Study on reduced-moderation spectrum BWR with an Advanced Recycle System

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ABSTRACT

A study is being evolved on neutronic and thermal characteristics of a fast spectrum BWR core with the tight fuel lattice for an innovative fuel cycle system named BARS (BWR with an Advanced Recycle System) aiming at Pu multi-recycling and MA's (Minor Actinides) burning. The BARS core has unique characteristics: its neutron spectrum is very hard through tight fuel lattice with fuel pin gap of 1.3mm and it has neutron streaming channel to keep negative void reactivity on top of one-third of the fuel bundles. Therefore, the neutronic benchmark tests and the thermal-hydraulic tests were planned to measure void reactivity effects, neutron streaming, and the thermal-hydraulic performance of the tight lattice bundle. Through our core design study and benchmark tests, we will make clear the feasibility of the conceptual design of BARS core.

1. INTRODUCTION

The LWR will play an important role in nuclear power generation for a longer time than expected before owing to the delay of commercialization of the LMFBR. It is one of the promising candidates for recycling various actinides such as U, Pu, and Minor actinides (MAs) because it is well demonstrated. In particular, BWR has an advantage in breeding or high conversion of fissile plutonium as boiling in water decreases the amount of moderator to induce a fast neutron spectrum. The fast spectrum is also advantageous in multi-recycling of Pu because the production of higher isotopes is suppressed. Moreover, it helps to burn fuels with recovered MAs and low decontaminated FPs such as rare earth's (REs), which endows the recycling system with nuclear proliferation resistance.

Taking into account the advantage of the fast spectrum BWR, an innovative fuel cycle system named BARS (BWR with an Advanced Recycle System) is proposed as a future fuel cycle option aiming at enhanced utilization of uranium resources and reduction of radioactive wastes [1, 2, 3]. As shown in Fig. 1, in the BARS, the spent fuel from conventional LWRs is recycled as a MOX fuel for a BWR core with the fast neutron spectrum through the oxide dry-processing and the vibro-packing fuel fabrication.

The fast neutron spectrum is obtained through triangular tight fuel lattice, which tends to shift coolant void reactivity more positive. Therefore, neutronics design efforts were made to keep it negative under the condition of target core characteristics. As a result, a core concept with neutron streaming channel aiming at negative void reactivity was proposed.

Verification of neutronics calculation method is required since the BARS core has the fast neutron spectrum through tight fuel lattice and reactivity effect due to neutron streaming phenomena as shown in the next section. Therefore, we have a program to perform benchmark tests to verify neutronics calculation method. In the tight lattice core, the gap between the rods must be narrow, about 1.3 millimeters, for achieving high conversion ratio. As the gap between the rods becomes narrower, the critical power performance may be worse due to smaller flow area. Because there are few critical power data for such tight lattice, the critical power measurement test should be performed in order to get the database. In this paper, the BARS core design is summarized and the program of the measurement tests is presented both for verification of the neutronics calculation method and thermal characteristics.

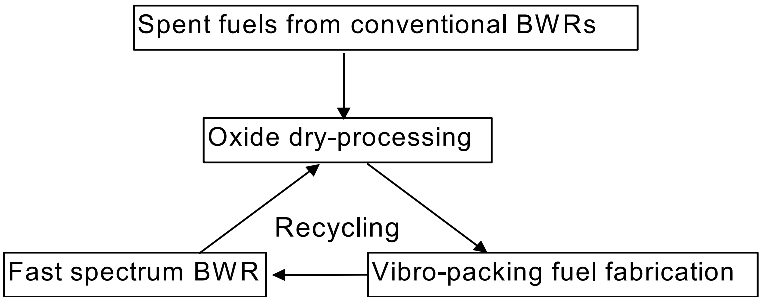


Fig. 1 Concept of BARS (BWR with Advanced Recycle System)

2. BARS CORE DESCRIPTION

At first, BARS core concept is summarized below. Table I shows typical BARS core and fuel specifications. To achieve a fast neutron spectrum, a tight lattice fuel assembly was adopted where a water to fuel volume (W/F) ratio is about 0.5. It is well known that the void reactivity coefficient in fast spectrum BWR core has the tendency to be positive. Then, a new core concept was proposed in order to improve the void reactivity coefficient under the restriction of core diameter by adopting a neutron-streaming channel.

