A Core Design Study of CANDU-SCWR by Three-dimensional Neutronics/Thermal-hydraulics Coupling

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Abstract

A CANDU-SCWR core is designed by using a 3D neutronics/thermal-hydraulic coupling method. In the fuel channel design, a typical 43-element fuel bundle is used, the coolant is supercritical light water, and the moderator is heavy water, the thickness of which is optimized to ensure the negative coolant coefficient during operation. The core has a power of 1220 MWe with a diameter of 4.8m and length of 4.95m, and there are totally 300 fuel channels, each of which consists of 10 fuel bundles. The coolant inlet temperature is set to be 350 ℃ and the operation pressure is 25 MPa. In order to flatten the radial power distribution, the loading pattern of the equilibrium cycle is optimized, and an improved in-out fuel management scheme is used with three batches refueling, burnable poison Dy_2O_3 is used to flatten the power peaking. The numerical results show that the average power density is 42.75 W/cm^3, while the maximum linear element rate(LER) is 575W/cm. The average discharged burnup of the equilibrium is 48.3GWD/tU, and a high average coolant outlet temperature of 625 ℃ is achieved with a maximum cladding surface temperature less than 850 ℃. Besides, the coolant temperature coefficient is negative throughout the cycle.

Keywords: CANDU-SCWR, Core design, 3D neutronics/thermal-hydraulics.

1. Introduction

The supercritical water reactor(SCWR) has been regarded as one of the Generation IV reactor concepts, and it has lots of advantages compared to PWRs and BWRs. Using supercritical water as working fluid, it could get a very high steam enthalpy, which means a higher thermal efficiency. And SCWR would operate on a direct cycle, which eliminates steam generators and makes the core simpler and more compact. Besides, high enthalpy rise of supercritical water makes low flow rate possible and reduces the main pump power. Furthermore, single-phase coolant prevents boiling transition or dry-out phenomenon. For these advantages, the role of SCWR has been emphasized and lots of research activities on SCWR are ongoing worldwide.

The SCWR system follows two main types, one is pressure vessel type, like PWRs or BWRs, the other is pressure tube type, like CANDU and RBMK reactors. The pressure tube type is characterized as simple, economical fuel bundle, multi-pass reactor flows , separated cool and moderator, back-up heat sink capability, by which inherently safety, reheat and flexible fuel cycle is achieved. One of the pressure tube type concept is CANDU-SCWR, which was raised by AECL in recent years. Khartabil et al have put forward some preliminary design parameters of CANDU-SCWR(Khartabil et al,2005). Based on these parameters, we perform some core design studies using three-dimensional neutronics thermal-hydraulics coupling method. This paper also includes preliminary fuel channel design to get negative coolant temperature coefficient.

2. Design goals and criteria

CANDU-SCWR should be competitive to survive in nuclear market. In order to achieve highly economical efficiency, the design goals of the reactor can be summarized as following:

(1) The core should be a 1000MWe class commercial scale.

(2) Core average outlet temperature should be around 625 ℃,which correspond to 48% thermal efficiency.

(3) Core average power density should be over 40 W/cm^3.

(4) Average fuel assembly discharge burnup is about 45MWd/tU.

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

(1) Maximum cladding surface temperature(MCST) should be less than 850 ℃.

(2) Linear element rate(LER) should be less than 60kW/m.

(3) Negative coolant temperature coefficient during operation.

3. Design Methods

3.1 Nuclear Design Method

The SRAC code has been used as the neutronics solver(K. Okumura et al, 2006). It is a multipurpose code system applicable to neutronics analyses of a variety of reactor types including CANDU reactors. The PIJ code of SRAC based on integral neutron transport method called as collision probability method (CPM) is used to perform the assembly transport calculation from the 62-fast and 45-thermal energy groups of the JENDL-3.3 nuclear data library to get few group cross-section. Few group cross section sets are prepared as a function of water density and burnup. For a given water density and burnup, the few group cross section can be interpolated.

Core depletion calculation is provided by COREBN code, which is an auxiliary code of the SRAC system for multi-dimensional core burnup calculation based on CITATION module(K. Okumura et al, 2007).

