

Large-Scale Simulations on Thermal-Hydraulics in Fuel Bundles of Advanced Nuclear Reactors

Project Representative

Kazuyuki Takase Japan Atomic Energy Agency

Authors

Kazuyuki Takase^{*1} and Hiroyuki Yoshida^{*1}

^{*1} Japan Atomic Energy Agency

In order to predict the water-vapor two-phase flow dynamics in a fuel bundle of an advanced light-water reactor, large-scale numerical simulations were performed using a highly parallel-vector supercomputer, the earth simulator. Although conventional analysis methods such as subchannel codes and system analysis codes need composition equations based on the experimental data, it is difficult to obtain high prediction accuracy when experimental data to obtain the composition equations. Then, the present large-scale direct simulation method of water-vapor two-phase flow was proposed. The void fraction distribution in a fuel bundle under boiling heat transfer condition was analyzed and the bubble dynamics around the fuel rod surface were predicted quantitatively.

Keywords: Large-scale simulation, Thermal hydraulics, Fuel bundle, Boiling heat transfer Bubble dynamics, Nuclear reactor

1. Introduction

In light water reactors each fuel rod is arranged in the shape of a square lattice with an interval of about 3 mm. Several spacers are installed on the surface of the fuel rod with arbitrary axial positions. Water flows vertically along fuel rods and is heated by those, and then many bubbles generate. The flow configurations of the liquid-gas two-phase flow change with some parameters such as the mass velocity, channel geometry, flow rate, pressure, heat transfer, etc. These give a large effect to the pressure drop, void fraction, heat transfer and so on. Therefore, in case of conducting the thermal design of the nuclear reactor core, it is requested to clarify the liquid-gas two-phase flow configurations in detail according to the above parameters. To satisfy this request, many two-phase flow experiments using large-scale test facilities have performed and then a lot of composition equations [1]–[3] which specify the two-phase flow configurations (i.e., bubbly flow, slug flow, annular flow, mist flow, etc.) were proposed based on those experimental data.

Two-phase flow analyses with the two-fluid model codes [4]–[6] have been carried out using the composition equations. Therefore, it is not easy to get high prediction accuracy by using the two-fluid model when experimental data are not enough as an advanced light-water reactor [7]–[8]. That is, the two-fluid model is only effective to the average and macroscopic phenomenon in the flow range as the fluid flow characteristic is already clarified. Therefore, it is not the mechanistic numerical method which predicts the unstable

interface structure characterizing the liquid-gas two-phase flow behavior. On the other hand, predicting directly the two-phase flow behavior including complex transient phenomena such as phase change and flow transition without the experimental data, development of a direct two-phase flow simulation method has been performed [9]. Here, "Predicting directly" means that the mathematical models based on the physical phenomena are only used and the composition equations obtained from the experimental data are not used. This paper describes the predicted results of three-dimensional void fraction in a tight-lattice fuel bundle under boiling heat transfer condition.

2. Void Fraction Distributions in Fuel Bundles of Advanced Nuclear Reactors

2.1 Advanced Light-Water Reactor

The advanced light-water reactor of Japan Atomic Energy Agency [10] has a higher conversion ratio more than unity by controlling the water flow rates. In order to obtain 1 or more conversion ratios, it is expected from the results of the previous studies that a volume ratio of water and fuel must be decreased to about 0.25 or less. To satisfy this condition, the fuel bundle with a triangular tight-lattice arrangement is required: a fuel rod diameter is around 10 mm; and, the gap spacing between each rod is around 1 mm. Although the coolant is 100% water at the core inlet, it changes a mixture of water and vapor along the flow direction, and then, the vapor occupies 90% or more at the core outlet. Therefore,

the advanced light-water reactor has very severe cooling condition on the viewpoint of the thermal engineering.

Figure 2.1.1 shows a bird-eye view of the actual advanced light-water reactor design. It consists of a core, control rod, separator and dryer region, and a pressure vessel. The pressure vessel diameter and height are around 9 and 19 m. The core region is composed of 282 fuel bundles. Each fuel bundle has a hexagonal shape horizontally. A length of one side of a hexagonal shape is about 0.13 m and the axial length of a fuel bundle is about 2.9 m. A heating section in the core consists of two seed and three blanket regions and its length is about 1.3 m (i.e., around 0.2 m in each seed region and 0.3 m in each blanket region). In the core, MOX (mixed oxide) is used to the seed region and then the depleted UO_2 is used to the blanket region.

