Experimental Study on Reduced Moderation BWR with Advanced Recycle System (BARS)

K. Hiraiwa*[1], K. Yoshioka[1], Y. Yamamoto[1], M. Akiba[1], M. Yamaoka[1], N. Abe[1], J. Mimatsu[2]
[1] *Toshiba Corporation, 8, Shinsugita-Cho, Isogo-Ku, Yokohama 253-8523, Japan*
[2] *Mechanical and System Engineering Department,Gifu University,Gifu 501-1112, Japan*

Experimental study has been done for reduced-moderation spectrum boiling water reactor named BARS (BWR with Advanced Recycle System). The critical assembly experiment for triangular tight uranium lattice has been done in TOSHIBA critical assembly (NCA). Experimental method based on modified conversion ratio was adopted to evaluate the void reactivity effect. Void fraction was simulated by formed polystyrene in this experiment. The measured void coefficient for tight uranium lattice agreed with calculation. The thermal hydraulic test study has been done to study the coolability of BARS lattice. Visual test and high-pressure thermal hydraulic test have been done as the thermal hydraulic test. Visual test has indicated the flow hehavior for BARS lattice is same as that of current BWR. The high-pressure thermal hydraulic test has indicated the applicability of modified Arai's correlation to the BARS lattice.

KEYWORDS: BWR, critical assembly, reduced moderation, NCA, BEST

1. Introduction

The reduced-moderation spectrum BWR with high conversion property features higher utilization of uranium resources, multi-recycling usage of plutonium and flexibility in loading the extracted long-lived radioisotopes, it can realize a sustainable nuclear energy [1]. The advanced recycle system means the combination of reduced moderation BWR, dry reprocessing and vibro-packing of MOX fuel fabrication. This system simplifies reprocessing and MOX fuel fabricating process and reduces related backend cost. The reduced moderation spectrum condition in BWR is obtained through triangular tight fuel rod lattice configuration and higher void fraction. These feature cause some design problem on nuclear and thermal hydraulic design. According to resolve the problem, critical assembly experiment and thermal hydraulic test have been done using tight lattice configuration.

The critical assembly experiment on uranium lattice has been carried out on Toshiba nuclear critical assembly (NCA) as the first step of tight lattice critical experiments. Since there are few benchmarking studies on reduced-moderation light water lattice such as BARS core, the accuracy of nuclear design method should be validated.

The thermal hydraulic test focuses on two subjects related with the cooling mechanism of tight lattice. The first subject is elucidation of mechanism of the boiling transition (BT) on tight lattice. The single channel visible experiments were planned to investigate the mechanism of the BT on tight lattice. The second one is establishing of the BT correlation equation on tight lattice bundle. Mini tight lattice bundle experiments have been planned to study the correlation. One of the important purposes of these experiments is to verify the existing the BT correlation equation such as Arai's equation on tight lattice bundle. These

* Corresponding author, Tel. +81-45-770-2047, FAX +81-45-770-2179, E-mail: kouji .hiraiwa@toshiba.co.jp

experiments have been done in Toshiba's high temperature and high-pressure BWR experimental facility for stability and transient test (BEST).

2. Nuclear Design Concept

In the design of BARS core and fuel, a tight lattice fuel assembly was adopted where a water to fuel volume (W/F) ratio is about 0.5 to achieve a fast neutron spectrum.

It is well known that the void reactivity coefficient in LWRs has the tendency to be less negative in harder neutron energy spectrum. Then, a new core concept has been introduced in order to improve the void reactivity coefficient under the restriction of core diameter by adopting a neutron streaming channel described below.

Partial fuel assemblies whose active fuel length is about half of the normal fuels are arranged by one-third of the whole core as shown in Fig.1.

When void fraction increases, the streaming channel located at the upper part of the partial assembly will enhance axial leakage of neutrons which have leaked out through the side of the normal assemblies and the top of the fuel bundle of the partial assemblies as shown in Fig.2. The cavity-can in the streaming channel not only provides a streaming path for the leaked neutrons from the fuel but also suppresses softening of the neutron energy spectrum by expelling water from the channel.