3.2 Thermal Hydraulic Calculation

Thermal hydraulic calculation is based on single-channel model. All fuel assemblies are averaged into single rod approximation. And there are two kinds of single rod models, one is the peak fuel rod and the other is the average fuel rod. Both of the two models have the same thermal hydraulic parameters but different power distributions. The peak rod has the peak liner generation rate in a fuel assembly, which is used to calculate the mass flux distribution at each assembly inlet and the peak cladding surface temperature in each assembly. While the average rod has the average linear heat generation rate in an assembly, and it is used to evaluate coolant temperature and density. Suppose we already calculate the power distribution in the core, in order to keep cladding surface temperature below limiting value, the mass flow rate is searched by the peak rod model and the maximum flow rate is taken. The mass flux distribution will not change due to burnup. Then, the coolant density distribution is calculated by average rod single channel model.

In order to couple the neutronics/thermal-hydraulics analysis, we developed a control module, which can exchange the data between COREBN and the thermal-hydraulics code, as shown in Fig. 1. Power distribution and coolant density are solved iteratively until the coolant density is converged.

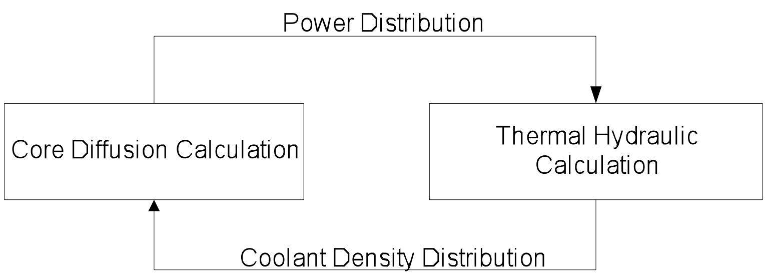

Fig. 1. Data exchange in coupling the neutronics/thermal-hydraulics analysis

4. Fuel channel design

In order to reduce the linear element ratings and to enhance the fuel performance, the 43-element fuel bundle is used as the fuel channel rather than 37-element fuel bundle, the 61-element fuel bundle will be considered in the further study because it could get a better results. For the 43-element fuel bundle, the material selections are shown in Table 2(C.K.Chow et al). The center pin contains burnable poison (U, Dy)O2 pellet with 7.5%wt Dysprosium in natural Uranium so that the flux could be flattened and the negative coolant temperature coefficient could be get easier, the other 42 pins are loaded with the same enrichment of UO_2.

Table 1 The 43-elment Fuel Channel Preliminary Material Selection

Name	Material
Fuel	UO_2

Enrichment	4%
Fuel cladding	Incoloy 800
Metal linear	0Cr13Al
Insulator	ZrO_2
PT	Zircaloy 2

Unlike CANDU reactors, CANDU-SCWR use enriched uranium as fuel, so it doesn't need as much heavy water as CANDU reactors, which makes it possible to reduce the heavy water inventory to make the core design more compact and improved the reactor physics characteristics. In this paper, the radial fuel assembly square lattice pitch is 200mm.

Considering the pressure tube suffers excessive corrosion at high temperature, ceramic material is placed inside the pressure tube to insulate the pressure tube from the coolant, and calandria tube is eliminated. There are small openings in the linear and insulator, so the coolant pressure is taken directly by the pressure tube (C.K.Chow et al), as shown in Fig. 2.

Preliminary fuel channel design principle is to keep coolant temperature coefficient negative. As we know, when the coolant temperature goes up, the coolant density goes down, which cause 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. Our study shows that the coolant temperature coefficient is positive when the channel is over moderated, so a under moderated channel design is chosen, in which the average thickness of the heavy water is 37.78mm, the other parameters are listed in table 2.