2.2 Numerical Analysis and Typical Result [11]

Figure 2.2.1 shows the analytical geometry consisting of a tight-lattice fuel bundle with 37 fuel rods. The geometry and dimensions simulate the experimental conditions. 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.

Figure 2.2.2 shows the present computational grids, which corresponds to one sixth of a horizontal plane. A non-uniform mesh division was applied. The total number of mesh division in the x, y and z directions are 20 million. Here, boundary conditions are as follows:

- 1) Fluid velocities for x, y and z directions are zero on every wall;
- 2) Developed velocity profile is given to the duct inlet; and,
- 3) Heat flux of each fuel rod was given to the heating section.

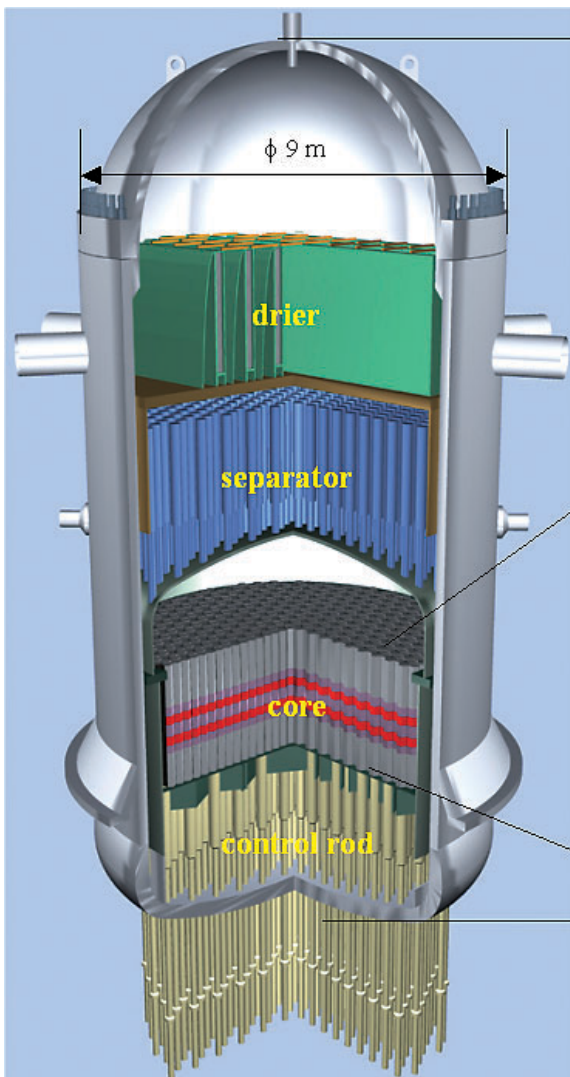


Fig. 2.1.1 A bird-eye view of the advanced light-water reactor.