Fig. 2 shows core profile of the BARS core in a large BWR plant (reactor thermal output of 3926MWt, core diameter of 5.2 m, core height of 1.6 m). The diameter of the BARS core should be approximately the same as that of conventional BWR cores to suppress the plant construction cost. The height is less than half of that of the conventional BWR, and 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. 3.

When void fraction increases, the streaming channel 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. 4. 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 spectrum by expelling water in the channel.

Figure 5 shows the horizontal cross sectional view of a fuel assembly. The follower above the control rods and water removal plates attached on the outer side of channel box expels surplus water and helps to decrease W/F ratio. These concepts as well as a large size channel box (□30cm) enable W/F ratio as low as 0.49 in the BARS core. The average void fraction is designed to be about 60%.

Table I Specifications of BARS Core

Item	Unit	BARS core	Conventional ABWR
Power	MWe	1356	1356
Core equivalent diameter	m	5.2	5.2
Core height	m	1.6;normal fuel 0.8;partial fuel	3.7
Number of fuel assemblies	-	208;total 132;normal fuel 76;partial fuel	872
Number of fuel pins	/assembly	658	60
Pin lattice type		Triangle	Rectangular (8×8)
Pin diameter	mm	11.2	12.3
Cladding thickness	mm	0.3(SUS)	0.86(Zr)
Pin gap	mm	1.3	4.0
Pin pitch	mm	12.5	16.3
Bundle pitch	mm	317	155
Ratio of flow area to fuel area		0.49	3.1