Table 2 The 43-elment Fuel Channel Preliminary Dimensions

Parameter	Value
Metal linear inside diameter/wall thickness, mm	104.11/1.00
Insulator inside diameter/wall thickness, mm	106.11/15.00
PT inside diameter/wall thickness, mm	136.11/7.00
Center/first ring/Second ring/Third ring diameter, mm	0/34.85/61.79/88.08
Center rod & first ring of fuel elements diameter, mm	13.53
Second ring & third ring of fuel elements diameter, mm	11.52
Fuel cladding wall thickness, mm	0.40
Length of bundle, mm	495.3

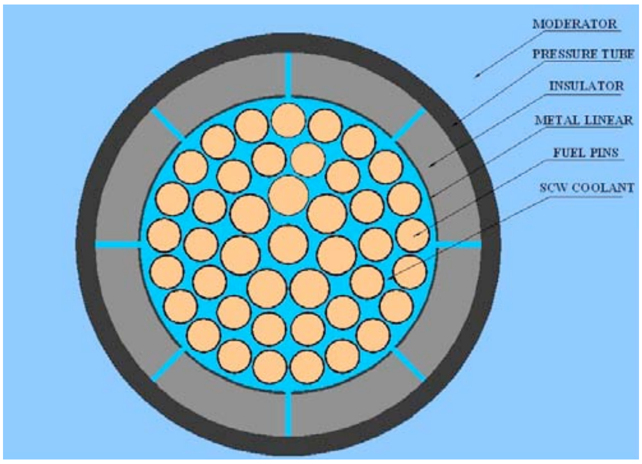

Fig. 2. The 43-element fuel channel

However, for this design, when the burnup gets higher, the coolant temperature coefficient tends to become positive. As shown in Fig. 3, when the burnup is 50GWd/tU and the coolant temperature is over 390 ℃, the coolant temperature coefficient is slightly positive. But it's acceptable because there are only few bundles get such a high burnup in the core at a time, so the core coolant temperature coefficient is still negative.

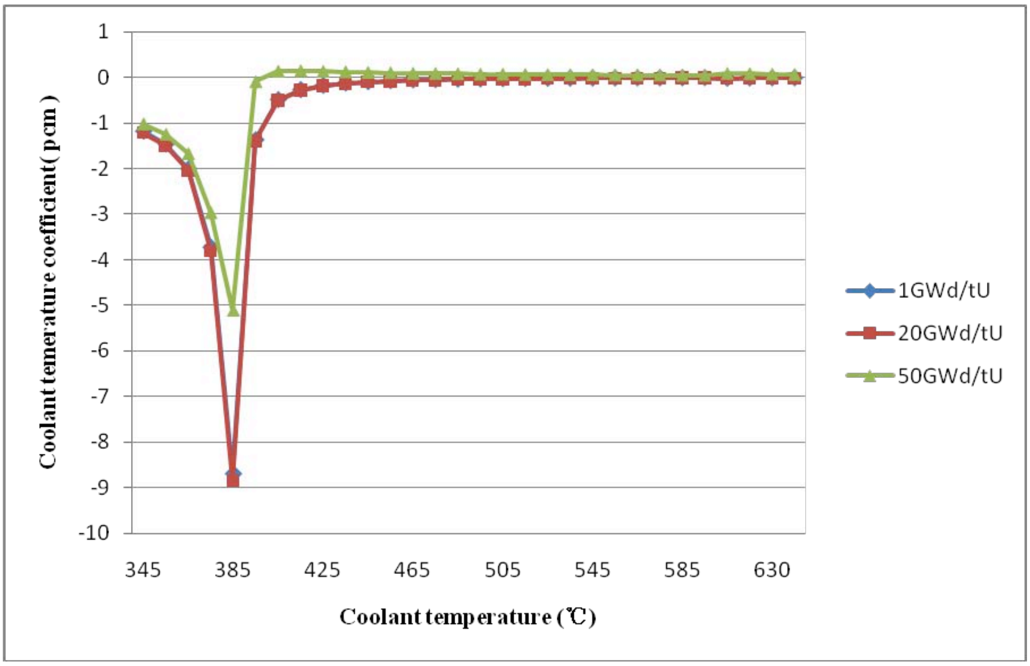

Fig. 3. Temperature-dependence of coolant temperature coefficient

5. Core design

Table 3 shows the preliminary design parameters for the CANDU-SCWR, which is mainly based on the parameters proposed by Khartabi at 2005.

