DEVELOPMENT OF DESIGN TECHNOLOGY ON THERMAL-HYDRAULIC PERFORMANCE IN TIGHT-LATTICE ROD BUNDLES: I – APPROACH AND VERIFICATION

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R&D project to investigate thermal-hydraulic performance in tight-lattice rod bundles for Innovative Water Reactor for Flexible Fuel Cycle (FLWR) has been progressed at Japan Atomic Energy Agency in collaboration with power utilities, reactor vendors and universities since 2002. (Uchikawa, 2005) The FLWR can attain the favorable characteristics such as effective utilization of uranium resources, multiple recycling of plutonium, high burn-up and long operation cycle, based on matured LWR technologies. MOX fuel assemblies with a triangular tight-lattice arrangement (about 1 mm gap between rods) are used to increase the conversion ratio by reducing the moderation of neutron. Increasing the in-core void fraction (about 70% on average) also contributes to the reduction of neutron moderation. The confirmation of thermal-hydraulic feasibility and the development of design technology are very important R&D items for the FLWR because of the tough situation for the feasibility due to the tight-lattice configuration and the thermal-hydraulic conditions.

In this paper, we will describe the master plan for the development of design technology and show the R&D achievements using large-scale test facility (37-rod bundle with full-height and full-pressure) and advanced numerical simulation technology.

In the large-scale tests, three test sections were used (Base test section (1.3mm gap width between rods), Gap effect one (1.0mm gap width) and Rod bowing effect one). The thermal-hydraulic characteristics and the feasibility were discussed from the experimental results.

The effects of mass velocity and inlet water temperature on critical power are similar to the previous 7-rod bundle experiments and the thermal performance is feasible against the design value of the FLWR.

- Fundamental characteristics of flow parameter impacting on critical power are similar between the Base test section and the Gap effect one. The thermal performance is almost the same between the two test sections.
- Effects of cross-sectional power distribution might be expressed by the power factor of subchannel where BT would occur.
- The critical power in the rod bowing test section is about 10% lower than that in the normal test section.
- Proposed evaluation method of pressure drop could yield good predictions in the tight-lattice bundle.

In the modeling engineering, detailed data-base has been accumulated to understand the governing phenomena and to verify advanced 3-D numerical simulation methods. The 3-D simulation methods have been developed using the advanced two-fluid model, the advanced interface-tracking method and the particle method. The verification of these methods is underway with several data-bases including those in this project. This paper indicated the verification of the advanced interface-tracking method using the 2-channel mixing experiment, and showed the application for the large-scale simulation in a tight-lattice rod bundle.

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ABSTRACT
R&D project to investigate thermal-hydraulic performance in tight-lattice rod bundles for Innovative Water Reactor for Flexible Fuel Cycle (FLWR) has been progressed at Japan Atomic Energy Agency in collaboration with power utilities, reactor vendors and universities since 2002. In this series-study, we will summarize the R&D achievements using large-scale test facility (37-rod bundle with full-height and full-pressure), model experiments and advanced numerical simulation technology.

This first paper described the master plan for the development of design technology and showed an executive summary for this project up to FY2005. The thermal-hydraulic characteristics in the tight-lattice configuration were investigated and the feasibility was confirmed based on the experiments. We have developed the design technology including 3-D numerical simulation one to evaluate the effects of geometry/scale on the thermal-hydraulic behaviors.

1. INTRODUCTION
An innovative water-cooled reactor concept named Innovative Water Reactor for Flexible Fuel Cycle (FLWR) is under development at JAEA in cooperation with Japanese utilities and Japanese light water reactor (LWR) vendors for the effective fuel utilization through plutonium (Pu) multiple recycling. (Iwamura, 2002a, 2002b) (Okubo, 2003) (Uchikawa, 2005) The reactor aims at achievement of a high conversion ratio more than 1.0 based on the well-experienced water-cooled reactor technology. Such a high conversion ratio can be attained by reducing the moderation of neutrons, i.e. reducing the water fraction in the core.

