

Sub-Channel Analysis in Hot Fuel Assembly's of VVER-1000 Reactor using Drift-Flux Model

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Abstract

The present work aims at verifying a sub-channel analysis program based on a drift-flux model by applying it to traditional VVER-1000 hot fuel assembly in steady-state and operational transients. Liquid enthalpy, mass velocity and liquid temperature calculated then compared with the final safety analysis report data. According to the results the code outputs are in good agreement with experimental data.

Keywords: Drift-Flux Model, Hot Fuel Assembly, Sub-Channel Analysis, VVER-1000

1. Introduction

As a result of primary-to-secondary leak there is a long-term loss of coolant beyond the boundaries of the containment. The consequence of this accident is radioactive release into the atmosphere and probable loss of reactor core cooling. In this research the reference power plant is Bushehr NPP site. Operational information from this NPP is available for the purpose of assessing how the thermal-hydraulic model compares against the plant data. At that place were several modeling by thermal-hydraulic codes done relate to VVER-1000 NPP up to today. Some of them are, Simulation of loss of flow transient in a VVER-1000 Nuclear Power Plant with RELAP5/Mod3.2¹, simulating a typical Steam Generator Tube Rupture (SGTR) on VVER-1000 reactor parameters², and modeling a control rod ejection accident in a VVER-1000/V446 by RELAP5/Mod3.2³. Sub-channel analysis consists of solution of mass, momentum and energy conservation equations written for elementary channels. Another important category of the thermal hydraulic codes beside the analysis method is the type of two-phase flow model. The principal types of flow models incorporated into thermal-hydraulic codes comprise the homogeneous mixture model, multi-fluid model, and

diffusion model. In some researches some VVER-1000 core design parameters are optimized⁴ or the radioactive material inventory was evaluated⁵ but a detail thermal-hydraulic analysis a two-phase model can be done on a fuel assembly region in normal operation and during a steam generator tube rupture accident due to loss of flow accidentally.

2. Description of VVER-1000 Reactor

VVER1000/V446, a Russian-type, pressurized water reactor has four coolant loops. The major divergence between the VVER and a Western PWR is the fuel assembly design and the core geometry. On each loop, it has a coolant pump and a horizontal steam generator. The plant has a standard emergency core cooling system, including high pressure pumps, low-pressure pumps and hydro-accumulators. The primary pressure maintenance system consists of pressurizer, surge line, and spray line. The reactor pressure vessel has four inlet and four outlet nozzles, where the outlet nozzles⁶. The characteristics of the studied reactor are presented in Table 1⁴.

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Table 1. Reactor specifications in normal condition

Reactor core and components characteristics	Value
Operation Condition	
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 (°C)	291.
Coolant temperature at the core outlet (°C)	321.
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

In this study, one-sixth of VVER-1000 fuel assembly is modeled. Figure 1 shows the core configuration and the geometry of one-sixth of hot fuel assembly which considered in modeling.

The core radial and axial Power Peaking Factors (PPF)