Table 3 The preliminary design parameters for the CANDU-SCWR

Parameter	Value

Thermal power, MW	2540
Electric power, MW	1220
Thermal efficiency,%	48
Operation Pressure, MPa	25
Inlet temperature,℃	350
Average outlet temperature,℃	625
Core diameter, m	4.8
Core length, m	4.95
Radial reflector thickness, m	0.4
Number of fuel channels	300
Number of bundles per channel	10
Lattice pitch, mm	200(square)
Total core mass flow rate, kg/s	1350
Cladding Temperature, ℃	<850

Unlike the traditional CANDU reactors, CANDU-SCWRs don't use NU as fuel, so the on line refueling scheme is replaced by batch refueling, which is much more like LWRs. In order to flatten axial power distribution and balance the assembly discharge burnup, an improved out-in fuel management scheme is used with three batch cycles and 350EFPD cycle length. The initial core is loaded with three batches of fuels, the enrichment of which are 4.7%, 4.3% and 4.0% respectively, see Fig. 4. And the core arrangement, flow direction of each channel and its loading pattern of the equilibrium cycle are given in Fig. 4, where "G_C" stands for the shuffling scheme, "G" means the fuel management group number and "C" means how many cycles the fuel has burnt(0 means fresh). To explain the shuffling scheme, take 1_C as an example, assembly 1_0 is fresh fuel with an enrichment of 4.7%, at the end of a cycle it is moved to the position of 1_1, while assembly 1_1 is moved to the position of 1_2 and assembly 1_2 is moved out of core.

The initial core loading pattern is used to do the first cycle calculation and the burnup of each assembly at the end of the first cycle can be obtained. Then we use the shuffling scheme(see Fig. 4) to shuffle the bundles, after that the second cycle will be calculated and we will get the burnup of each assembly at the end of the second cycle. Repeat this process until the burnup of each assembly stays almost constant, which means it is the equilibrium cycle. When equilibrium cycle is got, find the burnup of assembly 1_2, 2_2, … , 25_2 and the average value is the average discharge burnup.

Since there is a large coolant density change across the core, shuffling the bundles with or against the flow direction will be different. In this paper, the direction of refueling the bundle doesn't change over the cycle, we change the flow direction of each channel instead so as to make the average axial power distribution more symmetrical. The final flow scheme is shown in Fig. 4.