For the reduction of water fraction, the tight-lattice fuel rod arrangement is commonly adopted. The triangular lattice with a narrow gap between the fuel rods and/or the rods with a large diameter is usually considered. Especially for the BWR-type reactor design, increase in-core void fraction is another realistic technique to be used. In the current design, the tight-triangular-lattice with the gap around 1mm is required and the core void fraction more than around 70% as average needs to be realized. This cooling condition is considered to be a tough situation and the confirmation of thermal-hydraulic feasibility is recognized to be one of the most important R&D items for the FLWR.

JAEA started R&D project to investigate the thermal-hydraulic performance in tight-lattice rod bundles for the FLWR in collaboration with Power Company, reactor vendors and universities since 2002. In this series-study, we will summarize the R&D achievements using large-scale test facility (37-rod bundle with full-height and full-pressure), model experiments and advanced numerical simulation technology. This first paper describes the master plan for the development of design technology and shows an executive summary for this project up to FY2005.
2. TYPICAL CONFIGURATION OF FLWR AND SUBJECTS ON THERMAL-HYDRAULIC DESIGN

The representative 1,356 MWe class large-scale BWR-type core design (Iwamura, 2002a) is schematically shown in Fig. 1. The cross section of the fuel assembly is also shown in Fig. 1. In order to achieve negative void reactivity coefficients, the core is designed to be short and flat to increase neutron leakage from the core. Axial blankets with depleted uranium are also introduced to increase the conversion ratio and to reduce the void reactivity coefficients. The diameter of the fuel rods and the gap between rods are 13.7mm and 1.3mm, respectively. Y-shaped control rods with the follower structure are introduced for every three fuel assemblies. The core average void fraction is increased to 70 %. Resultantly, the effective volume ratio of the water to the fuel (Vm/Vf) is reduced extremely to about 0.17 in the present design, trying to attain a high conversion ratio. The conversion ratio of fissile Pu is about 1.05.

The void reactivity coefficient is evaluated to be negative value of –0.5x10^-4/k/k%void. The electric output of 1,356MWe is accomplished as in the case of ABWR. The core can be cooled by natural circulation because the core pressure drop is as low as 0.04MPa due to the low flow velocity and short core height. Therefore, the reactor internal pumps used in the ABWR can be eliminated in the present design. The number of fuel assemblies is 900 and each assembly has 217 fuel rods. The core outer diameter is about 7.6m.

From the thermal-hydraulic point of view, we encounter some difficulties. Figure 2 summarizes the subjects of FLWR thermal-hydraulic design. As mentioned above, the FLWR adopts the tight-lattice core with the relatively small gap width of about 1mm and the short axial length, double-flat-core with a high axial peaking. And the core flow rate is relatively low to realize the high void fraction, about 70% on the average. These conditions are tough situation for the core cooling and the thermal-hydraulic characteristics under such conditions are not well understood. To design the FLWR, the applicability of design equations/analytical methods against the design value in Fig. 2, i.e. MCPR (Minimum Critical Power Ratio), pressure drop and void fraction, is therefore one of the most important R&D items.

3. DEVELOPMENTAL TARGET AND MASTER PLAN FOR THIS PROJECT

The target of this project is to understand the thermal-hydraulic characteristics in the tight-lattice rod bundles and to develop a predictable technology for the thermal-hydraulic performance of the FLWR.

In previous studies, the applicability and improvement of CHF (Critical Heat Flux) correlation and pressure drop estimation have been performed using small-scale experiments with 7-rod and 4 x 5 rod bundles.(Yamamoto, 2002)(Kureta, 2003a)(Liu, 2003) Arai et al. (1990) developed a CHF correlation from the data-base of the 4 x 5 rod bundle (gap width between rods more than 1.5mm) and the correlation was reported to be applicable to the 7-rod bundle experiment by Toshiba Corp.(Yamamoto, 2002) However, the applicability of the Arai correlation was not necessarily reasonable against the 7-rod bundle one by JAEA (Kureta, 2003a). The correlation was not able to predict the effect of flow rate and axial power distribution observed in JAEA experiments.

