NEUTRONIC ANALYSIS OF HPLWR FUEL ASSEMBLY CLUSTER

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ABSTRACT

In the present study the neutronic analysis of fuel assembly cluster of the High Performance Light Water Reactor (HPLWR) is discussed. Neutronic calculations are performed using WIMS-D4 and CITATION codes. The obtained neutronic results including axial power distribution, neutron flux, and power peaking factors are discussed. Neutron flux is strongly influenced by the coolant and moderator density. Because of the high density of the coolant fluid in the lower part of the fuel assembly, the axial power distribution and thermal neutron flux reaches its maximum value in the lower part of the fuel assembly. Thermal neutron flux falls in the upper parts of the assembly significantly.

Key words: (Neutronic, HPLWR, Supercritical-pressure, Fuel assembly)

Introduction

The High Performance Light Water Reactor (HPLWR) is a supercritical water reactor (SCWR) design, proposed and investigated within an EU-Project [1]. The main characteristic of the supercritical water reactor (SCWR) concept [2] is its operating pressure above the critical point of water (22.1 MPa, 374 °C). This gives many advantages compared to current light water reactors (LWRs): increase in thermal efficiency due to higher operating temperature (44% against 33–35% for LWR), no boiling crisis given the essentially single-phase nature of supercritical water, and a better economics for the absence of steam separators and dryers as well as steam generators and smaller containment building [3]. A once through cycle SCWR was developed by Oka and Koshizuka [4] where the water enters into the core as liquid and exits at a high temperature and pressure. A first validation of coupled neutron physics and thermal–hydraulic analysis was carried out by Broeders [5]. Waata [6] developed a coupled simulation approach with MCNP and the sub-channel code STAFAS for a HPLWR fuel assembly analysis.
Development of a coupled neutronic/thermal–hydraulic tool with multi-scale capabilities and applications to HPLWR core analysis has been presented by Monti [7]. The neutronic analysis of fuel assembly cluster of the HPLWR using neutronics/thermal–hydraulic coupling is presented in this study. Due to strong interdependence between power distribution, coolant and moderator density, as well as drastic changes in coolant density in HPLWR reactors, the neutronic/thermal–hydraulic coupling is therefore essential for neutronic analysis of fuel assemblies. In what follows, the neutronic results including power distribution, thermal neutron flux, and axial distribution of power peaking factor will be discussed.

**Methods and materials**

The coolant flow path in the reactor pressure vessel (RPV) is illustrated in Figure 1. The super critical water at 25 MPa and 280 °C flows into RPV through the cold nozzle and splits into two parts. Part of the inlet coolant is led to the top dome and the rest flows to the bottom dome via the down comer. The coolant in the top dome then flows down to the mixing plenum through the water rods (moderator channel) and assembly gap via the control rod cluster guide tube. At the mixing plenum, the coolant from the down comer and the water rods are mixed and the mixture rises up the coolant channels in the fuel assemblies. This counter current flow scheme provides good moderation capacity at the top of the core.

![Figure 1 SCWR Reactor pressure vessel.](image)

The fuel assembly consists of 7 by7 fuel rods array arranged in square lattice. Due to insufficient moderation capability inside the fuel assembly additional water gaps are added between the fuel assemblies. Figure 2 illustrates the HPLWR fuel assembly. The width of the outer box of the HPLWR fuel assembly is only 67.2 mm which is about one third of the typical fuel assembly size of a PWR. The handling of a core consisting of such small fuel assemblies would be complex during a refuelling process. To overcome this problem, a fuel assembly cluster is proposed instead, comprising nine fuel assemblies in a 3 × 3 arrangement. Figure 2 shows such a 3 × 3 arrangement of the. The computer codes WIMS-D4 and CITATION were used for neutronic analysis. The codes have jointly been used for neutronic
calculation of different thermal power reactors. WIMS-D4 code has been used for cell calculation of fuel assembly. Fuel assembly is then simulated by CITATION code. This code uses the finite-difference representation of diffusion theory in three dimensions with arbitrary group-to-group scattering. In the thermal–hydraulic analysis, the continuity, momentum and energy equations are solved numerically using finite volume (upwind) method. After inserting the initial guess of power distribution in the entire control volume, density and temperature distribution of water and the moderator as well as fuel temperature in the selected control volume is calculated. The obtained thermal–hydraulic parameters are used in the neutronic code (WIMS-D4). The effective cross-sections and group constants obtained from WIMS-D4 for various regions of the fuel assembly along with the fuel assembly geometry and boundary condition are then provided as input data to the CITATION code to calculate power distribution, effective multiplication factor and power peaking factor. The output data of WIMS-D4 and CITATION is used as input to a thermal hydraulic code to calculate thermal hydraulic properties of fluid and fuel which are then used as input data in neutronic codes.

