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Core design study on CANDU-SCWR with 3D neutronics/thermal-hydraulics coupling

Ping Yang, Liangzhi Cao*, Hongchun Wu, Changhui Wang

Department of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an 710049, China

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ABSTRACT

Using separated heavy water as moderator and supercritical water (SCW) as coolant introduces challenge for CANDU-SCWR to get a negative coolant void reactivity (CVR), due to which the moderator thickness of the fuel channel is optimized in this paper. When SCW flows through the core, there is a rapid variation in SCW density, which is directly related to the neutron spectrum and subsequently to the power distribution, so the 3D core neutronics/thermal-hydraulics coupling is needed to accurately evaluate the core coolant density and power distribution. In this paper, the neutronics calculation is computed with 3D fine mesh diffusion code while the thermal-hydraulic calculation is based on single channel model, they are coupled with each other automatically by a link code. Further, the in-core fuel management can be simulated by the link code to search the equilibrium cycle. Based on these calculation models, a CANDU-SCWR equilibrium core is designed with a thermal power of 2540 MW, the core equivalent diameter is 4.30 m and the active length is 5.94 m. A 3-batch fuel management scheme with a cycle length of 350 EFPD is used. The numerical results show that a high average outlet coolant temperature of 625 °C is achieved with a maximum cladding surface temperature less than 850 °C. The maximum linear heat generation rate is 50.6 kW/m, the average discharged burnup is 38.1 GWd/tU, and the CVR is negative throughout the cycle.

1. Introduction

The supercritical water-cooled reactor (SCWR) is considered one of the most promising Generation IV reactor concepts for its high thermal efficiency and considerable plant simplification (Generation IV International Forum, 2002). Several design options using pressure-vessel and pressure-tube in both thermal and fast neutron spectrum have been studied worldwide. University of Tokyo developed pressure-vessel type SCWRs in both thermal and fast spectrum, and the thermal one is called super LWR (Dobashi et al., 1998) while the fast one is called super fast reactor (Cao et al., 2008). High Performance Light Water Reactor (HPLWR) is another pressure-vessel type developed in Europe (Schulenberg and Starflinger, 2007).

CANDU-SCWR is one of the pressure-tube type SCWR proposed by Atomic Energy of Canada Limited (AECL) with a separate heavy water moderator (Duffey et al., 2005), it has lots of advantages compared to the pressure-vessel ones. Firstly, the moderator is separated from SCW coolant so that the large density change of SCW has less effect on neutronics, and by interlacing the flow direction of neighboring channels, the average axial power distribution can be flattened (Torgerson et al., 2006). Secondly, the moderator provides back-up heat sink capability, which makes the pressure-tube type inherently safe. Thirdly, by using multi-pass flows the reheat and superheat designs are feasible without overheating the pressure tube. Moreover, it is easier for pressure-tube type reactors to reach a much higher pressure boundary. Table 1 summarizes the preliminary design parameters of CANDU-SCWR (Khartabil et al., 2005).

When SCW goes though the core from inlet to outlet, the density may decreases by 90%. Since the SCW density has a strong link between neutron spectrum (moderation and absorption) and subsequently power distribution, such a large variation should be carefully treated. In order to deal with the SCW density variation in the axial direction, this paper has established a 3D coupled neutronics and thermal-hydraulics (TH) calculation model, in which the neutronics calculation is computed with 3D fine mesh diffusion code while the TH calculation is based on the single channel model. Besides, the in-core fuel management is studied. The refueling schemes are simulated to find that if the equilibrium cycle can be reached. Based on the calculation models mentioned above, the lattice physics and full core neutronics have been studied and a detailed CANDU-SCWR core design based on the parameters presented in Table 1.

This paper is organized as follows. Section 2 introduces the methods used in lattice and core design. Section 3 performs some lattice physics analysis and determines the fuel channel design.

* Corresponding author. Tel.: +86 2982663285; fax: +86 2982667802.
E-mail address: caolz@mail.xjtu.edu.cn (L. Cao).

