A Large-Scale Simulation on Water-Vapor Two-Phase Flow Dynamics in Fuel Bundles of Advanced Nuclear Reactors

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In order to predict the water-vapor two-phase flow dyanamics in a fuel bundle of an advanced light-water reactor, large-scale numerical simulations were carried out using a highly parallel-vector supercomputer, the Earth Simulator. Conventional analysis methods such as subchannel and system analysis codes need composition equations based on the experimental data. When experimental data regarding the thermal-hydraulics in the tight-lattice core are insufficient, therefore, it is difficult to obtain highly prediction accuracy on the thermal design of the advanced light-water reactor. Then, the large-scale direct simulation method of two-phase flow was proposed. The axial velocity distribution in a fuel bundle changed sharply around a spacer. The bridge formation occurred at the position of adjacent fuel rods where an interval is narrow, and vapor positively flows the triangular region where the interval of adjacent fuel rods is large. From the results of the present study, the high prospect was acquired on the possibility of establishment of a new thermal design method for the advanced light-water reactor cores with the large-scale two-phase flow simulation.

Keywords: Fluid dynamics, Large-scale simulation, Two-phase flow, Fuel bundle, Nuclear reactor, Thermal design

1. Introduction

Subchannel [1]-[3] and system analysis [4],[5] codes are used for the thermal-hydraulic analysis of fuel bundles in nuclear reactors from the former, however, many composition equations and empirical correlations based on experimental results are needed to predict the water-vapor two-phase flow behavior. When there are no experimental data such as an advanced light-water reactor [6]-[8] which is currently studied by the Japan Atomic Energy Research Institute and named as the reduced moderation water reactor (RMWR), therefore, it is very difficult to obtain highly precise predictions.

The RMWR core has remarkably narrow gap spacing between fuel rods (i.e., around 1 mm) and a triangular tight-lattice fuel rod configuration in order to reduce the moderation of the neutron. In such a tight-lattice core, there is no sufficient information about the effects of the gap spacing and the spacer configuration on the two-phase flow characteristics. Then, the author considered analyzing the water-vapor two-phase flow dynamics in the tight-lattice fuel bundle with a large-scale simulation under the full bundle size condition. Although lots of calculation memories are required to attain the two-phase flow simulation for the RMWR core, the Earth Simulator [9] enabled such a request.

In JAERI, numerical investigation on the physical mechanisms of complicated thermal-hydraulic characteristics and the multiphase flow behavior with phase change in nuclear reactors is carried out In this numerical research the author pointed out the improving points of the conventional reactor core thermal design procedures and then proposed predicting two-phase flow characteristics inside the reactor core more directly than the conventional procedures for the first time in the world by controlling the concept of composition and empirical equations based on experiment data as much as possible [10]. Based on this idea, a new thermal design procedure for advanced nuclear reactors with the large-scale direct simulation method is developed. Especially, thermal-hydraulic analyses of two-phase flow positively for a fuel bundle simulated by the full size using the earth simulator are performed [11]. This is the largest in the world as the reactor core thermal design analysis. This paper describes the preliminary results of the large-scale water-vapor two-phase flow simulation in a tight-lattice fuel bundle of the RMWR core.

2. Numerical Analysis

The two-phase flow analysis code, TPFIT, which was developed by Yoshida et al. [12] is discretized by the CIP method [13] using the modified interface-tracking method [14]. The surface tension of bubble is calculated using the continuum surface force model proposed by Brackbill [15]. The tracking of an interface between the liquid and gas phase is accomplished by the solution of a continuity equation for the volume fraction of a couple of the phases. A detail of governing equations for the present two-phase flow analysis is shown in Reference [16].

Figure 1 shows the analytical geometry consisting of 37 RMWR fuel rods. The geometry and dimensions simulate the experimental conditions of JAERI [17]. Here, the fuel rod outer diameter is 13 mm and the gap spacing between each rod is 1.3 mm. The casing has a hexagonal cross section and a length of one hexagonal side is 51.6 mm. An axial length of the fuel bundle is 1260 mm. The water flows upward from the bottom of the fuel bundle. A flow area is a region in which deducted the cross-sectional area of all fuel rods from the hexagonal flow passage. The spacers are installed into the fuel bundle at the axial positions of 220, 540, 750 and 1030 mm from the bottom. The axial length of each spacer is 20 mm.