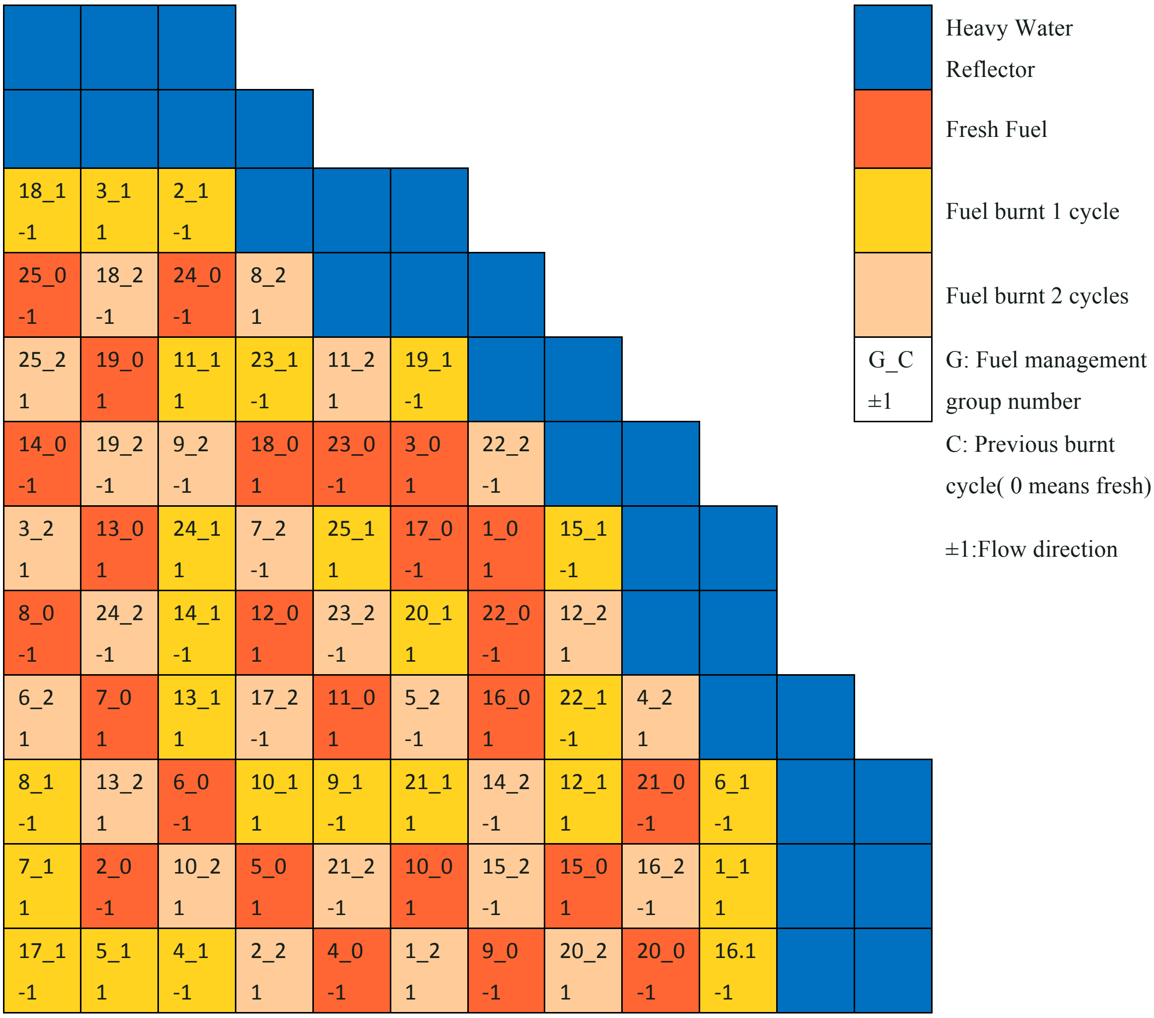

Fig. 4. 1/4 core arrangement

Table 4 shows some results of the equilibrium cycle, from which we can see that radial power peaking factor is 1.355 at BOEC and 1.219 at EOEC, that the coolant temperature coefficient is negative at both BOEC and EOEC, and that the k-effective is higher than 1 over the cycle since compensating the excess reactivity is not considered yet. The average coolant outlet temperature is 625 ℃. The total mass flow rate is 1350kg/s, which is much less than the PWRs with the same electric power. The average power density is 42.75W/cm^3, and the maximum LER is 575W/cm. All of these values are within the limiting design criterion. And Fig. 5 shows the k-effecitve trends as a function of EFPD.

Table 4 K-eff, Radial power peaking factor and coolant temperature coefficient over the cycle

Operation day	0 EFPD	50 EFPD	100EFPD	170EFPD	250EFPD	350EFPD
K-eff	1.203	1.143	1.122	1.094	1.063	1.025
Radial power peaking factor	1.355	1.309	1.285	1.258	1.24	1.219