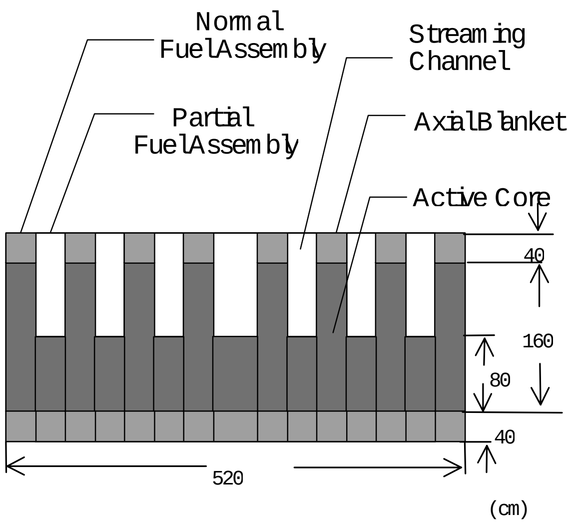


Fig. 2 Vertical View of BARS Core

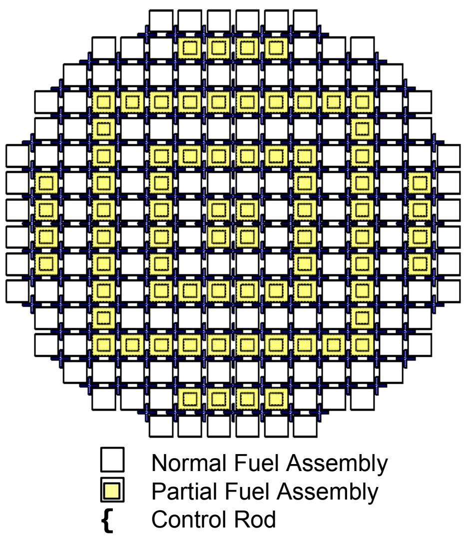


Fig. 3 BARS Core Layout

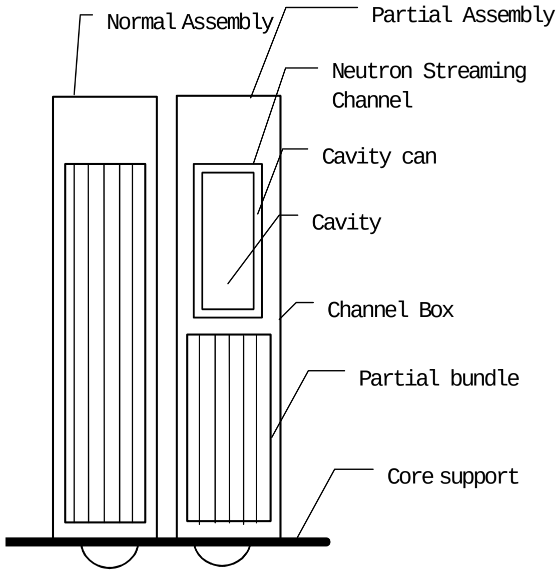


Fig. 4 Vertical View of Fuel Assemblies

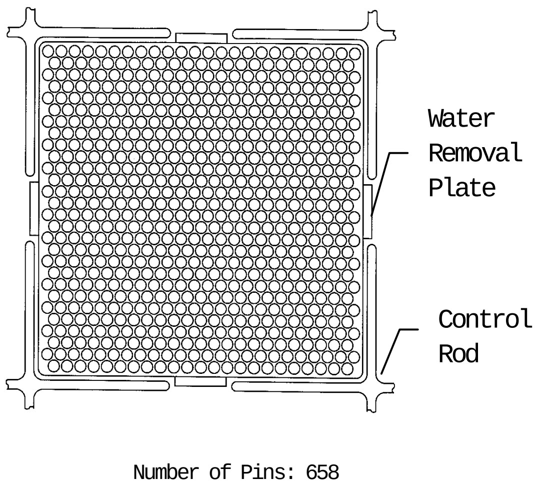


Fig. 5 Horizontal Cross Section of Fuel Assembly

3. CHARACTERISTICS OF BARS CORE

Table II shows neutronic characteristics of the core loaded with MOX fuel accompanied with no MAs and REs. The burnup calculation was done by using the conventional diffusion burnup code SRAC [4], taking into account the fuel bundle heterogeneity. The void reactivity coefficient was analyzed by the continuous energy Monte Carlo code MVP [5]. The cross section library used was based upon JENDL-3.2, which is attached to each code.

The operation cycle length is 1year, and the number of refueling batch is four. The Pu enrichments of load fuel were determined as shown in Table III and Fig. 6 so that, criticality is attained at the end of the equilibrium cycle (EOEC). The enrichment of outermost assemblies has been set higher than the other fuels aiming at radial power flattening. The enrichment is varied in the axial direction: The enrichment located in the lower region of fuel assemblies near the axial core center is about 20% to 30% lower than those above and below the region. The breeding ratio is 1.04 with the average discharge burnup of 44GWd/t.

The void reactivity coefficient is negative through the equilibrium cycle. It was found that the axial enrichment distribution leads to more negative void reactivity than the uniform distribution, as described below. Since there is no fuel in the upper half of the partial fuel assembly and the neutrons tend to leak from the streaming channel, the neutron flux in the upper-half of the normal assembly is lower than that in the lower-half of that. By placing a low enrichment zone on lower half of fuel assembly, it is possible to flatten the axial neutron flux distribution. Therefore, the neutron streaming effect by void fraction changes remarkably increases due to the increase of the neutron flux in the upper-half of the normal assembly adjacent to the streaming channel. The axial enrichment distribution is also advantageous from the viewpoint of power flattening.

As the next step, core design is evolving for the core loaded with MOX fuel accompanied with MA(minor actinides) and REs (rare earths) recycled by use of oxide-dry processing method.