![Figure 2](image2.png)

**Figure 2** HPLWR fuel assembly (left), fuel assembly cluster (right)

**Results and discussion**

In this section the results of neutronic analysis of the HPLWR fuel assembly cluster using neutronic/thermal–hydraulic coupling will be presented. HPLWR fuel assembly cluster analysis is performed with nine fuel assemblies (FA1–FA9). In this process, the hottest fuel assembly (FA5) is obtained. Next, the neutronic analysis of the hottest fuel assembly is performed using 1/4 symmetry containing 15 coolant channels (5 A-type channels, 8 B-type channels, 1 C-type channel, and 1 D-type channel) and 12 fuel rods (R1–R12). **Figure 3** shows the axial power distribution along the fuel rods height. The produced axial power in fuel rod R4 is less than other fuel rods because of its lower enrichment. Fuel rods R1 and R6 have the highest axial powers respectively. As seen, power distribution reaches its maximum value in the lower part of the fuel assembly. **Figure 4** shows the distribution of thermal flux at the height of 0.75 m (hottest plane) from core inlet. The thermal flux reaches its maximum value in the moderator channel and the assembly gap. **Figure 5** illustrates the distribution of fast neutron flux at the height 0.75 m (hottest plane). The maximum value of the fast neutron flux is found in R7, while the minimum is observed at the centre of the moderator channel.
Figure 3 Power distribution in the fuel rods along the active height.

Figure 4 Thermal neutron flux distribution \( (n \text{ cm}^{-2} \text{s}^{-1}) \) at fuel assembly hot plane.

Figure 5 Fast neutron flux distribution \( (n \text{ cm}^{-2} \text{s}^{-1}) \) at fuel assembly hot plane.

Figure 6 shows the axial distribution of power peaking factor (PPF) in fuel assembly. Axial PPF reaches its maximum value in the lower parts of the fuel assembly. The maximum value of axial PPF is 2.03. Figure 7 depicts the radial distribution of power peaking factor in fuel assembly. The maximum and average distribution of thermal neutron flux in the moderator channel can be seen in Figure 8. Figure 9 shows the changes in effective multiplication factor with respect to burn-up, where the control rods are outside of the fuel assembly.
Figure 6 Normalized axial power along the fuel assembly height.

Figure 7 Radial power peaking factor in fuel rods.

Figure 8 Thermal neutron flux in moderator channel along active height.
Figure 9 Changes in effective multiplication factor in proportion to burn-up.

Figure 10 illustrates the changes in the effective multiplication factor, during reactivity insertion at different moderator mass flow rate. Decreasing the moderator mass flow rate decreases thermal neutron flux in the fuel assembly. At the moderator mass flow rate more than 0.3 kg/s and between 0.234 kg/s to 0.26 kg/s, decreasing the moderator mass flow rate increases the multiplication factor because of decreases of the total loss of neutrons. Total loss of neutrons comprises neutrons leakage and absorption in structure materials such as fuel assembly box, the moderator channel, the assembly gap and the fuel rod cladding. In other range of moderator mass flow rate, decreasing the moderator mass flow rate decreases the multiplication factor due to increases of total loss of neutrons (Figure 11).

Figure 10 Effective multiplication factor in different moderator mass flow rate.

Figure 11 Total loss of neutrons ($n \text{ cm}^{-2} \text{s}^{-1}$) in different moderator mass flow rate.
Conclusion

In the present study, the neutronic analysis of the HPLWR fuel assembly is discussed. Neutron flux is strongly influenced by the coolant and moderator density. Because of the high density of the coolant fluid in the lower part of the fuel assembly, the axial power distribution and thermal neutron flux reaches its maximum value in the lower part of the fuel assembly. Thermal neutron flux falls in the upper parts of the assembly significantly. Thermal neutron flux reaches its maximum value at the center of moderator channel at the height of 0.75 m. The maximum axial power is discovered in R6 at the approximate height of 0.75 m. The maximum values of axial and radial PPF are 2.03 and 2.05 respectively. The fuel rods which are nearer to the moderator channel have a higher thermal neutron flux level than the ones closer to the periphery of the fuel assembly. The corner fuel rods have a lower neutron flux compared to other rods because of lower enrichment. The moderator mass flow rate should be less than 0.234 kg/s due to negative reactivity insertion at abnormal events such as loss of moderator accident.

References