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Table 1
Preliminary parameters of CANDU-SCWR.

Parameter	Value
Thermal power (MW)	2540
Electrical power (MW)	1220
Thermal efficiency (%)	48
Operation pressure (MPa)	25
Inlet temperature (°C)	350
Average outlet temperature (°C)	625
Mass flow rate (kg/s)	1320
Number of fuel channels	300
Cladding temperature (°C)	<850

Section 4 studies the in-core fuel management and gives a CANDU-SCWR core design. Finally, some conclusions are summarized in Section 5.

2. Design methods

2.1. Nuclear design method

The DRAGON code (Marleau et al., 2000) based on collision probability techniques is used to perform the 2D assembly transport calculation for lattice physics study and generating the macroscopic cross sections and diffusion coefficients for subsequent core calculations. Cross sections are taken from the 69-group WIMS-D library based on ENDF/B-VII data. For the current calculation, the overall fuel channel is homogenized and condensed to 7 energy groups. Fuel depletion calculations are done under a specific bundle power density, and the burnup steps are input in days. Macroscopic cross section sets are prepared as a function of water density and burnup, by which the macroscopic cross section can be interpolated for a given water density and burnup. The lattice code Dragon has been widely used for modeling 2D or 3D CANDU reactor cluster geometry, and it has also been used for lattice physics study of supercritical water reactor nowadays and gains good accuracy.

Core depletion calculation is based on the 3D multi-group diffusion code CITATION in Cartesian geometry (Fowler et al., 1971). As CITATION has no function of depletion calculation, an auxiliary code has been developed. The core calculation only models one quarter of the core, and no control rod or other absorber is considered yet. There are 1664 zones in the calculation model, 900 of which represent the core, and the rest are heavy water reflectors. As we know, using CITATION to do 3D core diffusion calculation is very time-consuming when the mesh number is big, and it may cause accumulation of rounding error for solving the large linear system of equations. Therefore, a little big mesh size with a radial mesh size is 5.5 cm by 5.5 cm and axial size of 12.4 cm is chosen to reduce the mesh number. The big axial mesh reduces the calculation cost a lot but has a little effect on accuracy because the bundles in a channel are with the same enrichment so far. The mesh size is a little big for a finite-difference method, but our results show that the accuracy is acceptable. The calculation mesh divisions are shown in Fig. 1. For core depletion calculation, 3D CITATION calculations are performed for 8 burnup steps, and also for coolant void conditions.

2.2. Thermal hydraulic calculation

The TH calculation is based on single-channel model. Every fuel channel is averaged into two single rod models, one is the hot fuel rod and the other is the average fuel rod. Both models have the same TH parameters but different power distribution, the hot fuel rod has the peak liner heat generation rate (LHGR) while the average fuel rod has the average LHGR. With the spatial power distribution calculated by core depletion calculation, the power distribution of hot fuel rods and average fuel rods can be obtained at all burnup steps. Using the hot fuel rods at the middle of cycle, the core mass flow rate distribution can be searched to satisfy the maximum cladding surface temperature (MCST). After that, we can easily calculate the coolant density distribution at all burnup steps by the average fuel rods and the MCST at BOC and EOC by hot fuel rods.

2.3. Equilibrium cycle search with neutronics/TH coupling

The neutronics/TH coupling and equilibrium cycle search are implemented into one scheme as shown in Fig. 2. The equilibrium cycle means after several cycles operation, the fuel characteristics of (n)th cycle is similar to ($n+1$)th cycle. The initial core is assumed to be cold and clean, which means the coolant density is the same and the fuel is fresh in the whole core. The initial core is calculated by core depletion calculation and the spatial power distributions are obtained. The TH code takes the power distributions as input to update the coolant density distributions, with which the core depletion calculation can get newer power distributions. The neutronics and TH code solves iteratively until the coolant density is converged, which indicates the first cycle calculation is finished, and the burnup distribution of EOC is recorded. For the next cycle, the fuel bundles are reloaded according to the refueling scheme with the recorded burnup distribution, the burnup distribution at BOC of this cycle can be obtained. Such process is repeated until the BOC burnup distribution is converged, which means the equilibrium cycle can be reached by these core design parameters.