Inlet conditions of water are as follows: temperature 283°C, pressure 7.2 MPa, and flow rate 400 kg/m^{2}s. Moreover, boundary conditions are as follows: fluid velocities for x, y and z directions are zero on every wall (i.e., an inner surface of the hexagonal flow passage and outer surface of each fuel rod, and surface of each spacer); velocity profile at the inlet of the fuel bundle is uniform. The present simulations were carried out under the non-heated isothermal flow condition in order to remove the effect of heat transfer due to the fuel rods to the fluid. A setup of a mixture condition of water and vapor at the heating was performed by changing the initial void fraction of water and vapor at the inlet of the analytical domain.

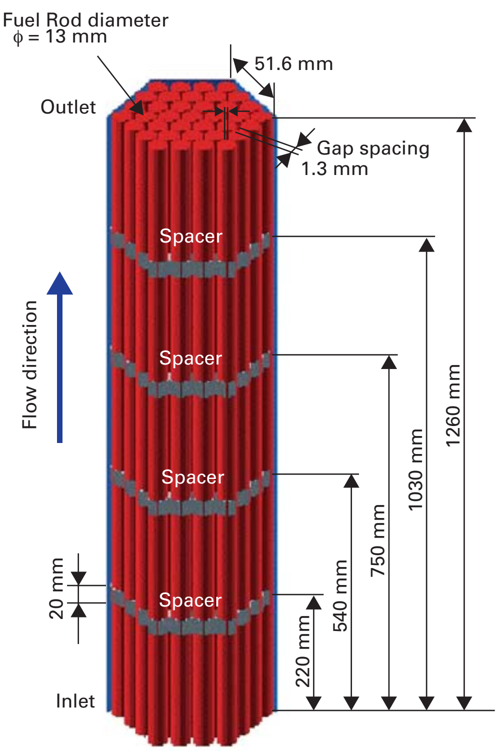


Fig. 1 Outline of three-dimensional analytical geometry of a tight-lattice fuel bundle

3. Results and Discussion

Figure 2 shows the void fraction distributions around fuel rods in the horizontal direction. Here, the void fraction is defined as the ratio of the gas flow (i.e., vapor) cross sectional area to the total cross sectional area of the flow channel In Fig. 2 the void fraction is indicated using color gradation from blue to red: 100% liquid water at blue and 100% pure vapor at red. Fig. 2 (a) is the predicted result. Each fuel rod surface shown with a circle is enclosed by thin water film, and vapor flows the outside. In the region where the

