# COMPARISONS OF THERMAL SCWR ASSEMBLY DESIGNS BY IN- OR INTER-ASSEMBLY MODERATION

**ZHANG Peng[1], WANG Kan[1]**
[1] Department of Engineering Physics, Tsinghua University, Beijing, China

## Abstract

As one of the six GEN-IV reactor systems and the only one with water as coolant, SCWR is thought to be the most hopeful future nuclear energy system. Many designs have already been proposed worldwide. For thermal SCWR designs, a key consideration factor is how to provide the dedicated moderation, which leads to various designs with water as moderator. It is much like BWR, where under-moderation is an important issue and central water rods are adopted to account for it. Naturally, water rods can be adopted in SCWR assemblies too, but with counter-flow scheme to increase the outlet temperature and thermal efficiency which is unlike the co-flow scheme in BWR. This type can be seen in the American, Japanese and European HPLWR designs. As the other option, moderation could be provided with inter-assembly gap like CANDU-SCWR design. It is worthy of review of these designs for better understanding the water moderating effects and putting forward any new designs.

## 1. Introduction

Since 1990s the thermal efficiency of Fossil Fired Power Plant has been greatly improved by introducing water at supercritical conditions. While for LWR, the thermal efficiency hasn't been improved a lot since 1960s. So in the recent decade of years, the idea of integrating the LWR (or HWR) and supercritical water techniques has been very popular, and Super-Critical Water-cooled Reactor (SCWR) was chosen to be one of the GEN-IV reactor systems.

Due to the rapid change of water density at supercritical water conditions, the way to provide sufficient moderation is one of the key issues for the thermal SCWR designs. The University of Tokyo has developed an assembly design with 6 by 6 water rods, and each water rod occupies the place of 3 by 3 fuel rods [1], as shown in upleft of Figure 1. A rectangular water channel around the whole assembly is designed to provide enough moderation for the peripheral fuel rods. The inlet water flow down through the water rods and peripheral water channels to provide moderation, and then flow upward through the fuel rod gaps as coolant. The densities of moderator and coolant water differ a lot, which provides the main heterogeneousness to the water rod designs.

The INEEL of America has also studied the assembly designs with water rods [2], as shown in upright of Figure 1. Their designs are more or less the same as the design of the University of Tokyo, except that there are no peripheral water channels which were developed by the University of Tokyo at a late time.

In addition, Jacopo Buongiorno of INEEL has studied an alternative hexagonal assembly design without water rods [3], as shown in lower-right of Figure 1. The feed water flowing downward in the inter-power-channel gaps acts as the neutron moderator. This design simplifies the mechanical design, but high enrichment fuel is need in the centre fuel rods to flat the local power distribution.

The FZK of Europe proposed another small assembly design [4], as shown in lower-left of Figure 1. The feed water flows downward through the centre water rod and inter-assembly gaps as moderator.