3. Fuel channel design

High thermal efficiency requires high core outlet coolant temperature, which can be obtained by high inlet coolant temperature and regenerative heat transfer (Duffey et al., 2004), and the key technology is the high efficiency channel (HEC) design as shown in Fig. 3. Considering the pressure tube suffers excessive corrosion at high temperature, this channel design places ceramic material inside the pressure tube to insulate the pressure tube from the coolant, and calandria tube is eliminated. And there are small openings in the linear and insulator, so the coolant pressure is taken directly by the pressure tube (Chow

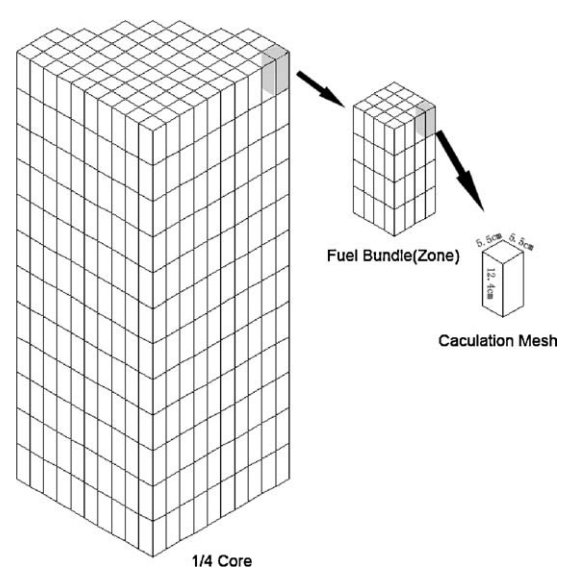

Fig. 1. 3D core calculation geometry and the mesh divisions.

