THREE-DIMENSIONAL POROUS MEDIA BASED NUMERICAL INVESTIGATION OF SPATIAL POWER DISTRIBUTION EFFECT ON ADVANCED NUCLEAR FUEL ROD BUNDLES CRITICAL POWER

Zoran V. Stosic\textsuperscript{1}, Vladimir D. Stevanovic\textsuperscript{21} and Tadashi Iguchi\textsuperscript{3}

Abstract. The influence of spatial power generation shape on thermal-hydraulics behaviour of the fuel rod bundle has been investigated. Particularly, the occurrence of the local Boiling Transition has been analysed, indicating that conditions for the Critical Heat Flux (CHF) are reached somewhere within the boiling water channels in the assembly. The two-phase coolant flow within the bundle is represented with the two-fluid model in 3D space. The porous medium concept is applied in the simulation of the two-phase flow through the rod bundle implying non-equilibrium thermal and flow conditions. The governing equations in three-dimensions are discretized with the control volume method. The 3D numerical simulation and analyses of thermal-hydraulics in a complex geometry of an advanced nuclear fuel assembly are performed for conditions of a partial and/or complete rods uncovering indicating occurrence of high quality CHF - Dryout.

The obtained results from numerical simulations are compared with experimental Critical Power data obtained from full scale tests. Employed is an electrically heated test rod bundle with real 1:1 geometry. Different radial and axial power distributions are used with wide range of inlet mass flow rates (2 - 19 kg/s) and coolant inlet subcoolings (25 - 185 kJ/kg). The coolant pressure, equal to 6.9 MPa, is typical for BWRs conditions. Comparison of the predicted Critical Power values with measured data shows encouraging agreements for all analysed power distributions and the results completely reflect measured two-phase mixture cross flows, steam void distribution and spatial positions of Dryout onsets. Based on performed numerical investigation, an improvement of Dryout criteria is proposed.

Dynamic effects of power shape change on spatial thermal hydraulics and hence on CHF occurrence as well as the influence of transfer function on thermal hydraulics under cyclic power and/or flow rate changes are also being analysed. Experiments for such verifications are being performed and here presented. The applied approach could be well used for optimisation of spatial power generation in respect to improved thermal hydraulics and increased margins to CHF. Hence, fully realistic simulation of nuclear fuel rod bundle thermal-hydraulic performance is going to be obtained. Preliminary comparisons of numerical simulation results with experimental data of transient controlled variation of axial power shape show favourable agreements.

Keywords: Two-Phase Flow, Modelling, Multi-Fluid Flow, Porous Media Approach, Critical Heat Flux

1. Introduction

Radial, local and axial design peaking factors of a typical Boiling Water Reactor (BWR) are arranged so that the total maximum-to-average peaking is about 2 to 2.5. These peaking factors are selected by analysing and evaluating performance data from large operating BWRs. The basic aim of an operating procedure is to locate the control rods so that the reactor operates with approximately the same axial power shape throughout an operating cycle, providing the minimum of the total maximum-to-average peaking over the cycle (Stosic, 1995; 1997). The relative assembly power distribution (i.e., radial distribution) is normally fairly flat in a BWR because of greater void fractions in the central bundles of the core.

Because of the presence of relatively high void fractions in the upper part of the core, there is a natural characteristic for a BWR to have the axial power peak in the lower part of the core. During the early part of an operating cycle, bottom entry control rods permit a partial reduction of this axial peaking by locating a larger fraction of the control rods in the lower part of the core. From the middle of the cycle the continuous depression of the peak occurs. Near the end of the operating cycle, the higher accumulated exposure and greater depletion of the fuel in the lower part of the core reduces further the axial peaking, moving it from the bottom part, over the middle to the top of the core. This propagation of the axial power peak, directed to the channel outlet, influences complete thermal hydraulics of the assembly and generally decreases thermal margin defined as the margin which guaranties operation without boiling crisis and fuel cladding overheating (Stosic, 1995; 1997).