Coolant temperature coefficient (pcm)	-5.12	-4.91	-4.35	-3.30	-2.14	-0.63

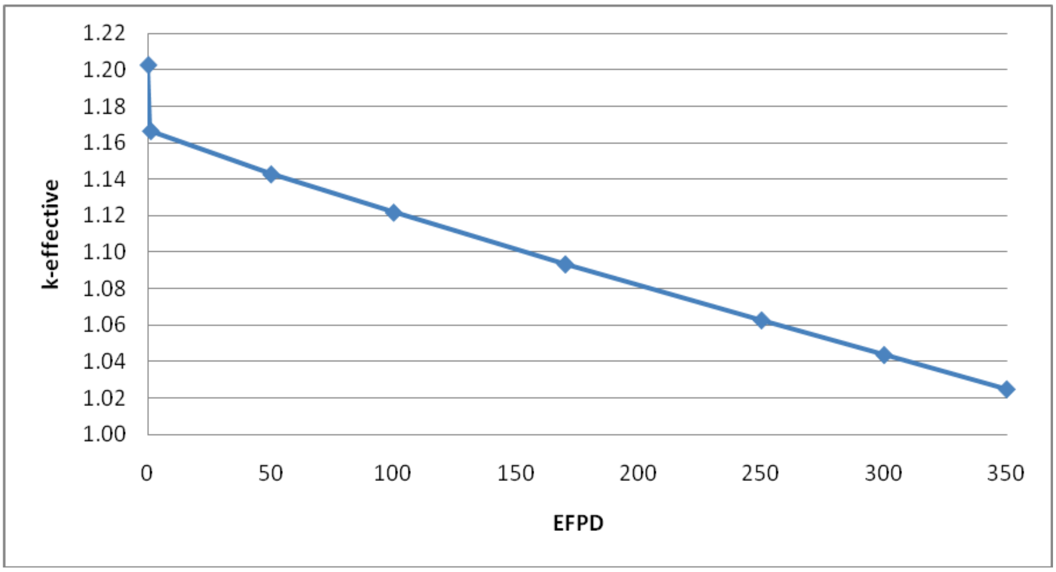

Fig. 5. K-effecitve trends as a function of EFPD

The mass flux distribution in the core searched to satisfy the limitation of MCST at the middle of the equilibrium cycle(MOEC), and the relative mass flux distribution is obtained by dividing the mass flux of each channel by the average mass flux. Fig. 6 shows the relative mass flux distribution, the negative means that coolant in these channels flow in the opposite direction. After the mass flux distribution is obtained, it is kept unchanged over the cycle.

-0.97	0.95	-0.95							
-1.07	-0.77	-1.01	0.89						
0.83	1.12	0.90	-0.90	0.86	-1.08				
-1.19	-0.86	-0.80	1.15	-1.17	1.16	-0.96			
0.90	1.20	1.03	-0.83	1.02	-1.19	1.16	-1.08		
-1.22	-0.89	-1.01	1.19	-0.88	1.02	-1.17	0.86		
0.87	1.21	1.02	-0.86	1.19	-0.83	1.15	-0.90	0.89	
-0.98	0.86	-1.22	1.02	-1.01	1.03	-0.80	0.90	-1.01	-0.94
1.00	-1.19	0.86	1.21	-0.89	1.20	-0.86	1.12	-0.77	0.95
-0.99	1.00	-0.98	0.88	-1.22	0.90	-1.19	0.83	-1.06	-0.97

Forward direction
Backward direction

Fig. 6. Relative mass flux distribution

The coolant outlet temperture and the MCST are given in Figs. 7-8, where “Max.” means the maximum value over the cylce and “Min” means minimum. Fig. 7 shows that the coolant outlet temperature changed with burnup most rapidly at the outer most region and the inner most region, that is because the flow mass flux distribution is calculated based on the power distribution of the MOEC, the power distribution of BOEC and EOEC is quite different from that of the MOEC at the outer most and inner most region. The middle region stays almost uniform along with the burnup. From Fig. 8, we can see that MCST is 818.4 ℃ all through the core over the cycle, which is within the limitation of 850 ℃. The MCST is calculated by single channel analysis based on the power distribution of an assembly, but the mass and heat transfer within the channel are not considered. The MCST will be evaluated by subchannel analysis in our further study.