Since there exist various location of streaming channel, the sensitive study for the location has been done. The layered location and the checkerboard location have been analyzed by three-dimensional Monte-Carlo analysis. The results show that streaming channel enables negative coefficient and the checkerboard location is favorable compared with layered location, because void coefficient becomes more negative and the conversion ratio does not become worse.

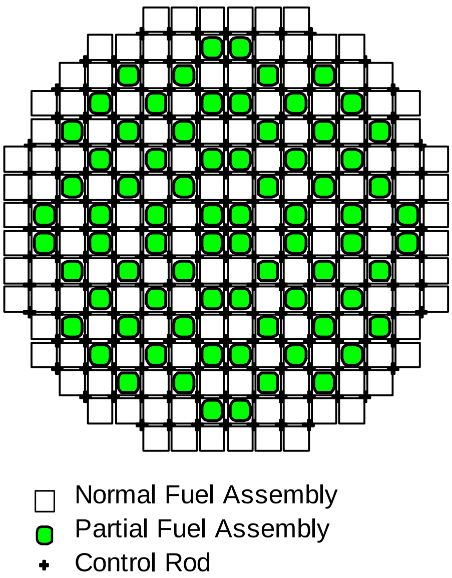

Fig.1 BARS core layout

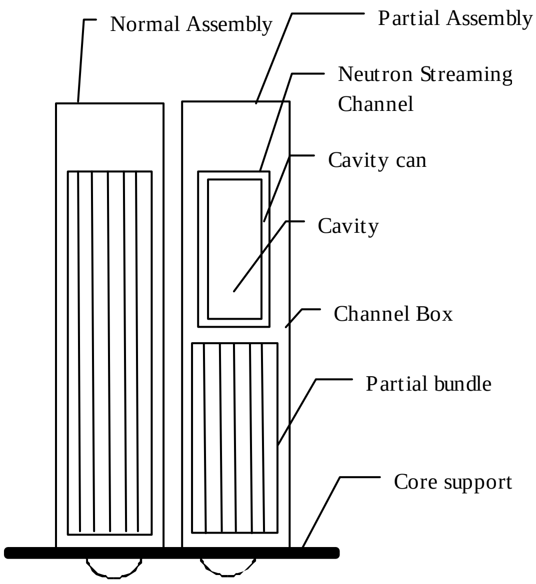

Fig.2 Vertical View of Fuel Assembly

3. Critical Assembly Test

Several experimental studies have been made on high conversion LWR (HCLWR) with triangular tight lattice [2,3]. Chawla et al. performed experiments simulating an H_2O voidage of 42.5% and estimated k-infinity void coefficients on the basis of the measured changes with voidage of the various reaction rate ratios. The BARS core has higher H_2O voidage of 60% to achieve the conversion ratio of 1.0 more than that for HCLWR. The neutron spectrum is significantly different from that in a current LWR, and therefore, applicability of a nuclear design method and nuclear data library should be validated. Doppler coefficients are also important factors for a nuclear design; however, there are a few measurements for LWR. Therefore the Doppler coefficients for the BARS core should be validated as well as the void coefficients.

In the BARS core, resonance neutron contributes to neutron absorption and fission more than current LWRs. The neutron is also closely related to reactivity coefficients such as void coefficients and Doppler coefficients.

The BARS core features triangular tight lattices, streaming channel for keeping void coefficients negative and MOX fuels. This study focuses on reactivity coefficients for a triangular tight lattice. In fact, MOX fuels are used for the BARS core design, but this study focuses on a UO_2 core as a first step for nuclear design method validation. The validation for UO_2 fuels is also important for MOX fuels because ^{238}U capture reaction significantly affects reactivity coefficients for MOX fuels. In the next stage, MOX fuel critical experiments are planned.

3.1 Void Coefficient Measurement

NCA is a slightly enriched, uranium-fueled, light-water-moderated critical assembly, which has been utilized to validate LWR nuclear design and to develop a new fuel design concept.