\textsuperscript{2}Framatome ANP GmbH – NBTT, Bunsenstr. 43, D-91058 Erlangen, Germany, Zoran.Stosic@Framatome-ANP.de
\textsuperscript{3}Temporarily at Framatome ANP GmbH, P.O.Box 3220, D-91050 Erlangen, Germany
\textsuperscript{4}JAERI - Japan Atomic Energy Research Institute, P.O.Box 319-1195, Tokai, Ibaraki, Japan
\textsuperscript{5}A.T.H.A. - Advanced Thermal Hydraulics with Applications. e-Mail: ATHA@Stosic.de, URL: http://www.Stosic.de
Another important variation of the axial power distribution is the dynamic change of the complete axial shape within couple of seconds caused by the hypothetical transient events resulting in a deterioration of the heat transfer mechanisms along the channel (Stosic, 1997). When the reactor protection system calls for a reactor scram, the control rods are driven into the core. This decreases drastically the bottom power, moving the peak to the upper part of the core, increasing it simultaneously. After the control rods are reaching the last third of the active fuel length the absorption effects are enough great to force the axial power shape practically to the initial distribution, before the rods are fully in the core (Stosic, 1997).

2. Objectives of the Study

The influence of ‘steady-state’ (exposure cycle dependent) and dynamic shape variation effects on Dryout occurrence and post-Dryout thermal and hydraulic behaviour of a BWR heated channel in transient conditions has been analysed by Stosic (1995; 1997) with the 1D-based thermal hydraulic model HECHAN. Steady-state (initial) axial power shape influence has been analysed for the case of the closure of the Main Steam line Isolation Valves, with neutron flux scram. The typical flat, bottom and top peak, chopped cosine, and double hump axial shapes were evaluated and compared (Stosic, 1995; 1997). The influence of dynamic variation of axial power shape during the transient has been analysed for the Loss of Heat Sink by Turbine Trip without Bypass with and without reactor scram (Stosic, 1997). The dynamic change of the axial power distribution was calculated by the 1D reactor kinetic model COSBWR, which has been coupled with the model HECHAN (Stosic, 1997) in that calculation. Performed analysis (Stosic, 1997) showed the importance of taking into account the power shape effects and their appropriate evaluation. The greatest effect of steady-state axial power shape on transient maximum channel cladding temperature has been found between the bottom and the top peak distribution, resulting in the temperature difference higher than 100 °C. The Dryout front reached about 0.5 m lower height under the bottom peak shape than in case of top peak (Stosic, 1997). Under the axial power distribution change during the transient for the bottom peaked power, the contribution of these effects on the maximum channel cladding temperature increase was found to be up to 30 °C, for a 3 seconds duration of a complete axial power shape transition till the initial one (Stosic, 1997).

Numerical simulation and analyses of the CHF conditions and nuclear fuel rod bundle steady state and transient thermal-hydraulics is performed with the three-dimensional two-fluid model and by the application of the porous media concept for the coolant two-phase flow within bundle complex geometry. The ability of this methodology to predict multidimensional two-phase flow thermal-hydraulics is an improvement in regard to the commonly used one-dimensional subchannel analyses methods. In regard to the full CFD calculation of two-phase flow, with the implementation of the boundary conditions at the walls of all fuel rods in the bundle, here proposed methodology requires much less computational effort with the acceptable reliability of obtained results for the engineering applications.

So, the objective of this study is to investigate the ability of here applied 3D porous media approach to model thermal hydraulics in advanced nuclear fuel rod bundles with emphasis on CHF prediction and its location under various patterns of spatial power distribution. The analyses cover both steady state and transient conditions. After verification the approach could be well used for optimisation of spatial power generation in respect to improved thermal hydraulics and increased margins to CHF. Hence, fully realistic simulation of nuclear fuel rod bundle thermal-hydraulic performance is going to be obtained.

3. Modelling Approach

Three-dimensional liquid and vapour two-phase flow is modelled by the “two fluid” model (Stevanovic, et. al., 1995; Stosic and Stevanovic, 2001a; 2001b; 2001c). Mass, momentum and energy fluid flow conservation equations are written for both phases. The general form of the conservation equations takes into account the geometry and hydraulic characteristics of free flow channels for two-phase flow around rods in a bundle through the application of the porous-media concept. Mass, momentum and energy transfer at the vapour-liquid interface, as well as rod bundle hydraulic resistance to two-phase flow, heat transfer from hot rods and boiling within a rod bundle are modelled by “closure laws”. This approach implies non-equilibrium thermal and flow conditions.