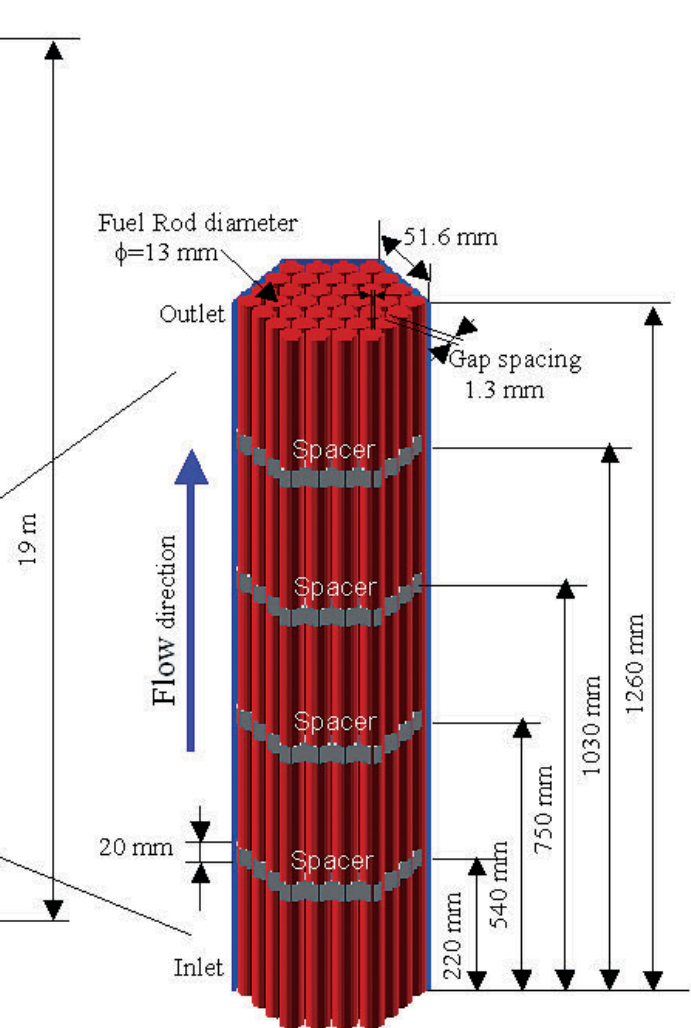


Fig. 2.2.1 Analytical geometry of a tight-lattice fuel bundle.

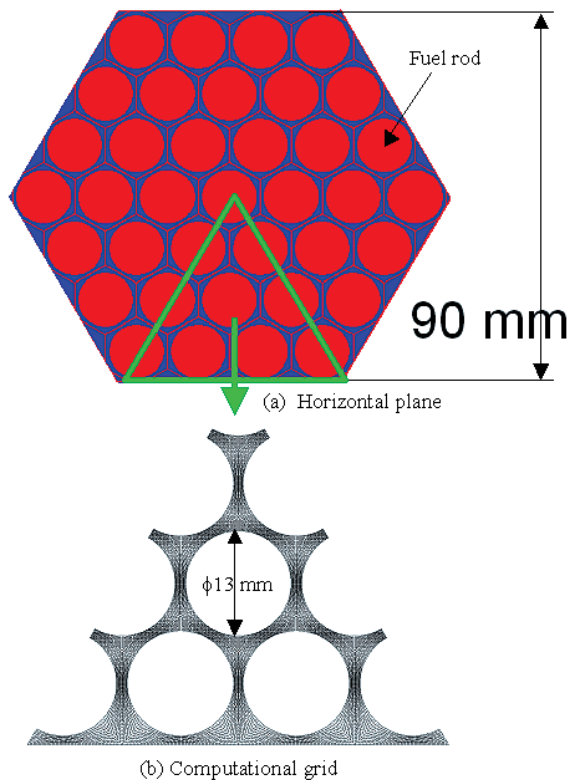


Fig. 2.2.2 Computational grid condition.

A three-dimensional predicted result of void fraction in a fuel bundle is shown in Fig. 2.2.3. The color contour indicates the void fraction distribution; blue is the liquid water (i.e., void fraction is 0) and red is the mixture of water and vapor (void fraction more than 0.6). The boiling occurs at the heated section which is positioned at the center for vertically. Although the coolant is the liquid water at the inlet section of the fuel bundle, it changes water and vapor due to the boiling by fuel rods. The void fraction near wall region is lower than the center region in the radial direction because the heat transfer rate at the near wall region is lower than that at the center region.

In addition, predicted void fraction distributions in the radial direction are shown in Fig. 2.2.4. Each predicted result on the radial void fraction distribution shown in Fig. 2.2.4 is the result of 1/6 cross-section of a fuel bundle. At the vicinity of the boiling position near the channel inlet, the void fraction shows the highest at the narrowest region of adjacent fuel rods. On the other hand, at the vicinity of the channel outlet, it shows the highest at the center region surrounded by three fuel rods. That is, the bubble generated by boiling moves from the narrowest region of adjacent fuel rods to the center region surrounded by three fuel rods along the flow direction. This tendency is the same result as an experiment.

3. Conclusion

In order to predict the water-vapor two-phase flow dynamics in a tight-lattice fuel bundle and to reflect them to