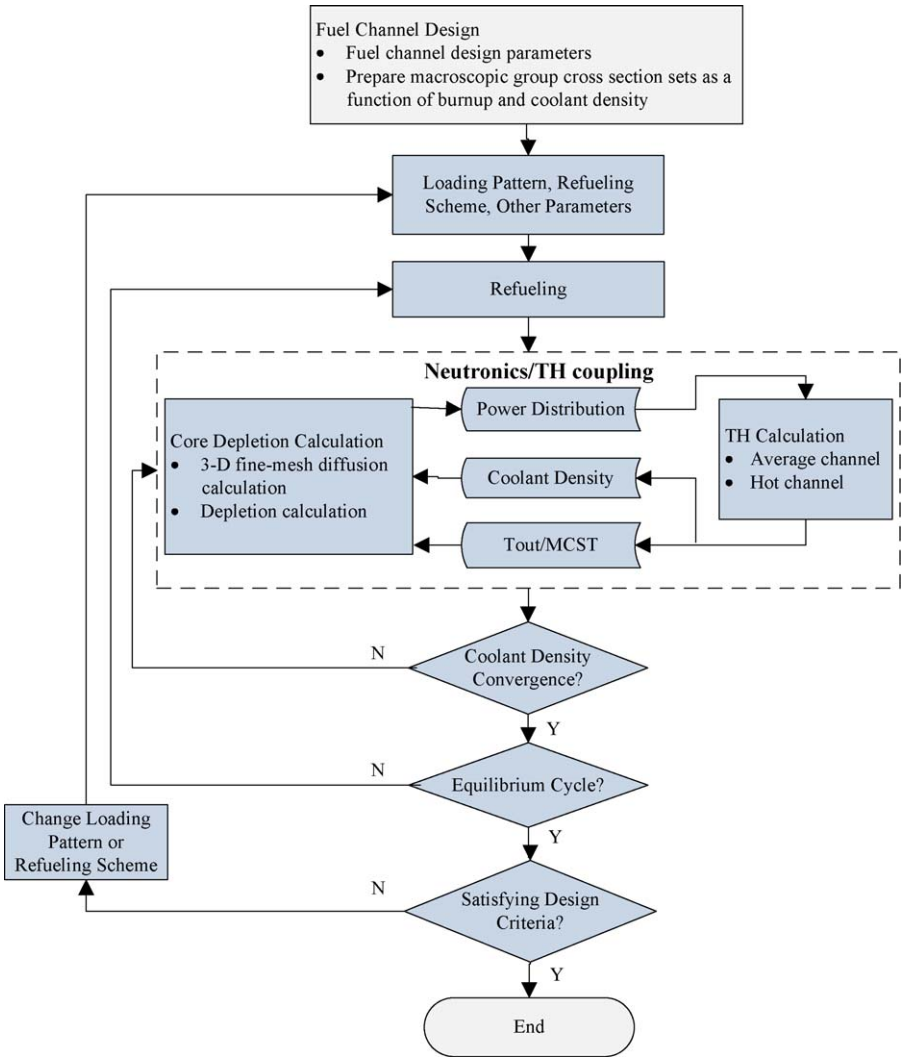

Fig. 2. Overall design procedure for the CANDU-SCWR core design.

and Khartabil, 2007). High pressure and temperature inside the core requires special material, which is given in Table 2. Most of the material has good chemical and mechanic performance but has a negative impact on neuron economy, so the fuel must be enriched. This channel design has 43 fuel rods, and it is geometrically similar to CANFLEX (CANDU Flexible) fuel bundle. At present, all the fuel rods are loaded with the same enrichment of UO_2, non-uniform fuel loading will be studied in the future.

Because the moderator is separated from SCW coolant, the density decrease of SCW has relatively less effect on neutronics, which is good for reducing the severe axial flux tilt but not good for getting a negative CVR. As we know, when the void fraction goes up, the density goes down, it causes a decrease in moderation capability but an increase in probability that neutrons escape from absorption, both of these effects are exhibited and the one that dominates depends on the neutron spectrum. As an important characteristic of the lattice, the moderator-to-fuel (M/F) ratio has a considerable effect on neutron spectrum. With a constant outside diameter of pressure tube, different M/F ratios are gained by variation of the lattice pitch (LP). Fig. 4 shows the *k*-infinite as function of SCW density with different LP. When the LP is 24 cm, the *k*-infinite goes down as the SCW density goes up, in other words, *k*-infinite goes down as the void faction goes down, which means the CVR is positive. The CVR becomes more negative as the LP decreases and becomes negative at all SCW density conditions when the LP is less than 21 cm. However, 21 cm LP cannot provide enough space for feeders between channels, which requires at least 22 cm LP. Therefore, the 22 cm is selected as LP, but the CVR is a little positive at low SCW density conditions for this design. In order to get negative CVR at all SCW condition with a 22 cm LP, the wall thickness of insulator and pressure tube are increased a little, which means the M/F is reduced. The finally specifications of lattice design are shown in Table 3, where the wall thickness of insulator and pressure tube is 20 mm and 9 mm respectively.

The CVR of the fuel channel design is given in Fig. 5. The results illustrate that the CVR decreases with void fraction and burnup, which means inherently safety when a loss of coolant accident happens. Except the negative CVR, the 22 cm LP design also reduces the

Table 2
Material selections for 43-element HEC (based on Chow and Khartabil, 2007).