3.1. Assumptions

The following assumptions are introduced:

- The porous medium concept is used in the simulation of two-phase flow within a rod bundle. The space of the numerical control volume can be occupied by one or both phases – vapour and liquid, as well as by rods. The flow volume reduction due to the presence of rods in a space occupied by a bundle is taken into account. Therefore, the conservation of the vapour and liquid flow parameters is performed only for the fractions of the numerical control volume occupied by corresponding phase.
- A rod bundle flow resistance is assumed continuously distributed in the space occupied by these elements.
- Flow governing equations are written in the non-viscous form, while the turbulent viscosity effects are taken into account indirectly through friction coefficients for the rod bundles flow resistance and two-phase interfacial drag force.
The two-phase flow is observed as semi-compressible, that is the acoustic flow effects are neglected, while the influence of the pressure change on the vapour and liquid thermo-physical properties is taken into account.

The surface tension is neglected as it is not important for bulk two-phase flow phenomena. Hence, pressure is the same for both phases within the numerical control volume.

### 3.2. The governing equations

**Mass conservation**

\[
\frac{\partial (c_k \rho_k u_k)}{\partial t} + \nabla \cdot (c_k \rho_k u_k u_k) = (-1)^{k} (\Gamma_e - \Gamma_c)
\]

**Momentum conservation**

\[
\frac{\partial (c_k \rho_k u_k)}{\partial t} + \nabla \cdot (c_k \rho_k u_k u_k) = -c_k \nabla p + c_k \rho_k \ddot{g} + \left((-1)^{k} \ddot{F}_{L2} + (-1)^{k+1} \ddot{F}_{VM} + (-1)^{k+1} \ddot{F}_{21} - \ddot{F}_{3k} + (-1)^{k} (\Gamma_e - \Gamma_c) \ddot{u}_k \right)
\]

**Energy conservation**

\[
\frac{\partial (c_k \rho_k h_k)}{\partial t} + \nabla \cdot (c_k \rho_k h_k u_k) = (-1)^{k} (\Gamma_e - \Gamma_c) h^* + \dot{q}_{3k}
\]

The index \( k \) is 1 for water and 2 for steam. The source terms for mass, momentum and thermal energy conservation are written on the r.h.s. of Eqs. (1)-(3). The intensity of phase transition, which is the mass of evaporation or condensation per unit volume and time, are denoted with \( \Gamma_e \) and \( \Gamma_c \) respectively. The force of vapour and liquid interfacial drag per unit volume is denoted with \( \ddot{F}_{21} \), while the forces of tubes/rods resistance to liquid and vapour flow within a bundle, per unit volume, are represented with \( \ddot{F}_{3k} \) and \( \ddot{F}_{32} \) respectively. Terms \( \ddot{F}_{L2} \) and \( \ddot{F}_{VM} \) represent lift force and virtual mass force, respectively. The term \( \dot{q}_{3k} \) represents volumetric heat rate from rods/tubes to corresponding fluid phase per unit volume.

The phases’ void fraction balance holds in the form

**Volume fraction balance**

\[
\alpha_1 + \alpha_2 + \alpha_3 = 1
\]

where the vapour volume fraction in the two-phase mixture (void) is determined according to the expression

\[
\varphi = \alpha_2 \left( \frac{\alpha_1}{\alpha_2 + \alpha_3} \right)
\]

and the porosity is defined as

\[
\psi = 1 - \alpha_3
\]

### 3.3. Closure Laws

Necessary closure laws are introduced for the interfacial drag force \( \ddot{F}_{21} \), lift force \( \ddot{F}_{L2} \), virtual mass force \( \ddot{F}_{VM} \), drag force on the fuel rods walls \( \ddot{F}_{3k} \), phase transition rates \( \Gamma_e \) and \( \Gamma_c \), volumetric heat flux \( \dot{q}_{3k} \), as well as criteria for two-phase flow pattern transitions are given in detailed in (Stosic and Stevanovic, 2001a; 2001b; 2001c).