594.4 504.7	581.9 490.7	579.4 492.1							Max. Min.
632.5 562.2	537.7 621.9	632.0 550.6	549.1 486.3						
621.6 576.0	617.5 577.3	619.3 566.9	619.3 568.6	577.6 538.1	556.5 533.9				
619.0 610.2	605.1 590.1	614.7 586.8	606.5 601.0	606.8 600.0	600.2 592.2	549.2 535.6			
630.7 602.5	634.9 601.8	616.4 600.3	618.9 614.0	629.8 609.5	636.5 608.2	600.2 592.6	557.2 534.0		
663.9 597.4	651.8 605.1	641.6 598.9	648.3 604.9	624.7 604.1	629.8 609.8	606.8 600.7	578.7 538.1		
643.7 590.5	668.1 595.2	654.7 592.4	641.4 601.9	648.4 605.2	618.4 613.8	607.1 601.0	620.6 568.5	550.6 486.2	
652.2 591.2	651.4 590.9	673.7 593.8	654.8 592.6	641.6 599.3	616.6 601.0	615.9 586.9	620.9 566.9	634.1 550.4	582.0 492.1
652.1 584.3	675.2 593.9	651.7 591.1	668.6 595.7	652.0 605.6	635.2 602.5	606.5 590.2	619.3 577.4	624.0 537.6	584.1 490.5
655.9 588.9	652.2 584.3	652.7 591.6	591.4 645.1	664.6 631.2	631.2 603.4	619.2 611.8	623.5 576.1	634.9 562.1	597.1 504.7

Fig. 7. Coolant outlet temperature over the cycle

813.5 637.7	814.6 634.9	813.3 641.2							Max. Min.
794.9 690.2	803.3 679.4	801.5 681.1	804.1 661.5						
780.9 720.1	776.6 719.0	782.5 702.9	779.5 713.8	789.5 690.7	772.8 704.8				
761.8 753.3	764.7 749.6	771.6 733.1	761.7 744.7	758.3 753.6	758.7 751.9	773.3 714.1			

780.8 747.3	780.3 744.9	765.1 747.5	761.5 757.2	757.7 751.0	767.8 749.1	759.2 751.9	773.8 704.8		
802.8 734.5	799.7 747.8	785.4 737.4	779.1 741.9	774.8 747.9	757.7 751.3	758.4 753.6	791.0 690.6		
796.2 740.9	804.5 732.5	803.1 731.2	786.6 742.9	779.2 742.3	760.8 756.8	762.7 744.8	781.3 713.8	806.7 661.3	
796.1 731.2	801.5 738.1	812.6 728.7	803.3 731.4	785.7 736.9	765.2 748.5	773.3 733.5	784.8 703.0	804.5 681.2	817.0 640.7
804.0 725.7	812.6 726.9	801.7 738.3	804.8 732.8	800.5 748.7	780.7 746.1	766.6 750.1	779.0 719.1	806.5 679.5	818.4 634.6
796.5 728.0	804.1 725.7	796.4 731.4	797.0 741.6	803.9 735.5	781.3 748.6	762.5 754.9	783.3 720.2	797.9 690.4	817.4 637.6

Fig. 8. Maximum cladding surface temperature over the cycle

The core axial and radial power distributions are given in Fig. 9-10, from which we can see that the power distribution becomes more flat with burnup, that the axial power distribution is almost symmetrical, which benefits for improving the reactor physics characteristics.

The average discharge burnup of the equilibrium cycle is 48.3GWd/tU, which may be a little high for such a tight lattice pitch. However, to reduce the discharge burnup means either to shorten the cycle length or to decrease the core power density, which is not economical for SCWRs. Therefore, we must balance these feathers to get an improved core design.

0.81	0.77	0.77							
1.00	0.69	0.92	0.71						
0.80	1.08	0.86	0.85	0.78	0.96				
1.23	0.84	0.79	1.15	1.18	1.16	0.86			
0.95	1.27	1.06	0.85	1.07	1.26	1.16	0.96		
1.34	0.96	1.08	1.28	0.91	1.07	1.18	0.78		
0.93	1.33	1.11	0.91	1.28	0.85	1.15	0.85	0.71	
1.06	0.92	1.36	1.11	1.08	1.06	0.79	0.86	0.92	0.77
1.08	1.33	0.93	1.33	0.96	1.27	0.84	1.08	0.69	0.76
1.08	1.08	1.06	0.94	1.34	0.95	1.23	0.80	1.00	0.81