Table II Neutronic Characteristics of BARS (Equilibrium cycle)

Item	unit	BARS
Refueling scheme	-	1 year ×4 batches
(1)Burn-up reactivity swing	%Δk/k	1.6
(2)Breeding ratio		1.04
(3)Core average discharge fuel burn-up	GWd/t	44
(4)Maximum linear heat rate	W/cm	400
(5)Void reactivity coefficient	10^{-4}Δk/k/%void	-0.1
(6)Heavy metal inventory	t	132

□Load fuel : MOX without rare earths

Pu fissile / Pu total = 0.575, U enrichment = 0.02

Table III Pu Enrichment Distribution

(Weight %)

	Low enrichment zone	Other zones
Inner Assembly	11	14
Outermost Assembly	11	17

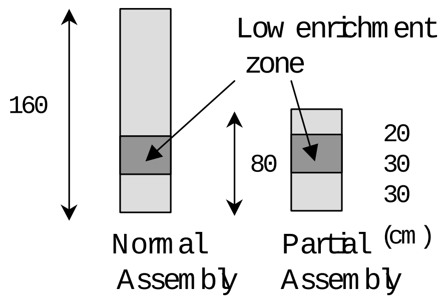


Fig. 6 Axial Pu Enrichment Distribution

4. BENCHMARK TESTS ON NCA

The BARS core has unique characteristics: its neutron spectrum is different from conventional BWRs and LMFBRs and it utilizes neutron-streaming phenomena that are difficult to evaluate accurately. Therefore, we have a program to perform benchmark tests in the NCA (Toshiba Nuclear Critical Assembly) to verify the neutronics calculation method. The NCA is a slightly enriched, uranium-fueled, light-water-moderated critical assembly, which has been utilized to verify both LWR design codes and the specific fuel design.

One purpose of the tests is to verify the method in a fast spectrum core within the limitation of the use of slightly enriched uranium fuel. The important feature to be measured is the U-238 reaction rate because it occupies the main part or one of the main parts in the evaluation of the conversion ratio and reactivity coefficient (void reactivity coefficients). The neutron streaming effect is also the important feature to be measured. The evaluation accuracy of the effect is not so sensitive to the difference of the neutron spectra between uranium and MOX fuelled cores because it is strongly dependent on the collision reaction rate. Another purpose is to verify the new method to evaluate the reactivity coefficients based upon the measured data of modified conversion ratio [6, 7], aiming at applying the method to the measurement in a MOX fuelled test core in the future. The new method is very advantageous because it can evaluate the characteristics of the tight lattice zone only, eliminating the effect of the driver core zone composed of a conventional LWR lattice.

So far, basic characteristics such as conversion ratio, neutron spectrum, and power distribution were measured in the NCA test core with tight lattice test zone [8, 9]. New tests are now prepared for the measurement of reactivity coefficients and neutron streaming effect. One of fuel rod arrangements of test core is shown in Fig. 7. The basic characteristics mentioned above are measured in the central test zone with the tight fuel lattice surrounded by the driver zone with conventional LWR fuel lattice to attain criticality.

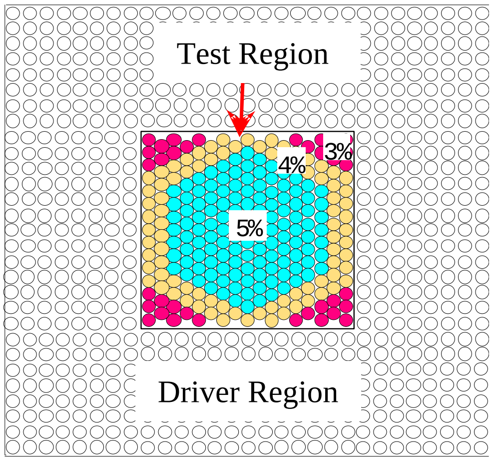


Fig. 7 One of the fuel rod arrangements of test core

5. THERMAL-HYDRAULIC TEST

In the BARS core, a tight lattice fuel must be adopted 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. In the previous section, 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 three tests:
(1) Visualization test
(2) High-pressure thermal-hydraulic test
(3) CCFL (Counter Current Flow Limitation) test