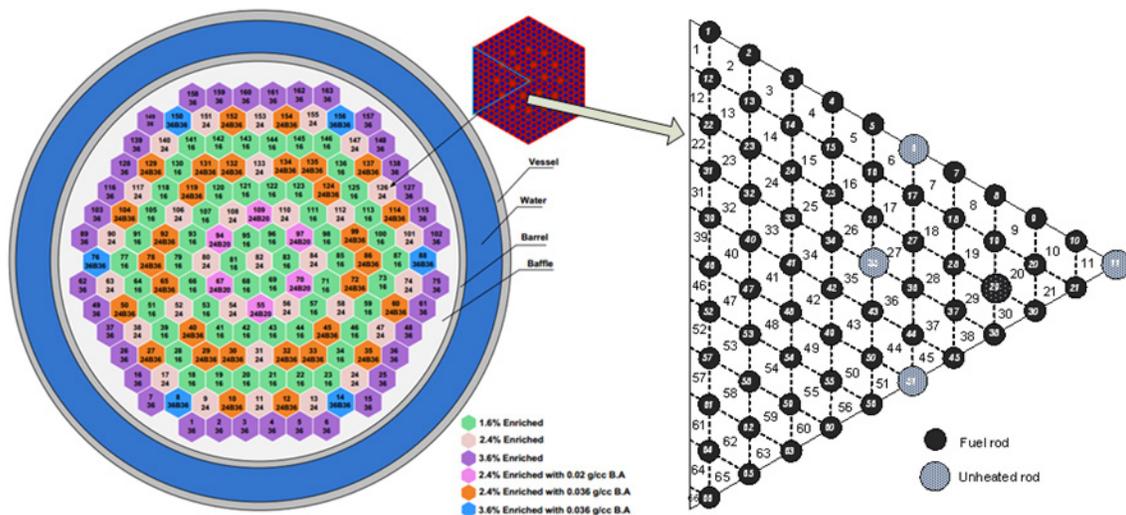


Figure 1. VVER-1000 core configuration and one-sixth of the modeled fuel assembly.

is presented in Figure 2 and Figure 3. It can be seen from Figure 2 that the hot fuel assembly radial PPF is 1.29⁴.

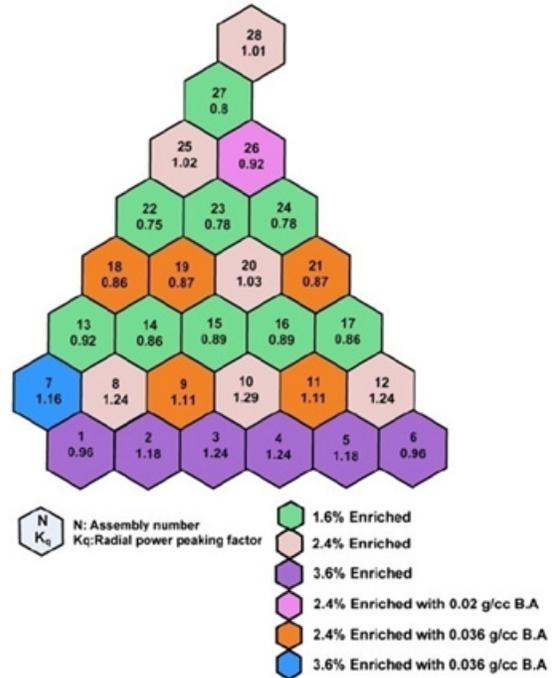


Figure 2. Assembly's radial power peaking factors in the core symmetry sector of 60^o.

It is noted that the fuel assembly has 13 grid spacers which their pressure loss coefficient considered about 0.09⁸. Figure 4 shows the position of the grid spacers.

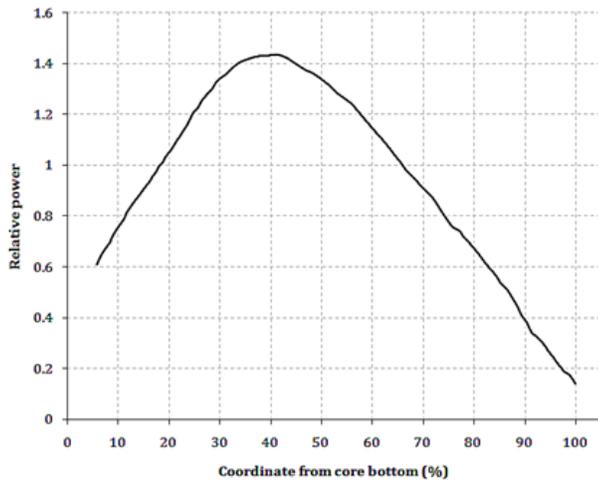


Figure 3. Relative axial power distribution over the height of fuel assembly⁷.

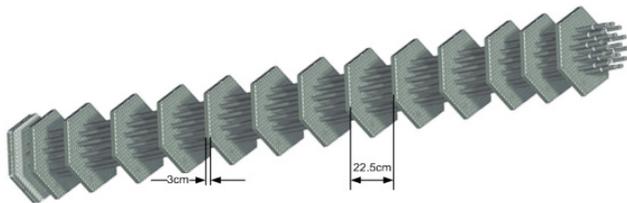


Figure 4. Grid spacer positions in the VVER-1000 fuel assembly⁷.