To develop the CHF correlation, there is a problem to be clarified, that is so-called scale effects. The thermal-hydraulic characteristics near unheated flow shroud are considered to be different from those in the area far from the wall. Tamai et al. (2003) investigated the thermal-hydraulic behaviors in 37-rod bundle by a subchannel analysis code NASCA. They showed the behavior is affected in the area of two rows of rod adjacent to the shroud. The flow quality in the area of two rows of rod was lower than that in a center region. This means all rods in the 7-rod bundle configuration are impacted more or less by the effect of shroud wall. In an actual situation (217-rod bundle), most of rods are located in the area without the wall-effect. Therefore, the thermal-hydraulic behaviors for the FLWR should be investigated from the data-base with the rod bundle configuration having rod number more than around 37.

In the design study, verified design correlations exemplified in Fig. 2 are needed and previous correlations against an actual BWR-type reactor have been developed using a full-scale test facility. We also consider that the full-scale test is important to confirm the feasibility, however it is not easy to perform such test because the test loop with suitable electric power for 217-rod bundle is not
available at present and it is a high risk in the stage where we have no confidence the tight-lattice core can be cooled under reasonable flow conditions. Therefore, during a research stage, we adopt the design is mainly performed by analysis verified with various data-bases from scaled and/or model experiments.

Figure 3 shows the master plan in this thermal-hydraulic feasibility project. The project mainly consists of a large-scale thermal-hydraulic test and development of analytical methods named modeling engineering.

In the large-scale test, 37-rod bundle experiments are performed and the thermal-hydraulic data-base with no significant wall-effect will be acquired. Figure 4 shows schematics of the test loop and simulated fuel bundle. We have a plan to construct three test sections to investigate the parameter effects on the gap width between rods and the rod bowing. There has been no experimental evidence to be able to cool the tight-lattice core under such a large-scale configuration. We expect to clarify whether the core cooling is achieved under reasonable flow conditions.

In the modeling engineering, model experiments are conducted to investigate typical governing phenomena and 3-D two-phase flow simulation codes will be developed based on the understanding of governing phenomena (Yoshida, 2003). We suppose the governing phenomena related to the boiling transition (BT) are those shown in Fig. 5. The target of this development is to construct an analytical tool for evaluation of the geometry/scale effects on BT and for confirmation of the FLWR feasibility.

In the model experiments, we focus on the governing phenomena, i.e. annular-mist flow including BT, spacer effects and flow mixing/distribution between subchannels. The thermal-hydraulic behaviors related to the phenomena are investigated and data-bases for the verification of 3-D simulation codes will be accumulated. The data-base including time-fluctuation and 3-D distribution will also be obtained by a high-speed neutron radiography (NRG) (Kureta, 2001), tomography techniques and an optical method with a high-frequency laser.

In the development of 3-D two-phase flow simulation codes, detailed analytical methods are considered to be used depending on the governing phenomenon. An improved two-fluid model with interface-tracking, an interface-tracking method and a particle method are mainly applied for the annular-mist flow, flow mixing between subchannels and spacer effects, respectively. These methods will be verified by the data-base from model experiments and we will also confirm the methods can predict the data-base including time-fluctuation and 3-D distribution.

4. LARGE-SCALE THERMAL-HYDRAULIC TEST

Most important objective of the large-scale test is to resolve a fundamental subject whether the core cooling under a tight-lattice configuration is feasible. Test items in the large-scale thermal-hydraulic test are shown in Table 1. The characteristics of critical power and flow behavior are investigated under different geometrical configuration and
boundary conditions. The configuration parameter is the gap between rods (FY2004) and the rod bowing (FY2005).