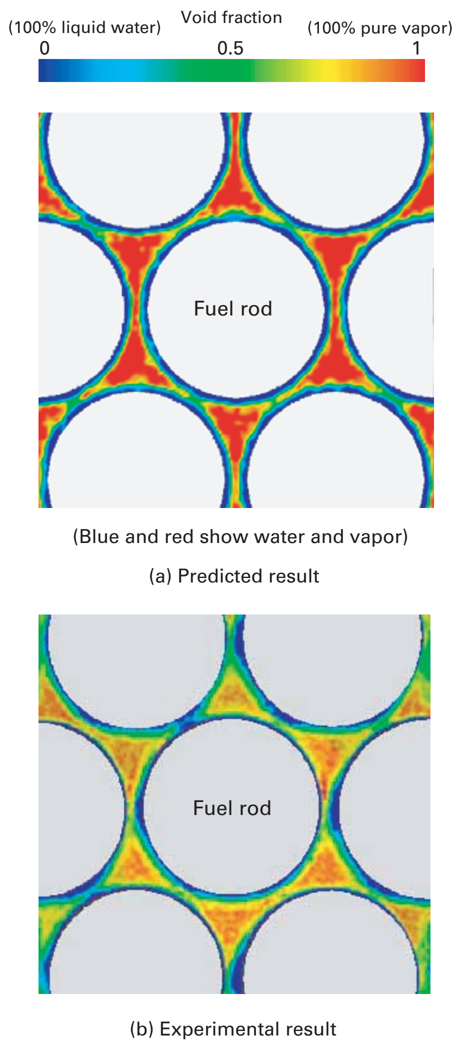
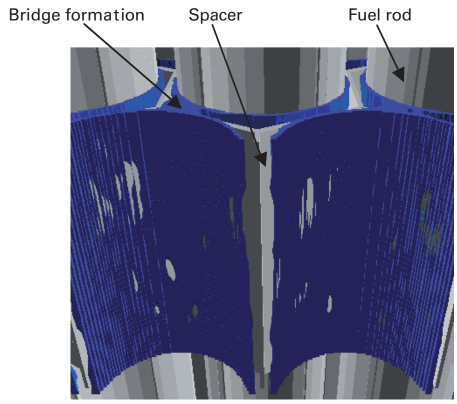
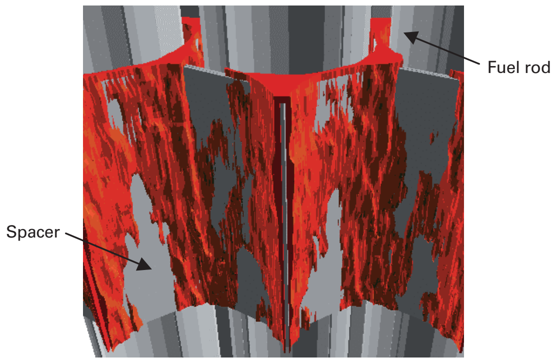


Fig. 2 Comparison of predicted void fraction distributions around fuel rods in the horizontal direction obtained from predicted and experimental results



(a) Predicted water distribution (Blue shows void fraction of 0.1 or less)



(b) Predicted vapor distribution (Red shows void fraction of 0.9 or more)

Fig. 3 Predicted water and vapor distributions in the vertical direction around a spacer

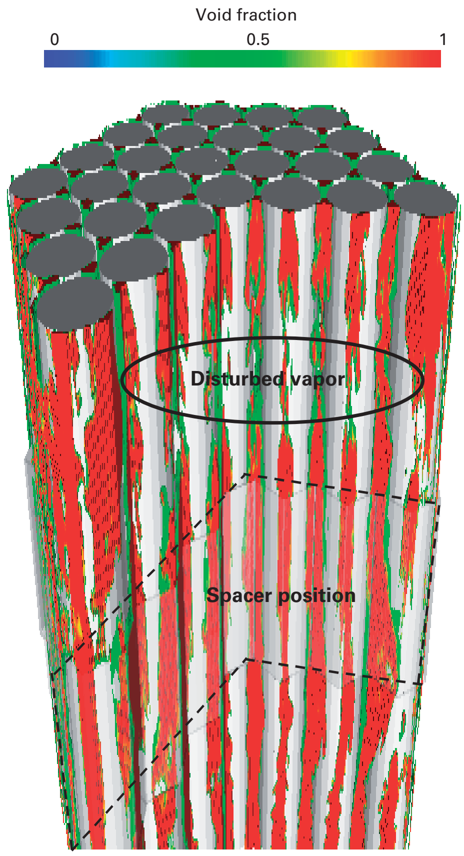


Fig. 4 Predicted vapor structure around fuel rods; where, red indicates 100% pure vapor and the void fraction is 1, and green indicates an interface between water and vapor and the void fraction is 0.5.

gap spacing between fuel rods is narrow, the bridge formation of water in which adjacent fuel rods are connected by the water film is confirmed. On the other hand, vapor flows through the center area of the fuel rods arranged in the shape of a triangular pitch. Because it is easier for vapor to flow, since the frictional resistance in this area is low compared with the narrow area.

Figure 2 (b) is an example of the experimental result of the void fraction distribution around the fuel rods which is obtained by an advanced neutron radiography technique which was developed by Kureta [18]. The general neutron radiography technique has been established based on the following features; neutron passes through materials but is blocked by water. It is possible to measure the void fraction inside a fuel bundle by non-contacting using this technology. The result was translated by the experimental result using a new image processing procedure. A tendency of the water and vapor distributions shown in Fig. 2 (a) and (b) is in good agreement.