Name	Material
Fuel	UO_2
Enrichment	4%
Fuel cladding	Incoloy 800
Metal linear	9Cr1Mo
Insulator	ZrO_2
Pressure tube	Excel

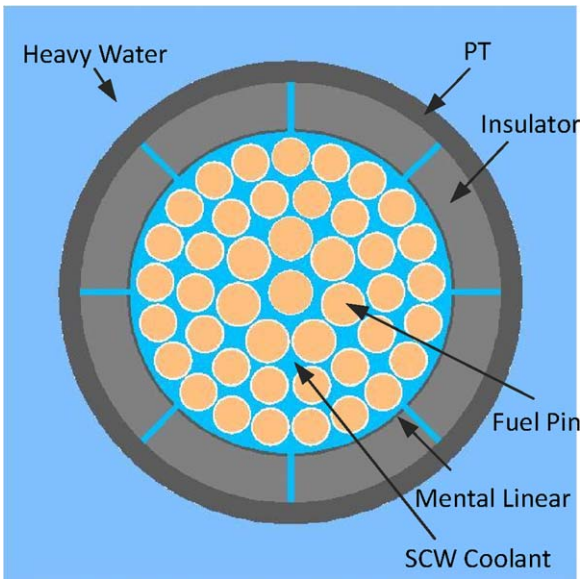

Fig. 3. Cross-sectional view of 43-element HEC.

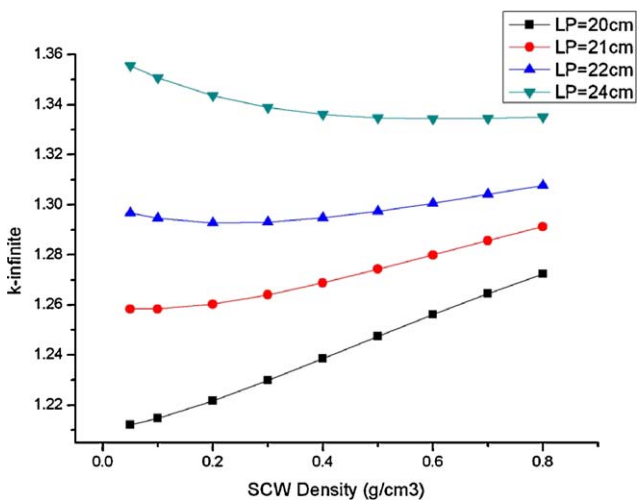

Fig. 4. *k*-Infinite as function of SCW density with different LP.

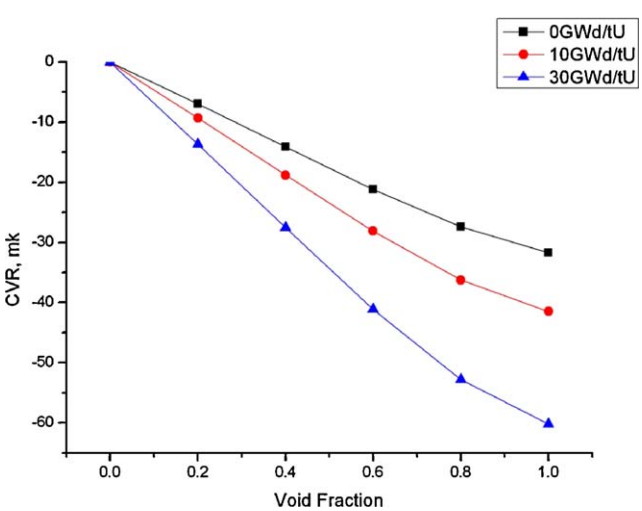

Fig. 5. The CVR of the 43-element HEC.

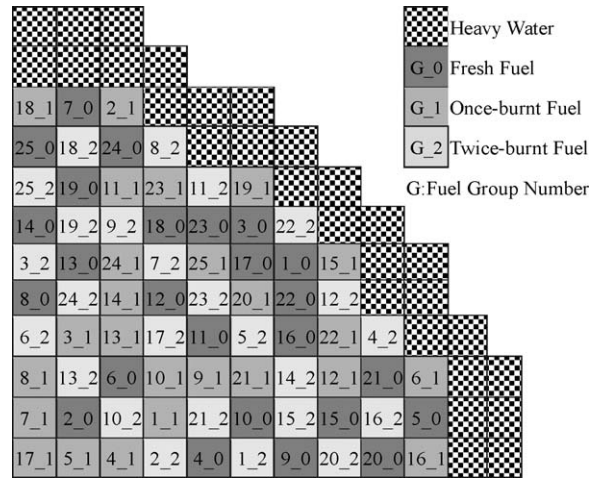

Fig. 6. Equilibrium core fuel loading pattern (1/4 core).