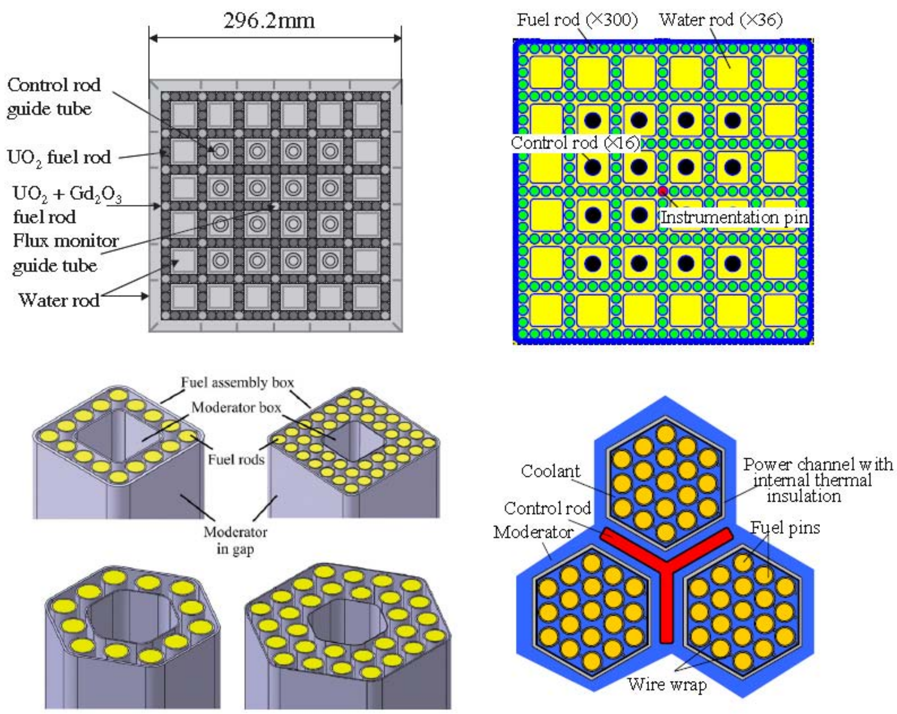



Figure 1 Various SCWR assembly designs.

## 2. Model Description

Based on the various designs been proposed, several models are chosen as the analysis objects in this study, as shown in Fig. 2. These models represent the above mentioned designs for simplicity. There are 3*3 and 4*4 water rod designs with one or two rows of fuel rods between water rods, 3*3 and 4*4 fuel channel designs with different power-channel gap widths, and the HPLWR assembly designs with different assembly gap widths. For preliminary comparisons, some parameters are fixed, which are listed in table 1. All the models are of two-dimension, with mirror reflected boundary conditions in four sides. The moderator and coolant water densities in the table are taken as a guess of the corresponding average densities. All of the following neutronics calculations are done with MCNP [5].

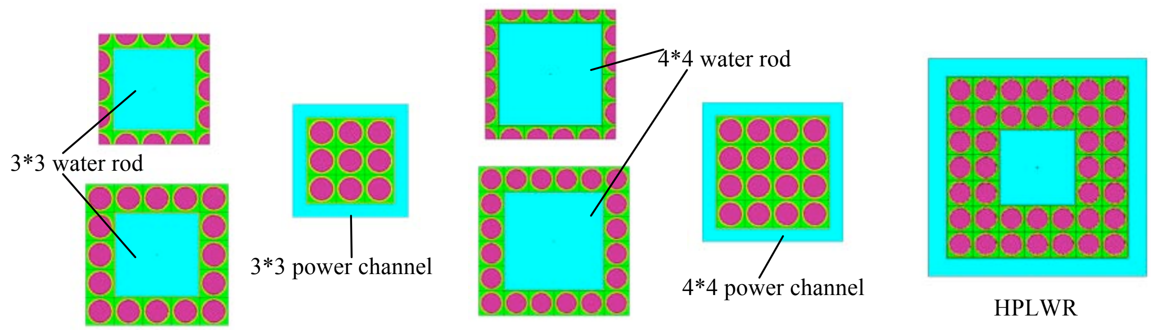



Figure 2 The analysis models.

Table 1 Some Fixed Parameters

| **Parameter** | **Value** |
|---|---|
| Width between pellet OD and clad ID | 80 microns (He) |
| Clad thickness | 0.63 mm (MA956) |
| Inter-fuel-rod gap | 1 mm |
| Channel-box thickness | 0.4 mm (MA956) |
| Moderator density | 0.6 g/cc |
| Coolant density | 0.3 g/cc |
| Fuel | 5% UO2 (10.4215 g/cc) |

## 3. Neutronics analysis

### 3.1 Model comparisons at zero burnup

First the pellet outside diameter is set as 9.1872 mm. Some results are listed in Table 2. The m/f value represents the effective cell number ratio of moderator to fuel. For 3*3 water rod designs, the m/f is too low for two-row-rod design, which usually means a bad moderation. For 3*3 power channel designs, k-inf is the maximum with 2 pitch gap, so the 1.5 pitch design is more proper. The 4*4 water rod design with two-row-rod has half m/f value of the one-row design, which means a double power density, so it's better. The 4*4 power channel designs are either with bad moderation, or with a high Peak Power Factor (PPF), so they are not idea designs. For the HPLWR assembly designs, the 2-pitch design has a very large PPF, while the 1-pitch design has small k-inf and m/f values which mean poor moderation. Finally four designs are chosen for further study: 3*3 water rod with one-row-rod, 3*3 power channel with 1.5 pitch, 4*4 water rod with two-row-rod, and the HPLWR assembly design with 1.5 pitch.