### 3.4. Solution method

A finite volume method is applied for the solution of governing partial differential equations. The discretization of these partial differential equations is carried out by their integration over rectangular control volumes of variable size in the 3D Cartesian coordinate system. The conservation equations for liquid and steam mass and enthalpy are integrated over the scalar control volumes, while the liquid and vapour momentum conservation equations are integrated over the staggered control volumes. The convective terms at the control volume boundaries are determined with the upwind numerical scheme. Fully implicit time integration is applied. Numerical schemes have been developed for the calculation of the void fractions and pressure fields. Liquid and steam mass conservation equations are discretized with the finite differences. The pressure field is calculated according to the modified SIMPLE numerical method (Patankar, 1980), taking into account the presence of two phases - liquid and vapour. The resulting set of discretized equations is solved iteratively by the Alternating Direction Implicit (ADI) method. For the calculation of a steady-state condition, the transient calculation procedure is performed with constant boundary conditions.

### 4. Dryout Criteria

The dry-out criteria for churn-turbulent two-phase flow, which is assumed to prevail for voids higher than 0.3 and steam superficial velocities lower than 15 m/s (Stosic and Stevanovic, 2001a; 2001b), is the condition under which the
steam void fraction reaches the value of 0.98.

For annular flow which is assumed to exist when void fractions are higher than 0.3 and steam superficial velocities higher than 15 m/s, it is adopted that the dry-out occurs when the liquid film reaches the minimum possible thickness, established from a force balance on the creeping film, which yields (Borkowski, Wade, 1992)

\[
\delta_{\text{min}} = \left(\frac{18\sigma\mu_1^2}{g^2\rho_1^3}\right)^{0.2}
\]  

(7)

This equation gives the value of 1.5 \times 10^{-4} m for the minimum film thickness at the pressure of 6.9 MPa, under which the experiments were performed. The annular flow consists of the liquid film flow, liquid droplets flow and the vapour flow within the vapour core. Numerical experiments have shown that the liquid film volume fraction is one order of magnitude higher than the liquid droplets volume fraction (Stevanovic, et. al., 1995), while the mass flow rates are of the same order of magnitude due to the higher droplets velocities than the liquid film velocity. Therefore, the liquid droplets contribution to the total liquid volume fraction in the annular flow is approximated as one tenth, and for the above calculated value of the minimum film thickness and the rod bundle geometry, the critical void value for the dry-out occurrence is 0.95.

5. Steady State Analyses

The steady state verification of spatial power distribution effects on prediction of CHF occurrence has been performed using the developed model and measured data on a Framatome ANP Karlstein ATRIUM 10-9Q full scale rod bundle (Figure 1) having an eccentric quadratic water channel and 8 part length fuel rods (Stosic, 1999).

Four bundles with different local and axial peaking shapes are used in here presented verification analyses, and are shown in Figures 2 and 3. Drawn are differences between bundle and rod axial power profiles - caused by the presence of part length fuel rods, positions of spacers and beginning of partly rodded region. In the first set of critical power tests with chopped cosine axial power shape the Dryout occurrence is first detected by the rod 4 at spacers 1 and/or 2 from the top of bundle active height, as indicated in Figures 2 and 3 (RPP-1). In the second set of critical power tests, also with chopped cosine axial power shape, the Dryout occurrence is first detected at the rod 87 at spacers 1 and/or 2 from the top of bundle active height, as indicated in Figures 2 and 3 (RPP-2). These two tests (Chopped Cosine with RPP-1 and RPP-2) have the same axial power generation shape but different Radial power Peaking Pattern (RPP) causing occurrence of Dryout in two different sections opposite to the water channel. In the case of bottom peak axial power shape the Dryout occurrence is first detected by the rod 79 at one or all three spacers from the top of active height, as indicated in Figures 2 and 3 (RPP-3). In the fourth set of analysed tests with top peak axial power shape, the rod 79 has first experienced occurrence of Dryout at spacer 1 from the top, as shown in Figures 2 and 3 (RPP-3). These three tests (Chopped Cosine with RPP-2 and Bottom and Top Peak with RPP-3) have similar radial peaking pattern but totally different axial power distributions.

The strategy of verification was based on the modelling and recalculating of three-dimensional bundle thermal hydraulics, based on measured boundary conditions and following the experimental procedures. The bundle critical power, i.e. the onset of Dryout, and the spatial position (axial height and rod number) of its occurrence were monitored and they were parameters of primary interest. The experimental procedure was followed and simulated by the model until the calculation predicted Dryout occurrence within the bundle. At these moment calculation stopped and the obtained power was the predicted bundle Critical Power, according to the criteria given in the previous section.