3. The Accident Scenario

The originating event is an instantaneous break with an equivalent diameter of 100 mm of cold collector of steam generator number 2 in the area of the lower course of heat exchanging tubes. The chronological sequence of transient events is presented in Table 2.

Table 2. Chronological sequence of transient events²

Description of Event	PSAR Value (S)
Steam generator 2 cold collector breaks (100.0 mm equivalent diameter)	0.0
Reactor cooling pump set trip of affected loop	9.75
Generation of radiation signal	10.0
Loss-of-power supply to the power unit auxiliaries	10.80
Reactor SCRAM	11.20
Turbine stop valve closure	11.40
Diesel-generators start-up	12.80
BRU-A valve opening	13.50
Additional boron injection start to pressurizer	41.75

End of Reactor cooling pump set coast down of operable loops	87.0
Closing of BRU-A	130.0
High pressure injection system of loop 3 and 4 starts	140.0
Emergency feed water pump loop 3 and 4 start injection	1000.0
Filling of operable steam generators 3 and 4 to the nominal water level	2400.0
End of calculation	5000.0

4. Drift-Flux Model

The conservation equations for the sub-channels are obtained by area-averaging, a movement in sub-channel method, of the two-Fluid Model (2FM) conservation equations and reducing them to the DFM formulation. It is apparent that the drift-flux model based on the four field equations is an approximate theory of the two-fluid model of four conservation equations instead of three mixture equations and half dozen two-fluid equations by separating liquid and vapor continuity equations and defining an interface mass exchange between the phases in terms of a vapor production. It is basically simplified set of six conservation equations as presented in 2FM code like THERMIT except for the energy equations which are here composed in terms of enthalpies instead of internal energies^{9,10}. The expression of the governing equations of the problem under consideration will be founded on the drift-flux approach developed by Zuber and coworkers⁹. The summary set of one dimensional DFM conservation Equations given as:

Conservation of vapor mass:

$$\frac{\partial}{\partial t}(\langle \alpha \rho_v \rangle)_i + \frac{\partial}{\partial z}(\langle \rho_v j_v \rangle)_i = \langle \Gamma_v \rangle_i - \frac{1}{A_i} \sum_k^{N_i} W_{vi \leftrightarrow k} \quad (1)$$

Conservation of liquid mass:

$$\frac{\partial}{\partial t}[\langle (1-\alpha) \rho_l \rangle]_i + \frac{\partial}{\partial z}(\langle \rho_l j_l \rangle)_i = -\langle \Gamma_v \rangle_i - \frac{1}{A_i} \sum_k^{N_i} W_{li \leftrightarrow k} \quad (2)$$

Conservation of mixture momentum:

$$\frac{\partial}{\partial t}[\langle \rho_v j_v \rangle + \langle \rho_l j_l \rangle]_i + \frac{\partial}{\partial z} \left[\left\langle \frac{\rho_v j_v^2}{\alpha} \right\rangle + \left\langle \frac{\rho_l j_l^2}{(1-\alpha)} \right\rangle \right]_i = -\left\langle \frac{\partial P}{\partial z} \right\rangle_i - \langle F_{vv} + F_{wl} \rangle_i - [\langle \alpha \rho_v \rangle + \langle (1-\alpha) \rho_l \rangle]_i g - \frac{1}{A_i} \sum_k^{N_i} \left[\dot{M}_{vi \leftrightarrow k} + \dot{M}_{li \leftrightarrow k} \right] S_{i \leftrightarrow k} \quad (3)$$

Conservation of mixture energy:

$$\frac{\partial}{\partial t} [\langle \alpha \rho_v h_v \rangle + (1 - \alpha) \langle \rho_l h_l \rangle]_i + \frac{\partial}{\partial z} [\langle \rho_v j_v h_v \rangle + \langle \rho_l j_l h_l \rangle]_i = -\langle Q_{wv} + Q_{wt} \rangle_i - \frac{1}{A_i} \sum_k \left[\dot{E} \right]_{vi \rightarrow k} + \dot{E}_{li \rightarrow k} S_{i \rightarrow k} \quad (4)$$