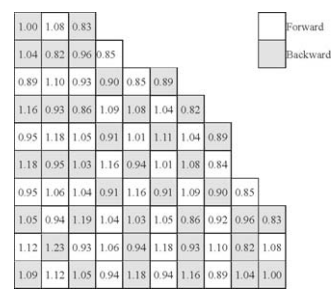

Fig. 7. Relative coolant flow distribution (1/4 core).

inventory of heavy water to make the core design more compact and improves the reactor physics characteristics.

4. Core design

4.1. Design-summary description

The 43-element HEC mentioned above is used in the core design, and there are 300 such fuel channels in the core as shown in Fig. 6 (75 channels in a quarter core). Every channel consists of 12 bundles with the same enrichment, so the active core has a length of 5.94 m and an equivalent diameter of 4.30 m. The reactor core is surrounded by an annular heavy water reflector with a thickness of 0.4 m to reduce the neutron leakage. The core is fueled with slightly enriched uranium, so it is not necessary to do online refueling as traditional CANDU reactors, and the batch refueling is used like PWRs. The average enrichment of fuel is about 4%, the quantity of uranium per fuel bundle is about 19 kg and in the core it is 68 tons. Up to now, no control rod or other absorber is considered in the core, and the average fuel temperature is set to be 1200 K. The core

Table 3
Specifications of the 43-element HEC.

Parameter	Value (unit mm)
Lattice pitch (square)	220
Pressure tube inside diameter/wall thickness	146.11/9.00
Insulator inside diameter/wall thickness	106.11/20.00
Metal linear inside diameter/wall thickness	104.11/1.00
First ring/second ring/third ring diameter	34.85/61.79/88.08
Center rod and first ring of fuel elements diameter	13.5
Second and third ring of fuel elements diameter	11.5
Fuel cladding wall thickness	0.30
Length of bundle	495

Table 4
Preliminary parameters of CANDU-SCWR.

Parameter	Value
Operation pressure (MPa)	25
Thermal/electrical power (MW)	2540/1220
Cycle length (EFPD)	350
Equivalent diameter/active length (m)	4.30/5.94
Heavy water reflector thickness (m)	0.4
Average outlet temperature (°C)	625
Mass flow rate (kg/s)	1320
Square lattice pitch (cm)	22
Number of fuel channels	300
Bundles per channel	12
Excess reactivity at BOEC/EOEC (mk)	140/4.4
CVR (mk)	−2.3/−3.5
Average fuel discharge burnup (GWd/tU)	38.1
MCST (°C)	841
Maximum channel power at BOC (kW)	11718
Maximum bundle power at BOC (kW)	1078
Maximum channel power at EOC (kW)	9979
Maximum bundle power at EOC (kW)	910
Average/maximum LHGR (kW/m)	33.1/50.6

is designed with an electrical power of 1220 MW and a thermal efficiency of 48%, thus, the thermal power should be 2540 MW. The operation pressure is 25 MPa, and the inlet coolant temperature is set to be 350 °C. The core parameters and operation conditions are selected for reaching:

- a thermal power of 2540 MW.
- a cycle length of at least 12 months.
- an average discharge burnup of at least 30 GWd/tU.

The following principles are considered to ensure fuel and core safety:

- MCST should be less than 850 °C.
- LHGR should be less than 60 kW/m.
- CVR should be negative during operation.