Performed analyses covered a wide range of bundle inlet subcooling between 20 kJ/kg and 190 kJ/kg and of inlet mass flow rates from 3 kg/s to 19 kg/s. About 150 cases are analysed.

Results of measured and calculated lateral locations of rods which first experienced the onset of Dryout are shown in Figure 4 for all analysed cases. Obviously, the model has predicted the right section of Dryout occurrence for all spatial peaking pattern, even the right rod number or surroundings ones in most cases. In Figure 5 the obtained results are shown as a ratio of calculated to measured bundle Critical Power data. As it can be seen from Figure 5 the obtained disagreements in calculated Critical Power are in the range from −18% till +27%. Important is that very satisfactory results of the influence of quite different spatial power generation on predicted Critical Power can be seen from Figure 5. The effect of coolant bundle inlet subcooling is modelled with encouraging agreement. It should be noted that from Figure 5 is obvious that obtained disagreements are very systematic and appear to be a strong function of bundle inlet mass flow rate.

Moreover, performed analyses have shown that this function completely reflects the influence of local vapour velocity on the occurrence of Dryout. This is illustrated in Figure 6 where the comparisons of all results of the Critical Power values are shown as a function of vapour superficial velocity at location of maximum void fraction occurrence. Dependency on flow regime is found to be weak, i.e. the criteria used for Dryout occurrence can be improved with simple linear function of vapour local superficial velocity, presented in Figure 6. In this way corrected Dryout criteria (discussed in Section 4) provide agreements of calculated to measured Critical Power values in an encouraging range of less than ±8% with very satisfactory mean of the response (calculated to measured Critical Power) variable of 1.0329 and estimated standard deviation of the model error equals to 0.0736.
6. Transient Analyses

Developed model has also been successfully verified against measured data of onset of Critical Power and duration of post-Dryout regime on a Karlstein ATRIUM™ 10-9Q full scale rod bundle (Figure 1) simulating loss of coolant flow transient (Stosic and Stevanovic, 2001b). The bundle had similar local and axial peaking pattern as presented in Figures 2 and 3 (RPP-2). Transient analyses presented by Stosic and Stevanovic (2001b) have not been performed with different spatial power distributions but only with one – chopped cosine – and furthermore kept constant during the whole transient.

In order to investigate phenomenological effects of variation of axial power shape on spatial thermal hydraulics, transient simulations of here presented model is now being performed employing JAERI (Japan Atomic Energy Research Institute) experiments on CHF and post-CHF conditions under dynamically controlled axial power shape. The purpose of these experiments is investigation on:

- Axial power shape (bottom-peak, middle-peak, or top-peak) effect on spatial thermal hydraulics as well as on CHF.
- Dynamic change effect of power shape on spatial thermal hydraulics as well as on CHF, and
- Transfer function on thermal hydraulics under cyclic power and/or flow rate changes.

Previous CHF experiments at JAERI have usually been performed under constant axial power shape conditions. However in nuclear reactors, due to void reactivity feedback effect, the axial power shape can be changed dynamically, and then the local quality is affecting the axial power shape and the void transit time. It is supposed that both the axial power shape and the void transit time affect positively on spatial thermal hydraulics and on CHF under some condition, while they can also affect negatively on CHF under other condition. This sequence is illustrated in Figure 7. Two void transit lines are supposed: a solid line Case-1 and a dotted line Case-2, mainly depending on mass flux. In Case-1 the fluid is heated by high power at the bottom, middle and top part, so that the supplied power is much higher in comparison to the case of constant power shape during transient. In Case-2 the fluid is heated by high power (bottom part), low power (middle part), and high power (top part), so that the supplied power is slightly higher in comparison to case of transient constant power distribution. These phenomena are not cleared yet in experimental work and have not been used for numerical model verifications. In order to analyse this phenomena, it is necessary to investigate the effect of transfer function on thermal-hydraulics under cyclic power and/or flow rate changes.