5.1 VISUALIZATION TEST

The purpose of the visualization test is to investigate the boiling transition behavior in the narrow gap bundle. It is hard to visualize the two-phase flow in the rod bundle under BWR operating condition. Therefore, it is planed to perform the two single channel tests under atmospheric condition. One is the unheated test for measuring the physical quantities of the two-phase flow in narrow channel, for instance, liquid film thickness, velocity and so on.

The other is the heated test to visualize the liquid film behavior just before occurring the boiling transition. The heater rods are made of glass, whose surface is coated by SnO2. Because the boiling transition behavior will be investigated, these test sections have the same flow geometry as shown in Fig. 8.

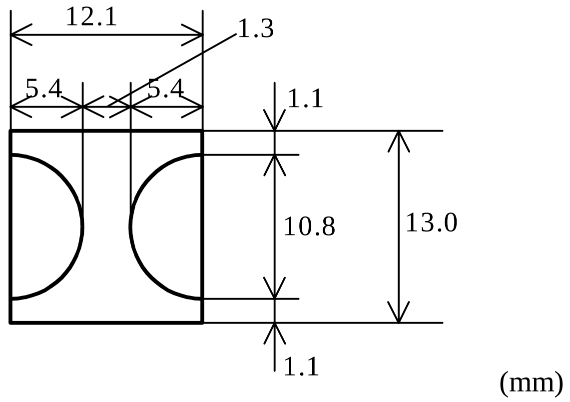


Fig. 8 Flow geometry of the test channel for Visualization test

5.2 HIGH-PRESSURE THERMAL-HYDRAULIC TEST

The purpose of the 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 paper [3], it was reported that the critical power test was performed with the tight lattice bundle and Arai's correlation could be applied to predicting critical power of tight lattice bundle. However, that previous critical power test was performed with the longer heating length than actual BARS core. Therefore, the thermal-hydraulic tests were planed to enhance the thermal-hydraulic database for the tight lattice bundle, to verify the applicability of Arai's correlation on the BARS core, and to develop the more accurate correlation for predicting critical power. Moreover, the two-phase flow instability test, transient boiling transient test and pressure drop test were planed to design the BARS core from the thermal-hydraulic point.

Fig. 9 shows the cross-sectional view of test assemblies. As shown in Fig. 9, it is planed to make two types of the test bundle. One is the 7 rods test bundle with the hexagonal channel box; the other is 14 rods test bundle with the rectangular channel box. 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. The other hands, the purpose of the 14-rod bundle test is to check the critical power for various radial power distributions.

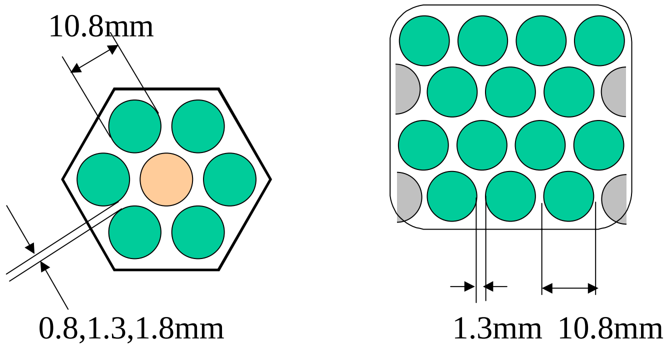


Fig. 9 Cross-sectional Views of the test bundle for the high-pressure thermal-hydraulic test

Fig. 10 shows the system diagram of the test facility. This test facility is usually called BEST (Toshiba BWR experimental loop of stability and transient). This loop has a capability of testing under BWR operating conditions.