Fig.3 shows the core configuration for this study. The test assembly consisted of a void-simulated zone and a driver zone. The void simulated zone is a triangular tight lattice consisting of 3.9wt% and 4.9 wt% UO_2 fuels with a pile of polystyrene plates for simulating void fraction. The rod pitch of the void simulated zone is 13.5mm. The polystyrene plates have been employed to realize hydrogen concentration corresponding to 0%, 35% and 60% void fraction of a typical BWR operating condition. The driver zone surrounding the void-simulated zone used to keep criticality shapes a square lattice consisting of 2wt% UO_2 fuels with cold water. The rod pitch of the driver zone is 15.2mm. The volume ratio of fuel to moderator for cold water, hot 0% void, hot 35% void and hot 60% void are 0.59, 0.44, 0.28 and 0.18, respectively. For these four cases of void fraction, modified conversion ratios of fuel rods in the center at the void simulated zone have been measured [4,5]. Gamma-ray spectra from the fuel rods have been measured with an HP-Ge detector after irradiation of 20 Watts x 30 minutes.

The measurement error in modified conversion ratio has been estimated to be 2~3%. The main components of the error corespond to the uncertainty in gamma-ray spectrum analysis, the error in gamma-ray self shielding of a pellet and the statistical error.

Fig.4 shows finite multiplication factors (k*s)[10] derived from measured conversion ratios based on modified conversion ratio11-16. The error of k* has been estimated to be ~1%. Void coefficients derived from the difference of k* in cold condition and hot 60% void condition was 0.153± 0.014%dk/k/%void. On the other hand, calculated void coefficient was 0.167%dk/k/%void and agreed with the measured one within the measurement error.

Consequently, current nuclear design method, MCNP4A with JENDL3.2, is applicable to BARS core design for UO_2 fuel.

3.2 Doppler coefficient measurement [6]

The test equipment for Doppler measurement was fabricated. For Doppler measurement, it is important to raise the temperature of fuel pellets as high as possible. Moreover, the size of equipment must be suitable for the core size in NCA.

The size of the test equipment is f15cm x H 150cm, and the equipment is designed to load into the NCA core. A micro heater consisting of a nichrome wire with a stainless steel sheath is used to raise pellet temperature. Fine flex fiber composed of silica-alumina fiber is used as a heat insulator. The equipment has been designed to raise pellet temperature up to 900°K. The temperature-raising test revealed that the pellets reached the temperature of 900°K without any trouble.