4.2. In-core fuel management

In order to flatten the power distribution and balance the bundle discharge burnup, an improved out-in fuel management scheme is used with three batch cycles and a cycle length of 350 EFPD. The initial core loading consists of three equal sized batches of fuel having an enrichment of 4.6%, 4.3% and 4.0% respectively. The initial core is clean and cold. At the end of the first cycle, batch 3 is removed from the core, batch 1 and batch 2 are rearranged according to the shuffling scheme, and a feed batch of fresh fuel (the enrichment is 4.6%) is placed into the core. The shuffling scheme, as shown in Fig. 6, is adjusted by experience. Take fuel group 1 as an example, bundles of 1_0 is fresh, at the end of a cycle it is moved to the position of 1_1, while bundles of 1_1 is moved to the position of 1_2 and bundles of 1_2 is moved out of core, and it is the same with the other 24 groups of fuel in one quarter of core. And it is worth mentioning that we always shuffling the bundles with the flow direction, because the axial power distribution is more flat by doing so.

The core flow direction distribution is not strictly like chessboard, as shown in Fig. 7. The principle to determine the core flow scheme is that the number of channels flow forward is equal to that of flow backward in every batch, and only in this way can make the average axial power distribution flat.

After the initial core is calculated, the burnup of bundles in batch 1 and 2 at EOC is known, and according to the shuffling scheme,

533 607	530 594	545 646							BOEC EOEC
609 651	571 638	576 641	516 594						
609 637	616 635	584 635	578 642	547 614	554 631				
663 635	623 629	611 635	627 639	608 628	582 615	561 631			
659 626	670 625	637 627	629 634	626 639	624 633	582 615	554 632		
701 625	668 626	662 630	672 632	635 632	626 639	609 629	547 615		
672 612	676 619	670 620	664 630	672 632	629 635	627 640	578 644	516 596	
684 608	678 607	706 618	670 621	662 631	637 628	611 637	584 637	576 644	545 649
692 598	707 603	678 607	676 620	668 627	670 627	623 631	616 637	571 641	530 597
709 611	692 598	684 608	672 613	701 626	659 627	663 637	609 639	609 653	533 609

Fig. 8. Coolant outlet temperature over the cycle.

0.85 0.97	0.92 1.03	0.73 0.86							BOEC EOEC
1.01 1.08	0.75 0.83	0.89 0.98	0.70 0.81						
0.87 0.91	1.08 1.11	0.87 0.94	0.83 0.92	0.74 0.83	0.79 0.90				
1.22 1.18	0.93 0.94	0.85 0.88	1.09 1.11	1.05 1.09	0.98 1.03	0.74 0.83			
0.99 0.95	1.25 1.18	1.07 1.06	0.91 0.92	1.01 1.03	1.10 1.12	0.98 1.03	0.79 0.90		
1.31 1.18	1.00 0.95	1.08 1.04	1.24 1.18	0.95 0.95	1.01 1.03	1.05 1.09	0.74 0.83		
1.01 0.93	1.13 1.05	1.11 1.03	0.96 0.92	1.24 1.18	0.91 0.92	1.09 1.11	0.83 0.92	0.70 0.81	
1.14 1.02	1.01 0.91	1.32 1.17	1.11 1.03	1.08 1.04	1.07 1.06	0.85 0.88	0.87 0.94	0.89 0.98	0.73 0.86
1.23 1.08	1.38 1.19	1.01 0.91	1.13 1.05	1.00 0.95	1.25 1.18	0.93 0.94	1.08 1.12	0.75 0.84	0.92 1.03
1.23 1.07	1.23 1.08	1.14 1.02	1.00 0.92	1.31 1.18	0.99 0.95	1.22 1.18	0.87 0.91	1.01 1.08	0.85 0.98

Fig. 9. Core radial relative power distribution (1/4 core).