Heater, which axial power shape can be dynamically controlled was installed in the JAERI experimental facility with a power control system. Operational pressure is in the range of 2 - 20 MPa and the mass flux is 100% - 20% of that of conventional BWRs. Power and mass flux can be controlled in cyclic change. Figure 8 shows a heater of dynamically controlled axial power shape. Heater elements are divided into three parts. Each part is connected to different power unit. The power unit can be controlled independently of each other, so that the axial power shape of the heater can be controlled dynamically to simulate transient between bottom peak, chopped cosine and top peak, as well as vice versa. Outer diameter and heated length of the heater are typical for BWR rods and are equal to 12.3 mm and 3.71 m, respectively.

Two types of experiments are being performed:

1. CHF experiments under constant axial power shape, and
2. CHF experiments under dynamically controlled axial power shape.

Experiments are performed under dynamically controlled axial power shapes, as well as constant chopped cosine, bottom peak and top peak axial power shapes. Experimental conditions (power, mass flux) are set either to be constant or to be cyclic. Pressure to be tested is 0.1 – 16 MPa. Mass flux to be tested is 100% - 20% of BWR. Peaking factor to be tested 1 – 1.4. Tested frequency of mass flux, power, and axial power shape is 1 – 0.1Hz. Phase difference of bottom-to-top powers is 0 – 360 degree. This phase difference simulates void transit time in BWRs.

Preliminary comparisons of the predicted thermal hydraulic parameters as well as occurrence of the CHF condition during transients under dynamically controlled axial power distribution are showing encouraging results and will be demonstrated soon.

7. Conclusion

The two-fluid modelling approach with the application of the porous media concept is applied to the simulation and analyses of the nuclear fuel rod bundle of the new design, which introduces strong multidimensional thermal hydraulic effects due to the complex bundle geometry.

Different radial and axial power distributions are used for steady state verification with wide range of inlet mass flow rates and coolant inlet subcoolings. The simple physically based criteria are used for the onset of dry-out. The proposed modelling approach and related improvements are tested against experimental data. The application of the proposed Dryout criteria and the correction of the results according to the observed influence of the local vapour superficial velocity on the CHF value, the agreements of the calculated to measured Critical Power values are obtained in the range of less than ±9% for all analysed spatial power distributions.

Dynamic change effects of power shape on CHF occurrence and post-Dryout effects, as well as the influence of a transfer function on thermal hydraulics under cyclic power and/or flow rate changes are being investigated.

The comparison of the numerical results with the measured data shows that the proposed numerical method
reliably predicts critical power, spatial void fraction distribution and location of dry-out onset in both steady state and transient conditions. Good predictions of the critical power are obtained without the use of some experimental correlation, which application could be limited by the experimental conditions for its derivation. The applied approach could be well used for the optimisation of spatial power generation in respect to improved thermal hydraulics and increased margins to CHF. Hence, fully realistic simulation of nuclear fuel rod bundle thermal-hydraulic performance is going to be obtained.

REFERENCES


Fig. 1. ATRIUM™-10 Fuel Assembly and Axial Geometry Parameters as a Consequence of Part Length Fuel Rods
Fig. 2. Different Radial Peaking Pattern (RPP) of Test Bundle used for performed Analyses

ROD No. 4 FIRST IN DRYOUT

(A) CHOPPED COSINE
RPP-1

ROD No. 79 FIRST IN DRYOUT

(C) BOTTOM AND TOP PEAK
RPP-3

ROD No. 87 FIRST IN DRYOUT

(B) CHOPPED COSINE
RPP-2

Fig. 3. Different Axial Peaking Pattern (APP) of Test Bundle used for performed Analyses

(A) CHOPPED COSINE

(B) BOTTOM PEAK

(C) TOP PEAK

DRYOUT DETECTED ON SPACERS 1 AND 2

DRYOUT DETECTED ON SPACERS 1, 2, AND 3

DRYOUT DETECTED ON SPACER 1

ROD PROFILE

BUNDLE PROFILE

END OF PLFR HEATED LENGTH

AXIAL POWER FACTOR
Fig. 4. Measured and Calculated Rods detecting first Dryout in Test Bundles with different Spatial Power Distributions

Fig. 5. Calculated to Measured Critical Bundle Power for Test Bundles with different Spatial Power Distributions
Fig. 6. Calculated to Measured Critical Bundle Power for different Spatial Power Distributions

Fig. 7. Illustration of Void Transit Lines and cyclic Power Change

Fig 8. Heater of dynamically controlled Axial Power Shape