Figure 3 shows the two-phase flow configurations around a spacer position in the axial direction. Fig. 3 (a) shows the water distribution. Here, blue represents a region where the void fraction is 0.1 or less and it is occupied with water of about 100%. Moreover, Fig. 3 (b) shows the vapor distribution. Here, red represents a region where the void fraction is 0.9 or more and it is occupied with vapor of about 100%. Much water can be seen at to the circumference of a fuel rod. Water exists as the liquid film. A bridge formation can be checked. In addition, much vapor exists along the spacer in the axial direction. Thus, a difference in the water and vapor distributions in the vicinity of the fuel rods and spacer is clear.

Figure 4 shows an example of the predicted vapor structure around the fuel rods. Here, the distribution of void fraction within the region from 0.5 to 1 is shown: 0.5 indicates just an interface between the water and vapor and is shown by green; and 1 indicates the non-liquid vapor and is shown by red. Vapor flows from the upstream to downstream like a streak through the triangular region, and the interaction of the vapor stream to the circumferencial direction is not seen. On the other hand, since the vapor is disturbed behind a spacer, the influence of turbulence by existence of the spacer can be predicted.

4. Conclusions

In order to predict the water-vapor two-phase flow dynamics in the RMWR fuel bundle and to reflect them to the thermal design of the RMWR core, a large-scale simulation was performed under a full bundle size condition using the Earth Simulator. Details of water and vapor distributions around fuel rods and a spacer were clarified numerically. A series of the present preliminary results were summarized as follows:

1) The fuel rod surface is encircled with thin water film;
2) The bridge formation by water film appears in the region where the gap spacing between adjacent fuel rods is narrow;
3) Vapor flows into the triangular region where the gap spacing between fuel rods is large.
4) A flow configuration of vapor shows a streak structure along the triangular region.

The author would like to establish the large-scale simulation method for detailed reactor core thermal design, and furthermore, to build the hybrid thermal design method with higher prediction accuracy, combining the conventional thermal design procedure and the presently proposed the large-scale simulation method.

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直接解析手法による原子炉内複雑熱流動挙動の大規模数値シミュレーション

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原子炉熱設計に必要である炉心内水－蒸気系二相流現象の詳細を大規模シミュレーションによって明らかにする研究を行っている。従来の熱設計手法ではサブチャンネル解析コードに代表されるように実験データに基づく構成式や経験式を必要とするが、新型炉に関しては熱流動に関する実験データが十分ではないため、従来手法による熱設計では高精度の予測は困難である。そこで、シミュレーションを主体とした先進的な熱設計手法を開発し、従来手法と組み合わせることによって効率的な新型炉開発の実現を目指している。

一例として、本報には、日本原子力研究所が開発を進めている革新的水冷却炉の燃料集合体内二相流挙動に関する大規模シミュレーションの研究成果を示す。革新的水冷却は、減速材の割合を減らして中性子の減速を抑制することで1以上の高い転換比が期待できる原子炉である。炉心には、直径13 mm程の燃料棒が1 mm程度の燃料棒間ギャップで三角ピッチ状に稠密に配置される。このような稠密燃料集合体1カラム(37本の燃料棒によって構成される)を対象にして3次元二相流シミュレーションを行い、次のような水と蒸気の分布挙動を明らかにした。

① 燃料棒表面は薄い液膜で覆われる。また、隣り合う燃料棒の間隔が最も狭い領域で液膜の架橋現象が起こる。
② 水平断面の燃料棒間隔が広い領域では狭隘部分に比べて局所的に流動抵抗が低いため水に比べて蒸気の方が流れ易い。
③ 気泡の運動は流れ方向に対する移動が支配的であり、クロスフローのような水平断面方向への移動は小さい傾向にある。

キーワード：熱流動，新型炉熱設計，二相流シミュレーション，解析手法開発，稠密燃料集合体