Schematics of the test loop, simulated fuel bundle and axial power ratio are already shown in Fig. 4. It simulates the axial power distribution in double-bumped-flat-core. The rod diameter and the total heated length are 13.0mm and 1.26m, respectively. Grid spacers (hexagonal honeycomb shape, 0.3mm thickness and 20mm height) are used to tie up the rod. Axial locations of the spacer are also shown in Fig. 4. The test section is installed into the high pressure water circulating loop simulating the flow conditions expected in the FLWR operation. BT was detected using 250 thermocouples which measure the rod surface temperatures. BT (and critical power) is defined when the rate of wall temperature increase exceeds 45 K/s.

BT occurred at the exit of the upper high heat flux region (MOX fuel region) in the central area of the bundle. Elevation of the BT was the same as that of the 7-rod bundle test (Liu, 2004). Effects of mass velocity, inlet water temperature, exit pressure and cross-sectional local power distribution were parametrically investigated. Typical results are shown in Fig. 6. Critical power increases monotonously with increasing mass velocity and with decreasing inlet water temperature. These trends are similar to those of the 7-rod bundle tests (Liu, 2004). The rated power range of FLWR is also shown in Fig. 6. The FLWR is feasible on the critical power with about 30% margin under the coolant mass velocity around 700 kg/(m².s).

Effect of local power distribution is shown in Fig. 7. Local peaking factor of center-subchannel is defined as the ratio of power at center-subchannel to averaged value in the bundle. In this study, the center-subchannel is equivalent to the subchannel where the BT occurs, since all BT was detected at center-subchannel. The reason why the BT was detected at the center-subchannel even in peripheral-peak power distribution might be that (1) there was better cooling performance in the peripheral region due to the so-called wall effect and (2) the cross-sectional power distribution was relatively simple and the lowest power factor was up to about 0.92. Figure 7 shows that (1) when the peripheral power factor is higher than that of the central one, critical power is higher than that in the flat power distribution, (2) when the central power factor is higher than that of the peripheral one, critical power is lower than that in the flat power distribution, (3) critical power decreases monotonously with increasing the local peaking factor of center-subchannel and (4) effect of local power distribution might be expressed by the local peaking factor of subchannel where BT would occur.

One of the most important purposes of this test is to investigate the effect of gap width between rods on thermal-hydraulic performance. Figure 8 shows the results of the effect and the critical quality using the test section with 1.0mm gap width is compared with that of base test section with 1.3mm one. The BT takes place under almost the same critical quality under about 10% range. This result indicates that the thermal-hydraulic performance is almost the same between 1.3mm and 1.0mm gap widths.

The critical power characteristics under transient conditions have also been investigated. The transient conditions were derived from several system calculations for abnormal transients. We confirmed that the critical power under transients is identical to that under steady state and the BT was not detected even under severe conditions.

### Table 1. Test items in large-scale thermal-hydraulic test

<table>
<thead>
<tr>
<th>Type of test section</th>
<th>Base case</th>
<th>Gap effect</th>
<th>Rod bowing</th>
</tr>
</thead>
<tbody>
<tr>
<td>FY</td>
<td>2003</td>
<td>2004</td>
<td>2005</td>
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</tbody>
</table>

<table>
<thead>
<tr>
<th>Critical power</th>
<th>Steady-state</th>
<th>Transients</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parameter: Flow rate, Fluid temp., Pressure, Local peaking</td>
<td>Flow reduction, Power increase, Pressure increase/decrease, Increase of fluid temp.</td>
<td></td>
</tr>
</tbody>
</table>

| Flow characteristics | Pressure drop, Void fraction, Form loss due to spacers |

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**Fig.6. Effect of mass velocity and inlet fluid temperature on critical power**

**Fig.7. Effect of local power distribution on critical power**

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Evaluation method of pressure drop in the tight-lattice bundle is investigated by comparison between measurements and calculations. The calculation was carried out using TRAC-BF1 code and Martinelli-Nelson’s correlation (Martinelli, 1948) for calculating two-phase multiplier. Morooka et al. (2003) reported that the pressure drop in the tight-lattice 7-rod bundle could be predicted using the similar method. Figure 9 shows the comparison between measured and calculated pressure drops under a wide condition (Pex = 3 - 9 MPa, G = 300 – 1000 kg/(m²s)). The figure clearly shows that this evaluation method could yield good predictions of pressure drop in the tight-lattice bundle with an uncertainty of ±10%.