5. Results and Discussion

Figure 5 shows the liquid enthalpy, liquid mass velocity and liquid temperature in hot fuel assembly at steady state

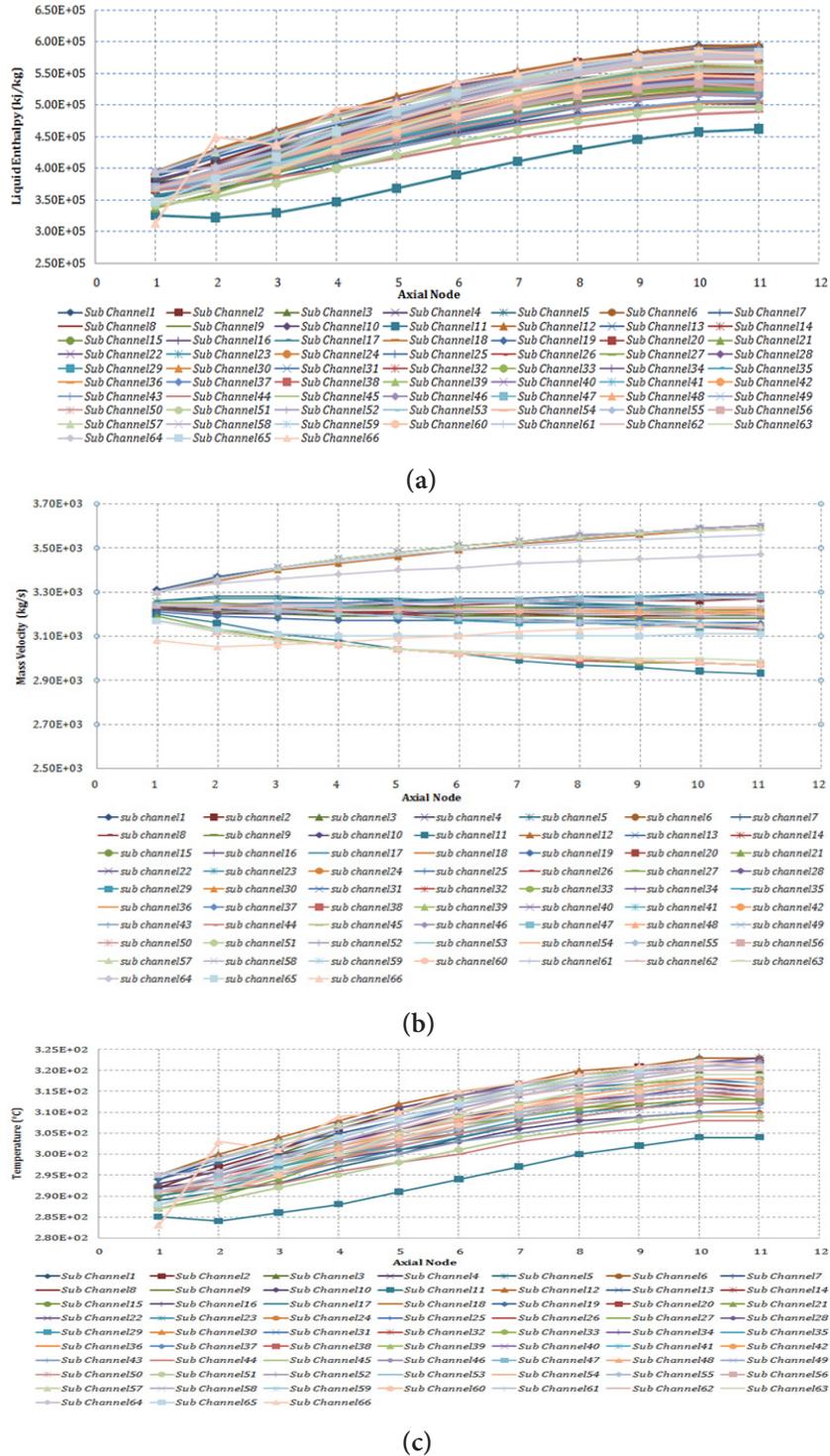
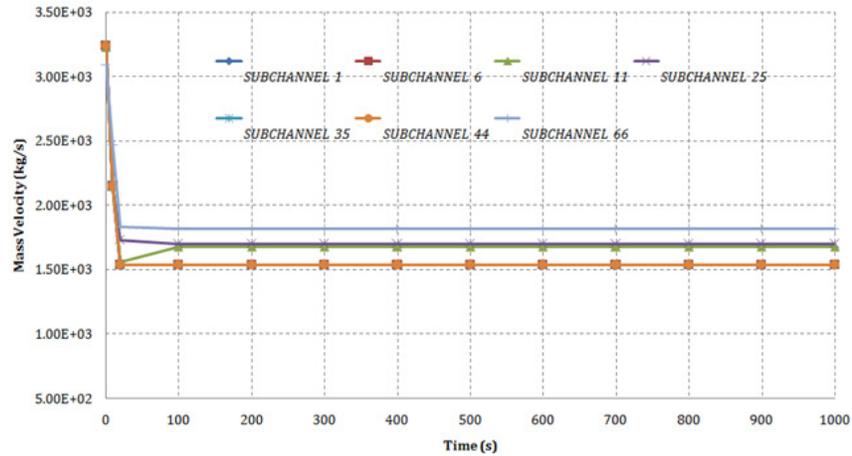