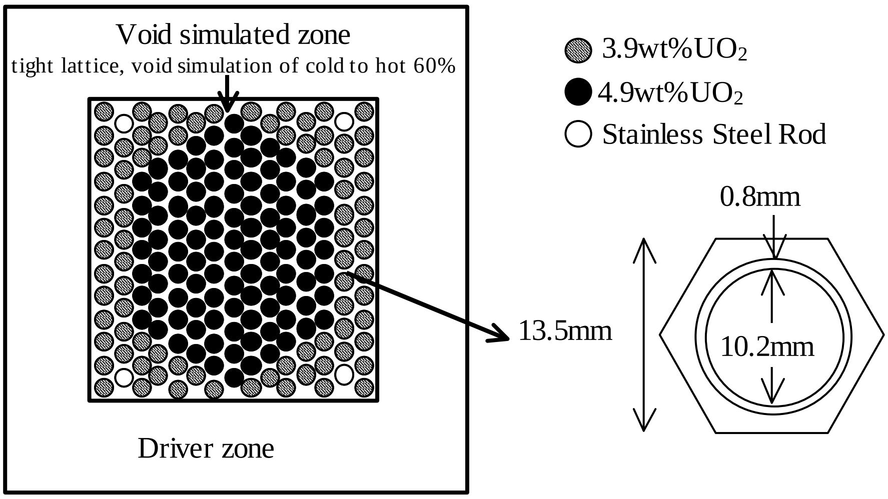

Fig.3 Experimental core cinfiguration for critical assembly test

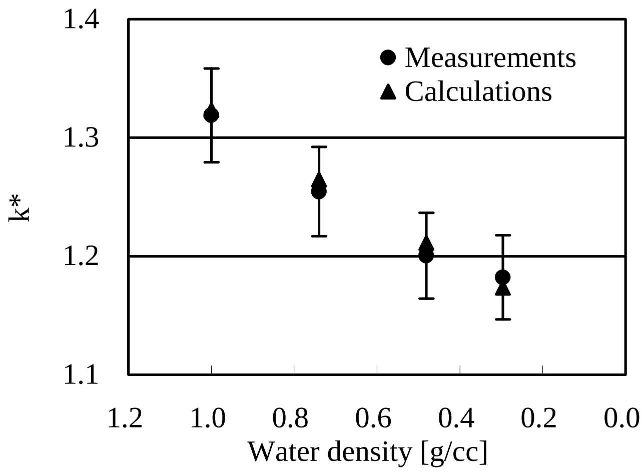

Fig.4 Finite multiplication factor k*

4. Thermal Hydraulic Test

In the BARS core, a tight lattice fuel design is required because the conversion ratio needs to be improved to about 1. However, as a fuel rod gap becomes narrower, thermal-hydraulic performance, especially critical power, becomes worse. Therefore, the thermal power of the BARS reactor core could be influenced largely by thermal-hydraulic performance of the tight lattice fuel. The fuel is designed as triangular lattice whose rod gap is 1.3 mm. In such a tight lattice bundle, there are scarce critical power test data and critical power correlation applicable to critical power prediction.

Then, it was planed to two tests:

(1) Visualization test

(2) High-pressure thermal-hydraulic test

4.1 Visualization Test

The purpose of the visualized tests is to observe flow behavior in tight lattice bundle and to compare observed results with current flow pattern map. Fig.5 shows the test section of the visualized test facility. The test section simulated single sub-channel of tight lattice bundle. The test section consisted of a preheating part and a visible part. The glass rods are included in the visible part in order to observe the two-phase flow behavior from the rear of the rod. Test section adopted atmospheric pressure, wide mass flux range (400 ~ 1500 kg/m^{2}s).

The flow behavior was very stable. The flow fluctuation as slug flow was hardly observed except in low flow rate (400kg/m2s). In the test channel, it has been observed that liquid phase has been rich in narrowest area of the rod gap and vapor has concentrated in a wide area. It is thought from these observation results that the narrow gap area is not severe location for rod cooling. Furthermore, these observation results were compared with Hewitt-Roberts flow pattern map5). Figure 6 shows the compared results. Hewitt-Roberts flow pattern map is based on observation data of low-pressure air-water and high-pressure steam-water flow in 10-30mm diameter of vertical tubes. The axes represent the superficial momentum fluxes of the liquid and vapor phases respectively. The hydraulic diameter of current BWR bundle is about 10mm. Therefore, this flow map may be the typical flow map

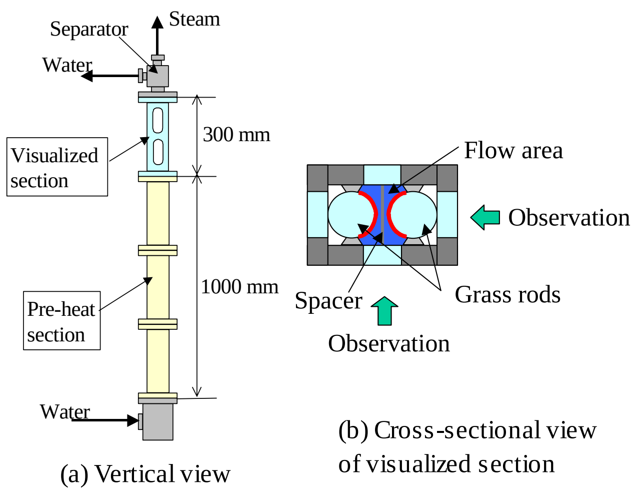

Fig.5 Schematic view of visualized test section

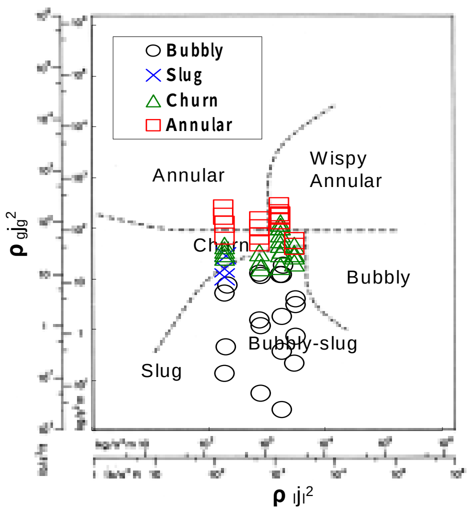

Fig.6 Comparison observed results with Hewitt-Roberts flow pattern map

for current BWR bundle. Test results are mostly in agreement with the Hewitt-Roberts flow pattern map. In the map, the area of definite slug flow is only in low flow rate. The observation results of this point agree with the flow pattern. Therefore, the flow behavior of tight lattice bundle is the same as that of current BWR bundle.