Table 2 Some results with pellet OD=9.1872 mm

| | | k-inf**) | m/f | PPF |
|---|---|---|---|---|
| 3*3 water rod | One-row-rod | 1.33514 | 9:7=1.2857 | 1.0835 |
| | Two-row-rod | 1.26718 | 9:16=0.5625 | 1.0900 |
| 3*3 power channel | 1 pitch*) | 1.28982 | 7:9=0.7778 | 1.0962 |
| | 1.5 pitch | 1.34921 | 11.25:9=1.25 | 1.1144 |
| | 2 pitch | 1.36524 | 16:9=1.7778 | 1.1260 |
| | 2.5 pitch | 1.35447 | 21.25:9=2.3611 | 1.1341 |
| 4*4 water rod | One-row-rod | 1.35091 | 16:9=1.7778 | 1.1112 |
| | Two-row-rod | 1.31363 | 16:20=0.8 | 1.1282 |
| 4*4 power channel | 1 pitch | 1.25307 | 9:16=0.5625 | 1.1361 |
| | 1.5 pitch | 1.32353 | 14.25:16=0.8906 | 1.1680 |
| | 2 pitch | 1.35448 | 20:16=1.25 | 1.1851 |
| | 2.5 pitch | 1.35948 | 26.25:16=1.6406 | 1.1976 |
| | 3 pitch | 1.34806 | 33:16=2.0625 | 1.2077 |
| HPLWR | 1 pitch | 1.25544 | 24:40=0.6 | 1.1094 |
| | 1.5 pitch | 1.30607 | 32.25:40=0.80625 | 1.2064 |
| | 2 pitch | 1.33408 | 41:40=1.025 | 1.2686 |

*) It is the gap width of adjacent power channel;

**) The standard deviation of k-inf is about 0.0006.

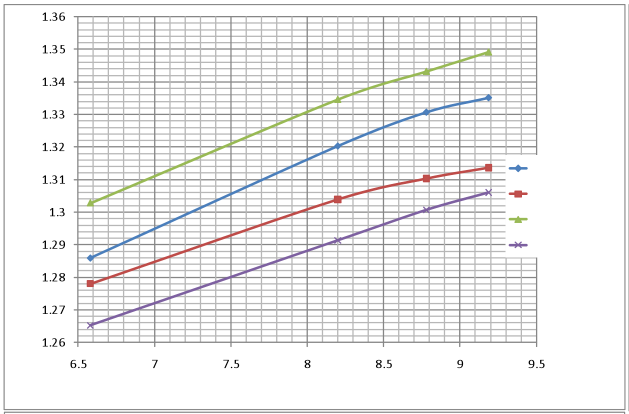
### 3.2 Sensitivity analysis of pellet outside diameter

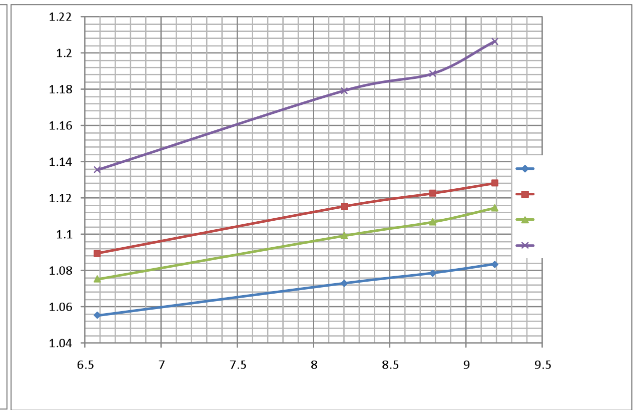
Based on literature investigation, the fuel pellet outside diameter differs from 6.58mm to 9.1872mm. The influence of this value to the neutronics parameters has been studied. As shown in Fig. 3, the k-inf and PPF are both increasing linearly approximately with fuel pellet OD. The 3*3 one-row and 3*3 fuel designs seem better than the other two designs, for they have relatively higher initial k-infs and lower PPFs. While from the power generation view, the following relation holds for the average power density p:

$$p \propto \frac{1}{(m/f+1)\cdot PPF} \tag{1}$$

As seen from Fig. 3, the average power density changes a little while the fuel pellet OD changes, and the 4*4 two-row and HPLWR designs are better.

In the following studies, the fuel pellet OD is set to be 8.78mm.





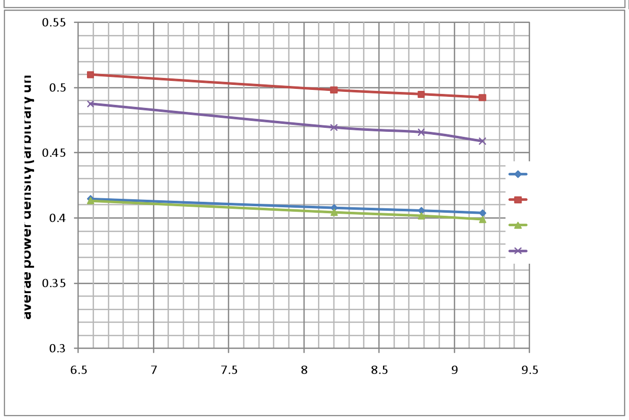


Figure 3 The influence of fuel pellet OD.

## 3.3 Burn-up analysis

The burn-up calculations are done using MCBurn code [6]. The specific power is assumed to be 40W/gU. The reactivity swing and PPF changes with burn-up are illustrated in Fig. 4. The 3*3 water-rod design and 3*3 fuel channel design have the same reduction rate of k-inf, so do the 4*4 water-rod design and HPLWR design. This may due to the similar moderator-to-fuel ratio of the two pair. The PPF of the four designs all decrease with burn-up, and much optimization work need to be done to improve the value.

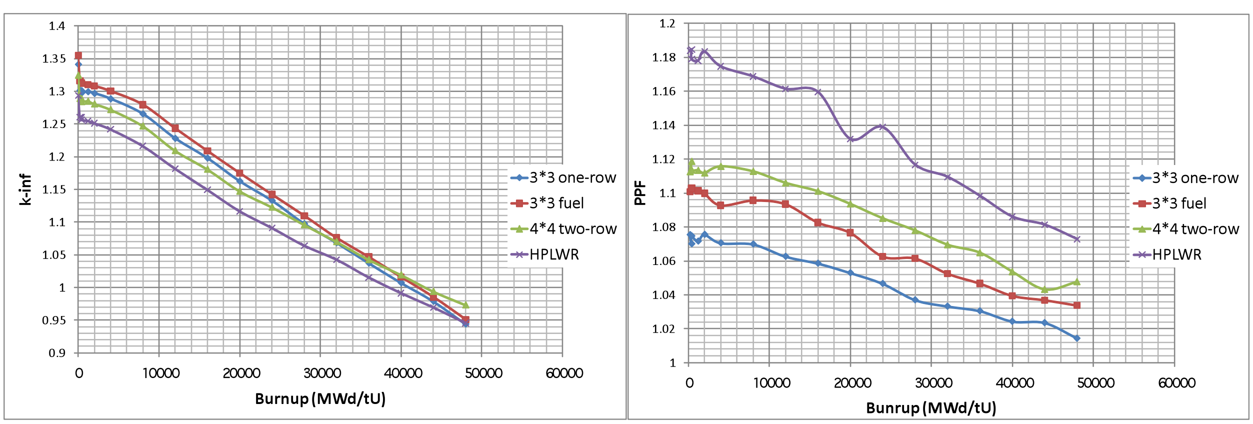



Figure 4 K-inf and PPF change during burn-up.