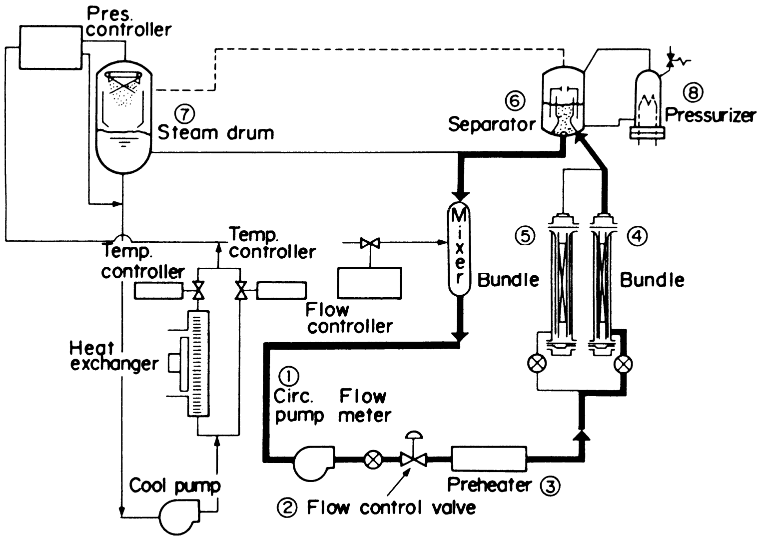


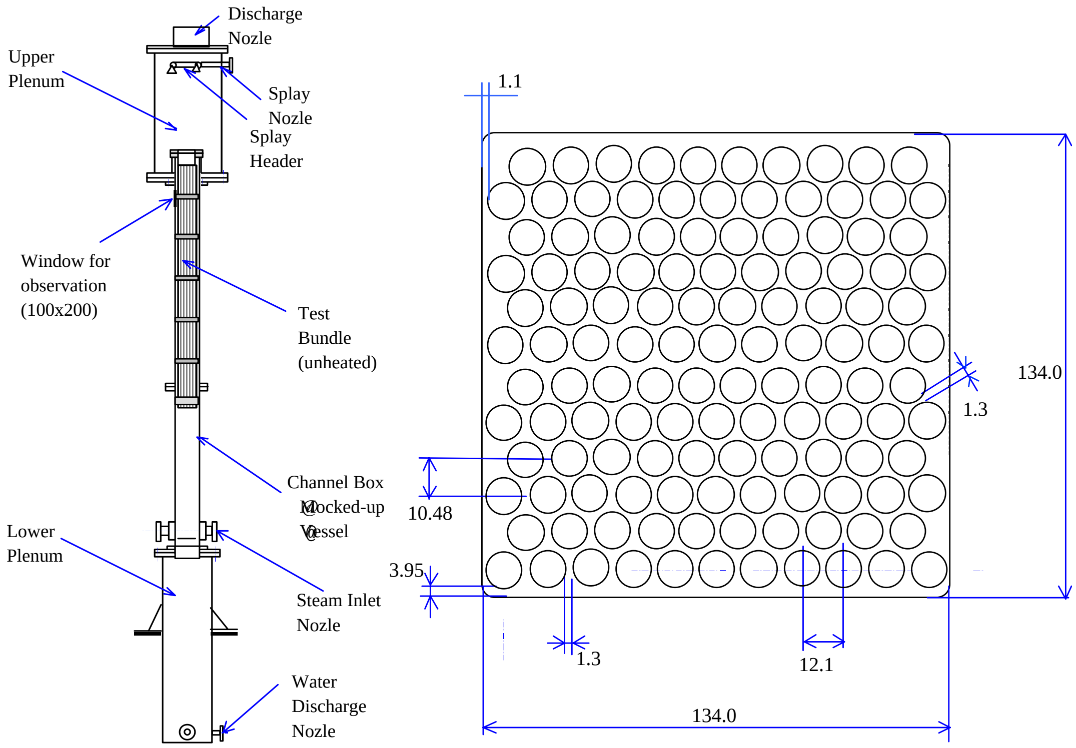
Fig. 10 Flow Diagram of Test Facility (BEST)

On critical power test, pressure, inlet water temperature and flow rate were set to planned level first. Then, the bundle power is raised step by step every a small magnitude. Critical power is defined as a power when the rod surface temperature jumps by 14 centigrade from the temperature under nucleate boiling conditions. Test conditions were planed to be as follows.