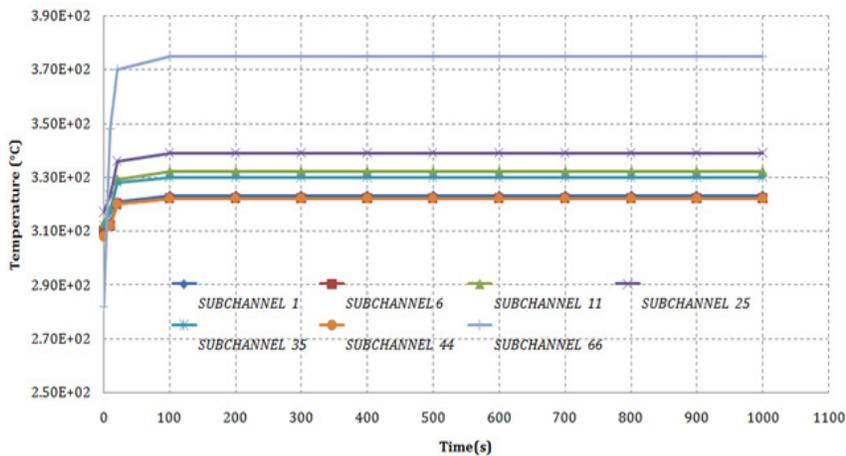
Figure 5. Thermal-hydraulic parameters calculated by drift-flux model. (a) Liquid enthalpy. (b) Mass velocity. (c) Temperature.

operating condition. The assembly divided into ten axial regions. It can be seen that the liquid enthalpy rises as similar as liquid temperature diagram.

According to the presented data in Table 1 and Figure 5, the liquid enter the assembly with a temperature of 290°C and exit with a temperature about 320°C. The internal channels have higher entropy and temperature.



(a)



(b)

Figure 6. Transient thermal-hydraulic parameters calculated by the drift-flux model.

However, the unheated rods decrease the temperature of neighboring channels. Besides the steady state calculation, transient evaluation was performed to calculate the cross sectional averaged transient temperature and flow variations.

It is obvious from the Figure that thermal hydraulic parameters raise during the first seconds of the abnormal condition, but the safety margin didn't exceed.

6. Conclusion

The drift flux model based sub-channel program is examined using steady state and transient parameters of VVER-1000 fuel assembly. Some deviation from the safety analysis report data can be found in the channels, thermal-hydraulic details which applied to the program. The calculated parameters of the program are in good

agreement with the final safety analysis report. Moreover, the drift flux which inherently is a two-phase model can predict the thermal hydraulic parameters in a single phase with a high accuracy, too. In addition, it can model the assembly with complicated shapes and could take into account the unheated rods effect.

7. References

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