The enrichment of fissile materials changes with burn-up, as shown in Fig. 5 for the four designs. The decreasing ratios of HPLWR design and 4*4 water-rod design are smaller, which means a bigger conversion ratio. This is coincident with the small decreasing ratio of k-inf of the two designs.

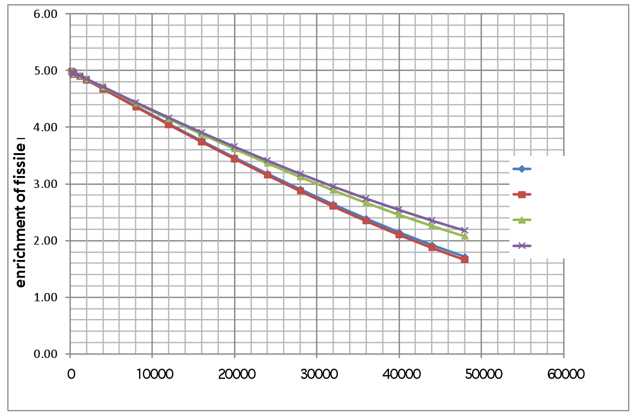



Figure 5 Enrichment of fissile nuclides changes with burn-up.

The coolant void coefficients and the Doppler coefficients of the four designs have also been compared, as shown in Fig. 6. The HPLWR design has the lowest coolant void coefficient, which may due to its more neutron absorption of moderator. The 3*3 fuel channel design has the highest coolant void coefficient, which may result from the neutron shielding reduction of outer fuel rods at coolant void condition. The two water-rod designs have similar coolant void coefficients. For Doppler coefficient, differences are not so clear for the four designs, but similar ranking can be observed.

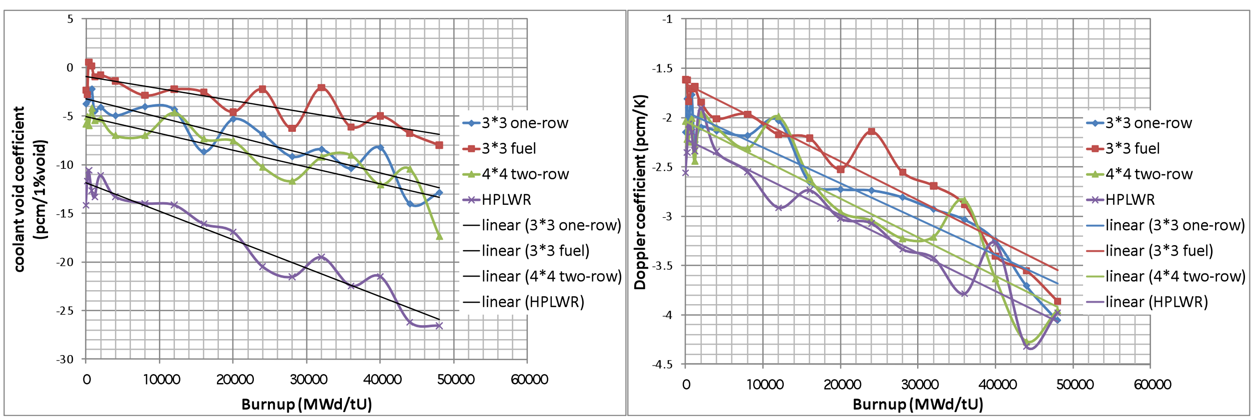



Figure 6 The coolant void coefficient and Doppler coefficient.

## 4. Mechanical consideration

Four models have been analyzed above, which are the simplification of four different designs. For the water rod designs, the simplified model and the actual assembly are compared in Fig. 7. It can be seen that for the 3*3 water-rod one-row design, two rows of fuel rods will locate between the peripheral water rods of two adjacent assemblies, and the assembly ducts and the inter-assembly gaps will decrease the uniformity also. Thus, compared with the simplified model, the neutronics parameters of the assembly will change a lot. For example, the k-inf will decrease due to the neutron absorption of the assembly duct, and the PPF will increase. As for the 4*4 water-rod two-row design, things seem better, but there are still the assembly ducts which will decrease the uniformity. If inter-assembly moderation is considered, the uniformity can be kept well, like the HPLWR designs.

