff} = \delta_g + \delta_{jump} \tag{12}$$

The gas conductivity of a pure gas is given by eq. (13), where A is 15.8 and T is the gas temperature.

$$k(pure\,gas) = A\times 10^{-6}T^{0.79} \quad W/cm \tag{13}$$

According to the Calza-Bini calculations, the closure heat transfer coefficient is given by eq. (14) [7], where p_i is surface contact pressure and for zircaloy cladding, Meyer's hardness number (H) is 14×10^4.

$$h_{closure} = c\frac{2k_f k_c}{k_f + k_c}\frac{p_i}{H\sqrt{\delta_g}} \tag{14}$$

Finally the inner fuel temperature can be obtained as follows:

$$\mathrm{T}_f^{in} = \mathrm{T}_f^{out} + \frac{\dot{q}'''R_{fo}^2}{4\bar{k}_f}\left\{1 - \left(\frac{R_{fi}}{R_{fo}}\right)^2\left(1 + \ln\left(\frac{R_{fo}}{R_{fi}}\right)^2\right)\right\} \tag{15}$$

The flowchart of the developed model for T-H calculation that is used to calculate fuel, clad and coolant temperatures for each channel is shown in Fig. 1.

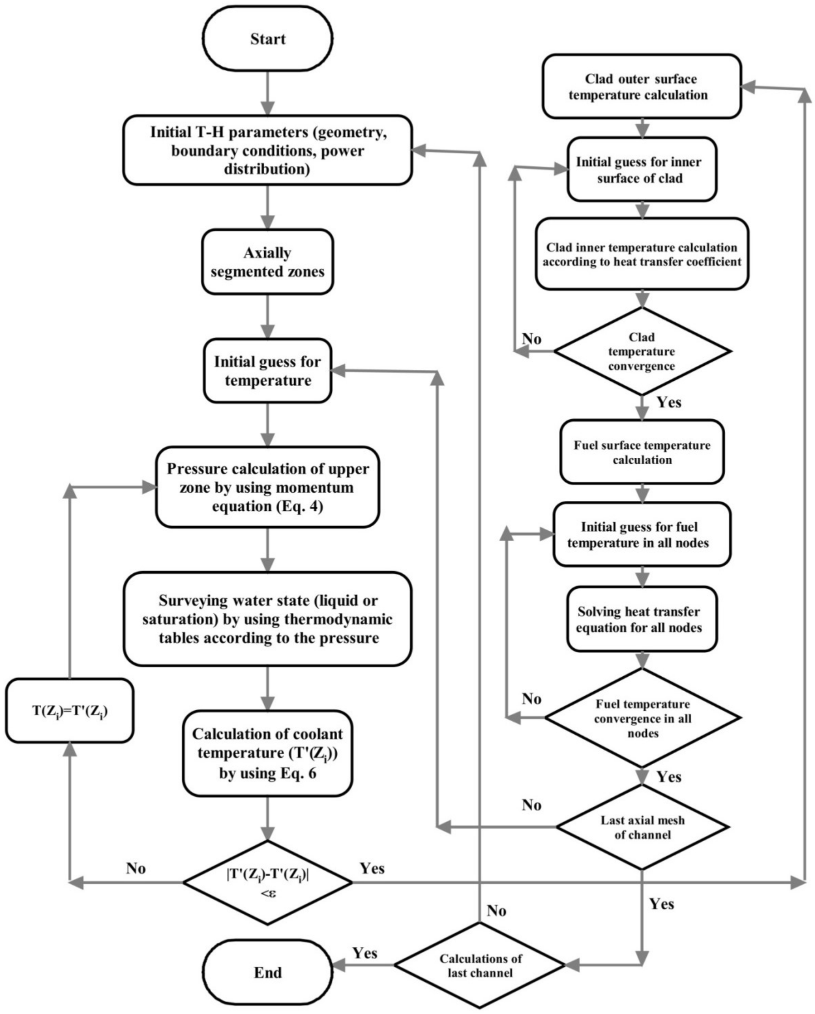

Fig.1. The flowchart of developed single heated channel model.

3. Description of Bushehr NPP

Bushehr NPP is one of the WWER-1000 (V-446) reactors that has a hexagonal configuration and 1/6 symmetric shape. This reactor has 163 hexagonal fuel assemblies of the same geometry and producing 3000 MWth at full power [7]. Moreover, each fuel assembly contains 311 fuel rods and 18 guiding channels for control rods or burnable poisons. Bushehr NPP full core model is given in Fig. 2.