The rod bowing effect is also one of most important subjects for the feasibility of tight-lattice rod bundles. Prior to design the test section, we had evaluated the impact of deformation on critical power under several configurations by using subchannel code NASCA (Ninokata, 1997) which can predict the critical power in the base test section within the error of ±10%. And we selected most severe configuration from the analyses. Figure 10 shows the outline of the fabricated rod bowing effect test section. The configuration is the same as that of the base test section except for the rod bowing. The center rod is bowed at the middle of the upper high heat flux region. The bowing rod attaches to adjacent two rods like the right-top drawing. The configuration was measured by X-ray CT (Mitsutake, 2005) and we confirmed the dimension is within a tolerance. More detailed discussion on the X-ray CT measurements and subchannel analyses based on the measurements are showed in this series study.

BT occurred also at the exit of the upper high heat flux region in the central area of the bundle. The clad temperature at the contact region was also oscillated but the amount of oscillation was smaller than that at the BT region. Effects of mass velocity, inlet water temperature, exit pressure and cross-sectional local power distribution were similar to those of the base and the gap effect test sections and however the critical power was lower than that in the base test section. Figure 11 compares the critical power with rod bowing to that without one. The rod bowing reduces the critical power up to about 10%. The current FLWR design has 30% margin for the critical power and we consider the design is feasible even under the bowing condition. The transients in FLWR reduce the margin but ΔMCPR was within about 0.1 (Okubo, 2001) and the feasibility is maintained.
5. DEVELOPMENT OF 3-D TWO-PHASE FLOW SIMULATION METHODS

As we stated in Chapt. 3, detailed analytical methods are adopted to perform the 3-D two-phase thermal-hydraulic simulation. Figure 12 shows the overall outline of analytical procedure including the detailed methods. Although the main method for the reactor design calculation is so-called subchannel code ((A) in Fig. 12), the detailed methods are developed to evaluate the effects of geometry/scale as described in Chapt. 3. The outline for the detailed methods has been reported by Yoshida et al. (2004) and several advances will be presented in this series study.

We have developed several 3-D simulation codes with the advanced methods and some codes have been converted to massive parallel architectures. The verifications are being advanced with several model experiments. Figure 13 shows one example for the interface-tracking method. The experiment is a 2-channel steam-water mixing one where the test channel consists of two parallel subchannels (8 × 8 mm square cross sections) and an interconnection. The gap clearance of the interconnection is 2 mm and the horizontal/vertical lengths are 4 mm and 80 mm, respectively. The fluid mixing was observed at the interconnection. A slug bubble behavior around the interconnection is shown. Once the top of an ascending steam slug reaches around the center height of the interconnection, a small amount of steam starts to flow toward the other channel. Then the tip of stretched steam flows through the interconnection. The similar behavior is well predicted. The cross flow rate was also well predicted.

We also applied the detailed methods to large-scale simulations where the geometry and dimensions simulate a part of tight-lattice FLWR fuel bundles. Figure 14 shows one example with the interface-tracking method for the comparison of void fraction distribution at a horizontal plane. The prediction shows a thin water film on rods and a higher void fraction in the subchannel region surrounded by three rods. A lower void fraction is recognized at the narrow gap between rods. Figure 14(b) shows an example of experiments obtained by an advanced neutron radiography technique (Kureta, 2003b). The characteristics of void fraction distribution correspond well to the prediction.
6. SUMMARY

An innovative water-cooled reactor concept named the FLWR is under development at JAEA in cooperation with Japanese utilities and Japanese LWR vendors for the effective fuel utilization through Pu multiple recycling. JAEA started the R&D project to investigate thermal-hydraulic performance in tight-lattice rod bundles of the FLWR since 2002. This first paper described the master plan and showed an executive summary for this project up to FY2005.

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