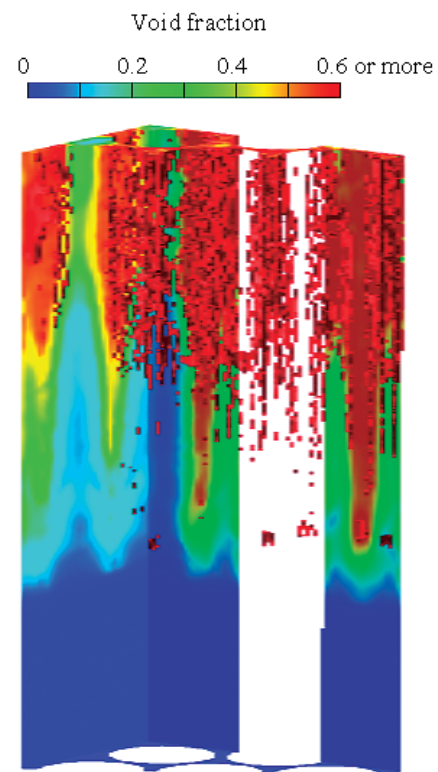


Fig. 2.2.3 Predicted void fraction distribution in the axial direction.

the thermal design of the advanced light-water reactor core, a large-scale simulation was performed under a full bundle size condition using the earth simulator. Details of water and vapor distributions under boiling heat transfer condition were shown numerically and the three-dimensional bubble dynamics in fuel channels were clarified quantitatively comparing with the experimental data.

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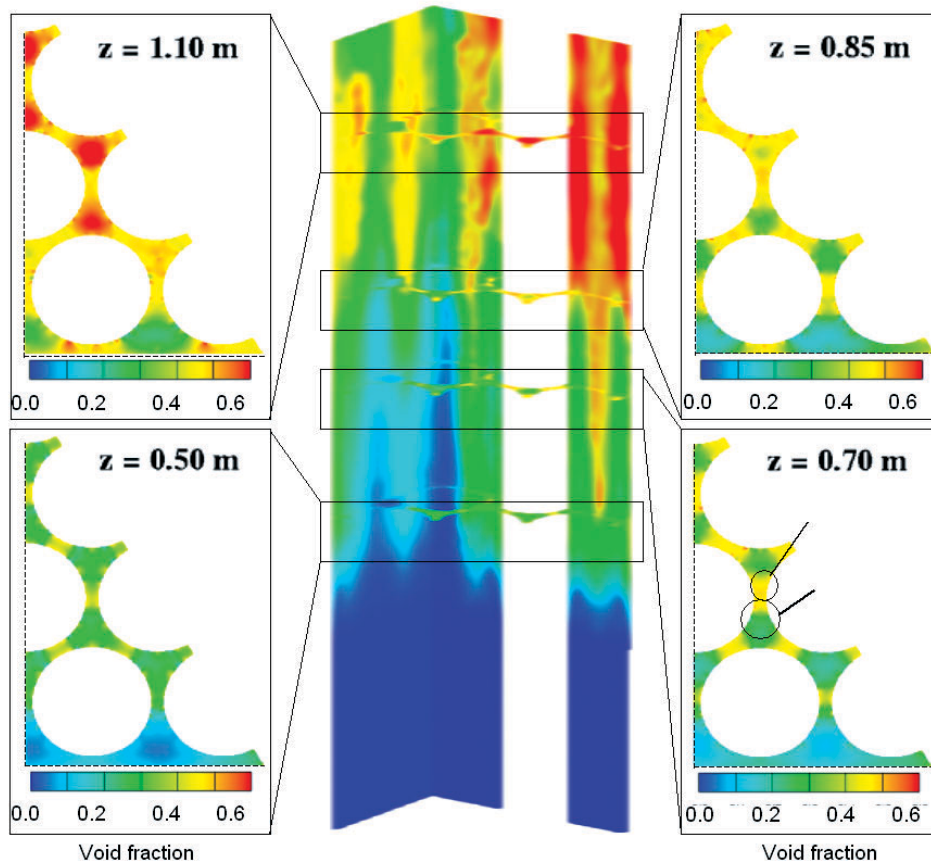


Fig. 2.2.4 Predicted void fraction distribution in the radial direction.

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将来型炉用燃料集合体内の熱流動に関する大規模シミュレーション

プロジェクト責任者

高瀬 和之 日本原子力研究開発機構

著者

高瀬 和之^{*1}, 吉田 啓之^{*1}

*1 日本原子力研究開発機構

原子炉の炉心内熱流動挙動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。本研究では、将来型軽水炉の燃料集合体内熱流動挙動を大規模シミュレーションによって地球シミュレータ上に再現することを最終目標としている。本報では、平成20年度に実施した稠密燃料集合体内水-蒸気二相流挙動の詳細予測結果について報告する。

キーワード: 大規模シミュレーション, 二相流熱流動, 沸騰現象, 原子炉, 燃料集合体