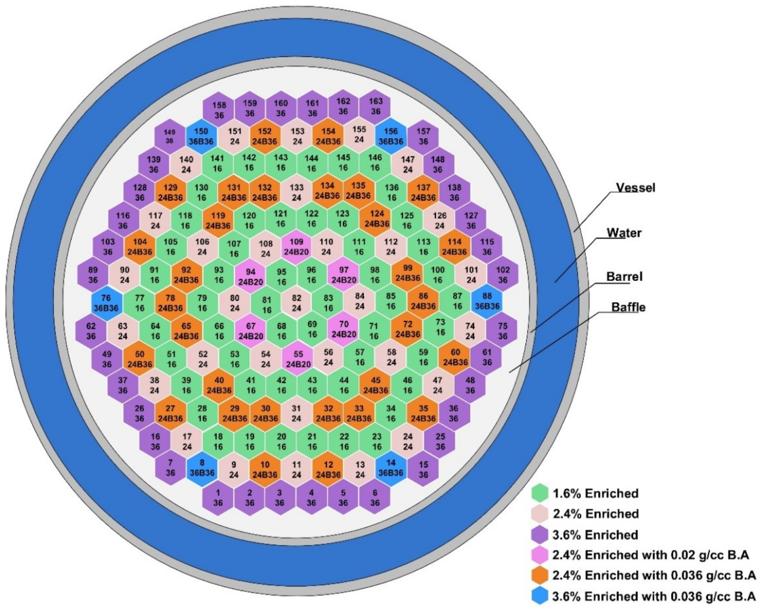

Fig. 2. Bushehr NPP reactor core arrangement.

Some specifications of Bushehr NPP that used in this simulation are given in Table 1.

Table 1. Bushehr NPP reactor core specifications.

Parameter	Value
Maximum heat power of reactor, 104% of nominal power (MW)	3120
Fraction of coolant leaks bypassing the core (%)	4.0
Coolant pressure at the core outlet (MPa)	15.7 ± 0.3
Coolant temperature at the reactor inlet (°C)	291 ± 2.5
Average coolant temperature at the reactor outlet (°C)	321 ± 5.0
Average linear heat rate (W/cm)	166.0
Maximum linear heat rate (W/cm)	448
Steam generator steam capacity (t/h)	1470
Fuel assembly form	Hexagonal
Arrangement of fuel rod	Triangle
Number of fuel assembly in the core	163
Fuel assembly pitch (mm)	236
Fuel rod pitch (mm)	12.75
Number of fuel rods in the FA (Pcs)	311
Hole diameter in the fuel pellet (mm)	1.5
Fuel pellet outside diameter (mm)	7.57
Cladding outside diameter (mm)	9.1
Cladding material	Alloy Zr + 1% Nb
Fuel pellet material	UO2
Fuel mass in the rod (kg)	1.575
Fuel rod effective height (cm)	353
Number of control rod	85 × 18
Control rod absorbing material	B4C + (Dy2O3_TiO2)
Number of burnable poison rod per bundle (Pcs)	18

4. Result

Initially for surveying the thermal-hydraulic model accuracy, the single heated channel model has been checked for WWER-1000 reactor core as a typical single heated with presented specifications in Table 1.

Fig. 3 shows a comparison between the coolant temperature profile versus core height attained from single heated channel model and RELAP5 code. There are excellent match between these two profiles and average relative difference between them is 0.0143%.

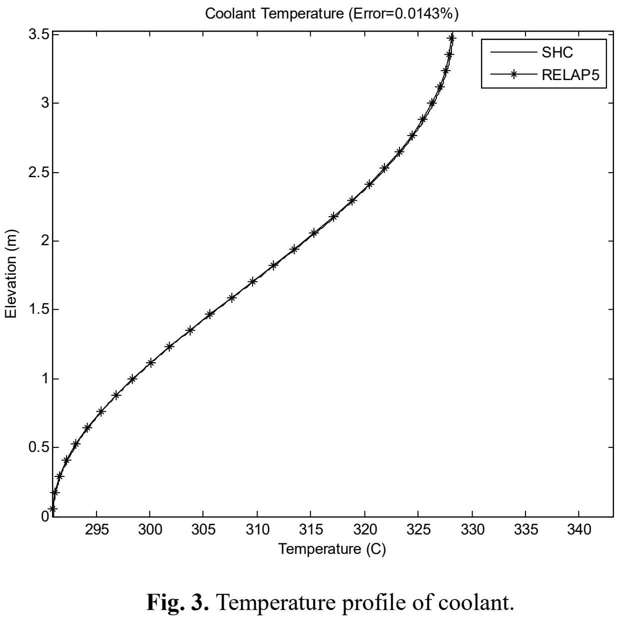

Fig. 3. Temperature profile of coolant.