4.2 High-pressure Thermal-hydraulic Test

4.2.1 Fundamental Thermal Hydraulic Test

The purpose of the high-pressure thermal-hydraulic test is to make a database of the critical power performance of the tight lattice bundle whose rod gap is about 1.3 mm. In our previous study, we had carried out the critical power test with the tight lattice bundle and found that critical power correlation, developed by Arai, could be applied to predicting critical power of tight lattice bundle [7,8]. However, the previous critical power test was performed with the longer heating length than that of BARS core in this paper. Therefore, the thermal-hydraulic tests were planned to enhance the thermal-hydraulic database for the tight lattice bundle, to verify the applicability of Arai's correlation to the BARS core, and to develop a more accurate correlation for predicting critical power. Moreover, the two-phase flow instability test, transient boiling transient test and pressure drop test were planned in order to design the BARS core from the thermal-hydraulic viewpoint.

Fig.7 shows the cross-sectional view of test assemblies. As shown in Fig.7, it has been planned to fabricate two types of the test bundle. One is a 7-rod test bundle with a hexagonal channel box. The other is a 14-rod test bundle with a rectangular one. The purpose of the 7-rod bundle test is to survey the rod gap effect and heating length effect on the critical power and pressure drop. On the other hand, the purpose of the 14-rod bundle test is to check the critical power for various radial power distributions.

In our critical power test, pressure, inlet water temperature and flow rate were set to the programmed levels first. Then, the bundle power was raised step by step in steps of a small magnitude. Critical power was defined as a power when the rod surface temperature jumped by 14 centigrade from the temperature under nucleate boiling conditions. Test conditions were planned to be as follows:

Pressure: 5~8 MPa

Mass flux: 500 ~ 2000 kg/m2s

Inlet subcooling: 25~100 kJ/kg

The critical power tests of the 7-rod test bundle completed.

4.2.2 Critical Power Correlation

In our previous study (Yamamoto, 2002), it was mentioned that the critical power correlation, developed by Arai et al., was applicable to prediction of the critical power of tight lattice bundle [8].

Then, Arai’s correlation was applied to our new data. Fig.8 shows the measured critical power versus calculated critical power by Arai’s correlation. Almost all the calculated critical power data for 0.8-mm and 1.3-mm gap test bundles were larger than the measured data. On the other hand, the calculated critical power data for 1.8-mm gap test bundle were the same as the measured data except for larger power data than 0.4 MW.

As a result, it was found that Arai’s correlation had insufficient applicability to predict the critical power of the tight lattice bundle. In particular, the rod gap dependence on the critical power must be improved.

To understand these results, the subchannel analysis was performed. The analysis showed that the mass flux at center subchannel decreased greatly downstream of the test bundle. On the other hand, the analysis showed that the mass flux at center subchannel did not decrease greatly downstream of the test bundle. In our previous test bundle, there were 6 dummy rods between channel tube and heater rods. It was supposed that these dummy rods contributed to unification of the coolant. These subchannel analysis results showed that the flow distribution had a large effect on the critical power performance in rod bundle.

Based on these results, it was found that the consideration of the flow distribution was necessary to predict the critical power accurately. Moreover, these results suggested that the critical power would become larger if the flow distribution were uniform.

The critical power tests using the 7-rod test bundle with narrow gap were performed in order to augment the database of the critical power performance applied for BARS core design. The following conclusions are obtained from this study.

(1) The dependencies of the flow parameter on the critical power for the tight lattice bundle are similar to those for the conventional BWR fuel bundle.

(2) The calculated critical power by Arai's correlation does not agree with experimental data for 1.6-m length bundle, whereas good agreement was obtained for the previous test for 3.7-m length bundle. Therefore, further improvement of the critical power correlation is needed for the tight lattice bundle.