Pressure: 1~8 MPa
Mass flux: 500 ~ 2000 kg/m2s
Inlet sub-cool: 20~80 kJ/kg

5.3 CCFL (COUNTER CURRENT FLOW LIMITATION) TEST

On the tight lattice bundle, because the flow area becomes narrower than that of the usual BWR fuel, it is worried that the little splay water falls into the core when LOCA (Loss of Coolant Accident) occurred. Therefore, the purpose of this test is to investigate the CCFL characteristic and to make a CCFL correlation for the tight lattice bundle. It is planed that the correlation based on this CCFL test is applied to the LOCA analysis. Figure 11 shows the CCFL test section and cross-sectional View of the tight lattice test bundle. The conditions that the water is unable to fall down into the test bundle are defined CCFL condition. In CCFL test, the water spays to the test bundle from the upper plenum. On the other hand, the vapor is blown into the test section under the test bundle.



Test Section Cross-section of the test bundle

Fig. 11 CCFL test section and cross-sectional geometry of the test bundle

CONCLUSIONS

An innovative core design concept for LSBWR is proposed. LSBWR assumes 15-year continuous operation, natural circulation cooling and upper entry control rod system. Addition to this, we adopt the key technologies as below;
1. 0.7-times size small bundles coupled with cruciform control rod
2. Isotope enriched gadolinium
3. Peripheral positioning of Gd rods

The design study on a fuel bundle and a core has been done. A bundle nuclear design is determined based on 3-dimensional calculation and core performance has been evaluated.
1. Two types bundle design is needed for the reduction of residual gadolinium worth.
2. Void reactivity coefficient keeps negative throughout the operation.
3. Maximum enrichment needed for 15-year operation has been estimated below 20wt%.

Addition to the above results, control rod operation plan has been successfully determined so that the maximum residence time is kept less than 6 years. It means that control rod combined with small bundle is effective measure for extension of control rod life.

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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).
2. Sakashita, Y., et al: “Core Characteristics of Breeding BWR for BARS (BWR with Advanced Recycle System),” 7th International Conference on Nuclear Engineering(ICONE-7), Tokyo, April 19-23(1999).
3. Yamaoka, M. et al., “Study on Fast Spectrum BWR Core for Actinide Recycle,” 9th International Conference on Nuclear Engineering (ICONE-9), Nice, April 8-12 (2001).
4. Okumura, K., et al: “SRAC95; General Purpose Neutronics Code System” (in Japanese), JAERI-Data/Code 96-015(1996).
5. Mori, T., et al: “MVP/GMVP: General Purpose Monte Carlo Codes for Neutron and Photon Transport Code based on Continuous Energy and multigroup Methods,” JAERI-DATA/CODE 94-007(1994).
6. 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 (in Japanese), G41 (2000).
7. 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 (in Japanese), G42 (2000).
8. Miyashita, S., et al:” Critical Experiments of Low Moderated Tight Lattice Core (1)-Measurements of Neutron Flux and Modified Conversion Ratios –“, Proc. of 1999 Fall Meeting of the Atomic Energy Society of Japan (in Japanese), E27 (1999).
9. Yoshioka, K., et al: “Critical Experiments of Low Moderated and Tight Lattice Core (2)-Analysis with MCNP code-“, Proc. of 1999 Fall Meeting of the Atomic Energy Society of Japan (in Japanese), E28 (1999).