Fig. 4 shows a comparison between the clad outside temperature versus core height attained from single heated channel model and RELAP5 code. There are proper match between these two profiles and average relative difference between them is 2.437%.

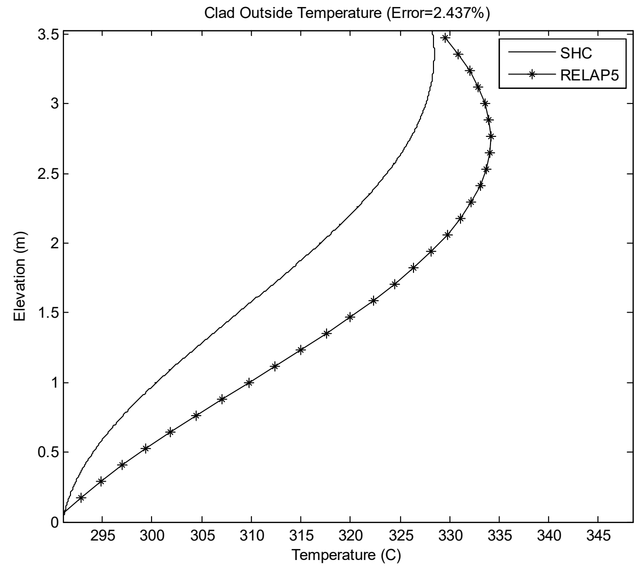

Fig. 4. Temperature profile of clad outside.

Fig. 5 shows a comparison between the coolant pressure profile versus core height of coolant attained from single heated channel model and RELAP5 code. There are good match between these two profiles and average relative difference between them is 0.99%.

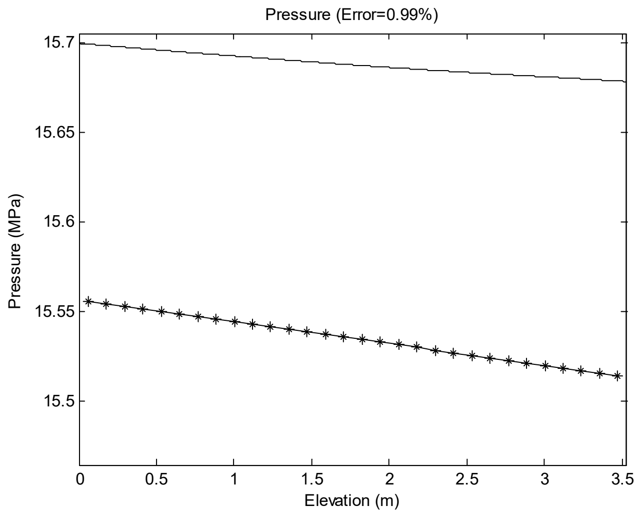

Fig. 5. Pressure profile of coolant.

5. Conclusion

One of the major objectives of the reactor thermal-hydraulics is to predict the temperature and pressure distributions in various parts of the reactor like coolant and clad. In this study single heated channel model has been presented to calculate these parameters. Also the reactor core has been modeled by RELAP5 code to compare with our model results. The comparison between the results have a proper match that shows the capability of single heated channel model for using in reactor core thermal hydraulic calculations.

Nomenclature

A_z	Cross sectional area of the channel;	R_{ci}	Clad inner radius;
C_p	Specific heat of the fluid;	R_{co}	Clad outer radius;

D_e	Hydraulic diameter;	R_{fi}	Inner radius of fuel;
f	Friction factor;	R_{fo}	Outer radius of fuel;
f_b	Body force;	T_{ci}	Inner temperature of clad;
g	Gravity;	T_{co}	Outer temperature of clad;
h	Enthalpy;	T_{fi}	Inner fuel temperature;
h_{cool}	Nusselt number;	T_{fo}	Outer fuel temperature;
$h_{g,closure}$	Heat transfer coefficient for gap closure;		
$h_{g,open}$	Heat transfer across the open gap;	δ_{eff}	Effective gap width;
k_f	Thermal conductivity coefficient of the fuel;	ε_c	Surface emissivity of the clad;
k_{gas}	Thermal conductivity of the gas;	ε_f	Surface emissivity of the fuel;
p	Pressure;	σ	StefaneBoltzmann constant;
p_h	Heated perimeter;	$\vec{v}$	Coolant velocity;
$\dot{q}'$	Linear heat source;	ρ	Coolant density; and
$\dot{q}''$	Wall heat flux;	τ	Shear stress.
$\dot{q}'''$	Volumetric heat source;		

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