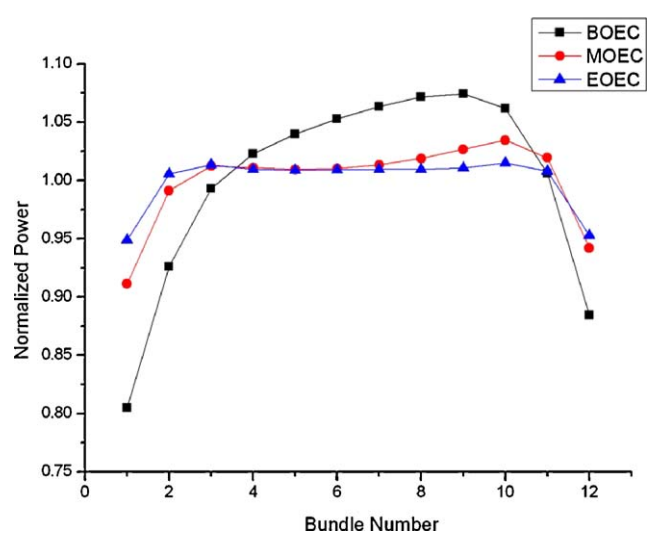

Fig. 10. Core average axial power distribution.

where they are reloaded at the second cycle is also known, so the burnup distribution at the BOC of the second cycle can be obtained. Then the second cycle is calculated. Repeat this process until the burnup of each bundle at BOC stays almost constant, which means the equilibrium cycle is reached. After that, find the burnup of bundles in G_2 channels and the average value is the average discharge burnup, other parameters of the equilibrium core can also be calculated.

4.3. Results of the equilibrium core

Table 4 shows primary system parameters of the equilibrium core. The excess reactivity of the equilibrium core at BOEC and EOEC is 140 mk and 4.4 mk respectively. The average discharged burnup of the equilibrium cycle is 38.1 GWd/tU, and the CVR is negative over the cycle, it is −2.3 mk at BOEC and −3.5 mk at EOEC.

Fig. 4 shows the core relative coolant flow distribution. As mentioned in Section 2.2, the coolant flow distribution is searched with a certain power distribution and is kept unchanged over a cycle, and the values are strongly related to that power distribution. If the power distribution of BOEC is chosen, the coolant flow distribution will not be suitable at EOEC because the power distribution of BOEC and EOEC is quit different, and the MCST at EOEC may become very high. Thus, the middle of the equilibrium cycle (MOEC) is selected to make a balance over the cycle.

The core coolant outlet temperature reaches to 625 °C averagely, and its distributions at BOEC and EOEC are given in Fig. 8. At BOEC, the coolant outlet temperature ranges from 530 °C to 710 °C, the coolant outlet temperature of inner channels is higher than that of peripheral channels. Further optimization should be made to reduce the coolant outlet temperature difference between channels. The distribution of the outlet coolant temperature becomes more uniform at EOEC because of the power distribution is flatter, and the coolant outlet temperature of peripheral channels becomes higher. The MCST is 841 °C, which is within the limitation of 850 °C. The MCST is calculated by single channel analysis, but the mass and heat transfer within the channel are not considered. The MCST will be evaluated by sub-channel analysis in the further study.

The core radial relative axial power distribution is shown in Fig. 9. Compared to the BOEC, the radial power distribution is flatter at EOEC. The radial power peaking factor is 1.38 at BOEC at 1.19 at EOEC. There is a tilt of axial power distribution at BOEC as shown in Fig. 10, and the axial power distribution becomes flatter and more symmetrical with burnup. The maximum LHGR is 50.6 kW/m and the average LHGR is 33.1 kW/m. The peak channel power is 11718 kW while the average channel power is 8467 kW, so the ratio of the maximum channel power to the average channel power is 1.38.

5. Conclusions

The 43-element HEC with a lattice pitch of 22 cm is selected as the fuel channel, the thickness of moderator is optimized to get negative CVR. A CANDU-SCWR equilibrium core is proposed with 3D full core neutronics/TH coupling. The core average core coolant outlet temperature reaches as high as 625 °C and a high thermal efficiency of 48% is achieved. The MCST is 841 °C, the maximum LHGR is 50.6 kW/m. The average discharged burnup is 38.1 GWd/tU, and the CVR keeps negative all over the cycle. The results obtained so far indicate that this design is feasible with a reasonable core performance. Further optimization of the lattice and core design is on going. The study on compensating the excess reactivity and the utilization of thorium fuel cycle in CANDU-SCWR will be carried out.

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