0.96	0.92	0.92							
1.12	0.80	1.06	0.82						
0.85	1.15	0.93	0.93	0.83	1.01				
1.21	0.86	0.82	1.16	1.17	1.15	0.88			
0.91	1.21	1.03	0.84	1.04	1.21	1.15	1.01		
1.22	0.90	1.01	1.20	0.88	1.04	1.17	0.84		
0.86	1.20	1.01	0.86	1.20	0.84	1.16	0.93	0.82	
0.97	0.84	1.21	1.01	1.01	1.03	0.82	0.93	1.06	0.92
0.98	1.18	0.84	1.20	0.90	1.21	0.87	1.15	0.80	0.92
0.98	0.98	0.97	0.87	1.22	0.91	1.21	0.86	1.12	0.96

Fig. 9. Core radial relative power distribution (BOEC, EOEC)

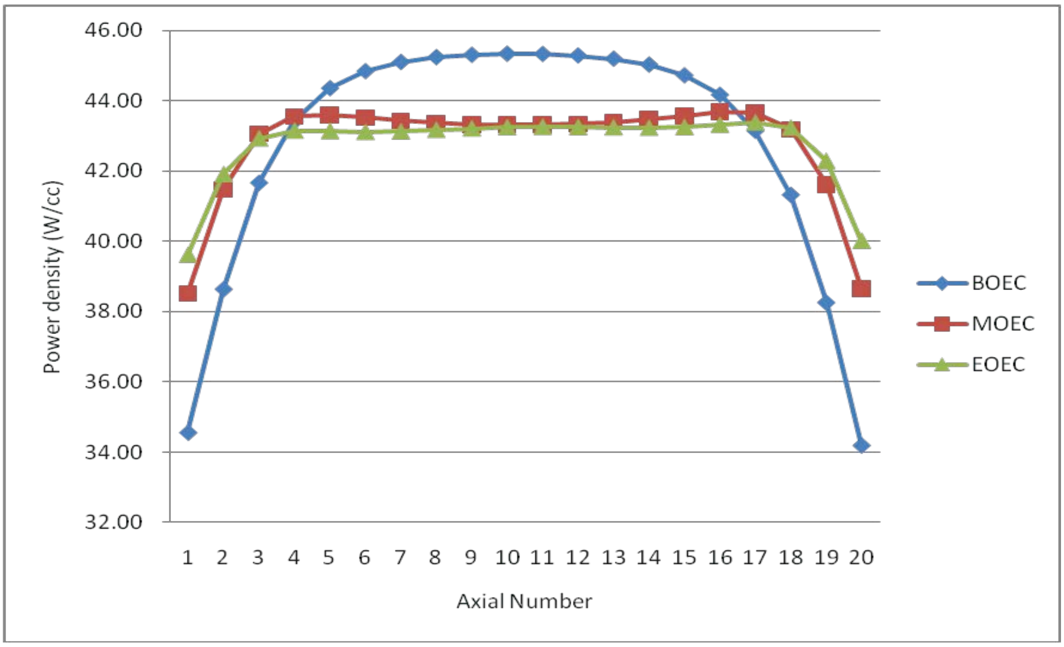

Fig. 10. Core average axial power distribution

6. Conclusion

Fuel channel uses 43-element fuel bundle with assembly lattice pitch of 200mm, the thickness of heavy water is optimized to get negative coolant temperature coefficient. A CANDU-SCWR core design is proposed by using 3D neutronics/thermal-hydraulic coupling method. The core is designed to be a class commercial scale with a power 1220MWe with a diameter of 4.8m and a length of 4.95m. The average coolant outlet temperature reaches as high as 625 ℃ and a high thermal efficiency of 48% is achieved. The flow rate is 1350Kg/s, which is much lower than that of PWRs with the same power. The maximum cladding surface temperature is 815 ℃, the average discharged burnup of the equilibrium is 48.3GWD/tU, the average power density is 42.75 W/cm^3 and the maximum linear element rate is 575 W/cm, all of which are within the limiting design criteria. The coolant temperature coefficient of reactivity keeps negative all over the cycle.

7. Acknowledgement

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