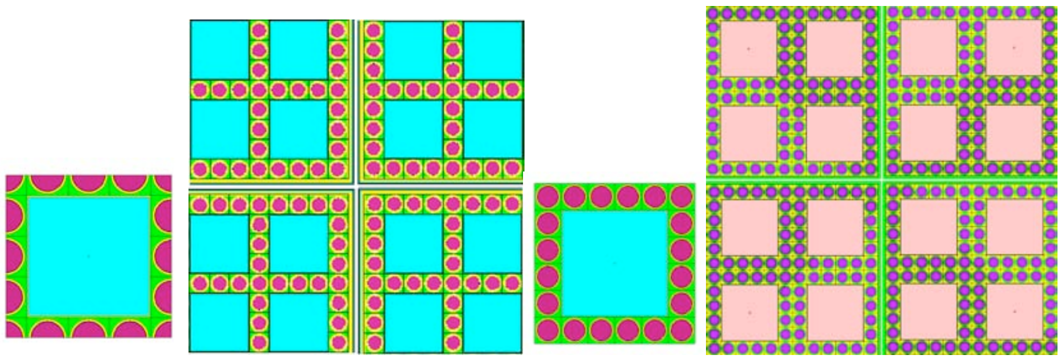


Figure 7 Comparisons of the water rod model and corresponding assembly designs.

## 5. Conclusion

Four models which represent four different SCWR assembly designs for simplicity have been analyzed to compare the effect of in- or inter-assembly moderation. It is found that some parameters are not sensitive with the moderation mode, such as the slope of the k-inf curves and the conversion ratio, which seem to be more sensitive with the moderator-to-fuel ratio. With similar moderator-to-fuel ratio, higher PPF would happen with the inter-assembly moderation mode. But since further work need to do to improve the PPF value, it’s hard to say which moderation mode is better. If considering the mechanical design, the inter-assembly moderation mode would have some advantages in the assembly construction.

For the power channel design, the neutron shielding effect of the outer fuel rods is obvious, so higher enrichment is needed in the inner fuel rods to flat the local power distribution, and the safety related parameters, like the coolant void coefficient and Doppler coefficient, need to be checked.

## 6. References

[1] Kazuhiro KAMEI, Akifumi YAMAJI, Yuki ISHIWATARI, Yoshiaki OKA and Jie LIU. Fuel and Core Design of Super Light Water Reactor with Low Leakage Fuel Loading Pattern. *Journal of NUCLEAR SCIENCE and TECHNOLOGY*, Vol. 43, No. 2, p. 129–139 (2006).

[2] Jacopo Buongiorno, Philip E. MacDonald. Supercritical Water Reactor (SCWR) Progress Report for the FY-03 Generation-IV R&D Activities for the Development of the SCWR in the U.S. INEEL/EXT-03-01210. September 30, 2003.

[3] Jacopo Buongiorno. An Alternative SCWR Design Based on Vertical Power Channels and Hexagonal Fuel Assemblies. Global 2003, New Orleans, LA. Nov. 16-20, 2003, pp. 1155-1162.

[4] J. Hofmeister, C. Waata, J. Starflinger, T. Schulenberg, E. Laurien. Fuel assembly design study for a reactor with supercritical water. *Nuclear Engineering and Design* 237 (2007) 1513-1521.

[5] Judith F. Briesmeister (editor), "MCNP™ — A General Monte Carlo N-Particle Transport Code, Version 4C," LA-13709-M, Los Alamos National Laboratory (April 2000).

[6] YU Ganglin, WANG Kan, WANG Yuhong, "MCBurn — A Coupling Package of Program MCNP and ORIGEN," *Atomic Energy Science and Technology*, Vol. 37, pp. 250-254 (2003).