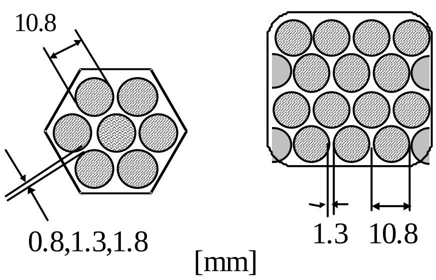

Fig.7 Cross-sectional view of test assemblies

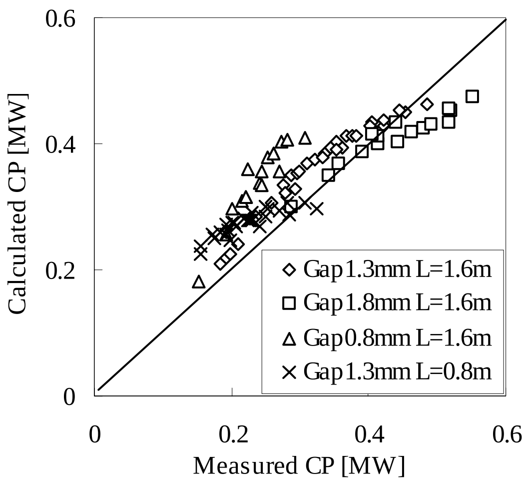

Fig.8 Measured critical power versus calculated critical power with Arai's correlation

5. Conclusion

Critical assembly tests for BARS based on modified conversion ratio method at NCA facility has been done. The experiment showed that nuclear design method on uranium core had adequate accuracy, because the conversion ratio and multiplication factor k* by experiment showed good agreement with the analysis.

Thermal hydraulic test study has been done as visualization test, high pressure thermal-hydraulic test. Visualization test has been planned to investigate the boiling transition behavior in the narrow gap bundle. High pressure thermal-hydraulic test has shown the thermal hydraulic performance and critical power for tight lattice. The flow parameter on critical power showed similar performance to those of conventional BWR fuel bundle. On the other hand, the calculated critical power did not showed good agreement with Arai's correlation.

Addition to the above discussion, the CCFL (Counter Current Flow Limitation) performance will be the important issue in LOCA. Therefore, the CCFL experiment has been planned as the part of thermal hydraulic test study.

Since the existence of streaming channel is important concept to realize the negative void coefficient, core design study assuming streaming channel has been being concurrently advanced. The evaluation of the boundary failure for streaming channel will be one of the important study item in the design study.

Acknowledgements

This project is funded by the Institute of Applied Energy (IAE), Ministry of Economy, Trade and Industry (METI), Japan.

References

1) Yokoyama, T., et al: "Study on Breeding Characteristics of Fast Spectrum BWR," International Conference on the Physics of Nuclear Science and Technology, Islandia Marriott Long Island, Long Island, NY, October 5-8(1998).Enter first reference here.

2) Chawla, R., et al.: "Reactivity and Reaction Rate Ratio Changes with Moderator Voidage in a Light Water High Converter Reactor Lattice," Nuclear Technology, Vol.67, p.360-p.380 (1984).

3) Nakajima, K., et al.: "Measurements and Analyses of the Ratio of 238U Capture to 235U Fission in Low-Enriched UO2 Tight Lattices," J. of Nuclear Science and Technology, Vol.31, p.1160-p.1170 (1994).

4) Yoshioka, K., et al.: "Determination of Void Coefficient from Modified Conversion Ratio Measurements," Proc. of 2000 Fall Meeting of the Atomic Energy Society of Japan, (2000).

5) Ueda, M.: "Determination of Multiplication factors Using Experimental Lattice Parameters of Cluster-Type Fuel Lattices," J. of Nuclear Science and Technology, Vol.12, p. 229-p. 242 (1975).

6) Mitsuhashi, I., et al.:"Doppler Coefficient Measurement Method in an LWR Pu Lattice Employing Modified Conversion Ratio,"Proc. of 2000 Fall Meeting of the Atomic Energy Society of Japan , (2000).

7) Yamamoto, Y. et al., 2002, "Critical Power Performance of the Tight Lattice Fuel Bundle", Trans. At. Energy Soc. Japan, Vol. 1, No. 3 (in Japanese).

8) Arai, K. et al., 1990, "Critical Power Characteristics of a High Conversion Boiling Water Reactor", IAEA Technical Committee on Technical and Economical Aspects of High Converters, Germany.