THERMAL-HYDRAULICS ANALYSIS OF ANNULAR FUEL APPLICATION IN SCWR

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Abstract

The annular fuel is a new fuel type with annular shape which allows both the internal and external cooling abilities. First, two coolant flow schemes, which are named in-out type and out-in type, have been compared with single channel model from both aspects of physics and thermal-hydraulic, and the latter one was finally selected. Then, with the selected flow scheme, subchannel analysis with modified ATHAS code was carried out. The results showed that application of annular fuel in supercritical water reactor can reduce both cladding temperature and fuel pellet temperature effectively.

1. Introduction

The supercritical water-cooled reactor (SCWR) is essentially a pressurized water reactor operating above the thermodynamic critical point of water. It is considered as one of the most promising Generation IV reactors because of its simplicity, high thermal efficiency, and nearly fifty years of industrial experience from thermal-power stations with a SCW cycle.[1] Like other pressurized water reactors, the SCWR also uses traditional solid fuel. However, some problems exist in this SCWR loaded with solid fuel. This reactor has high cladding and pellet center temperature, which are very close to their limiting values. If a loss of coolant accident (LOCA) happens, the core will be easily burned. It can be said that the safety performance of this SCWR is not excellent.

Lately, MIT has proposed an internally and externally double-cooled annular fuel, which can endure a substantial power uprating.[2] The annular fuel rod has two cooling surfaces: the outer cladding in contact with the coolant flowing in the outer channel and the inner cladding in contact with the coolant flowing in the inner channel. The larger cooling surface results in a significantly higher Departure from Nucleate Boiling (DNB) margin. And, with lower fuel temperatures, the annular fuel provides significant benefits in terms of a low peak cladding temperature following a Loss of Coolant Accident (LOCA).[3] Therefore, loaded with annular fuel rods, PWR can greatly enhance safety performance under any accidents. Compare with traditional solid fuel rod, the annular fuel rod will increase complexity of fuel design and manufacturing. It has proved that the current press, sinter and grind technology used in current nuclear fuel plants could be adapted to the annular fuel, and no major changes in fuel manufacturing processes would be required.[4]

Based upon the above, we have started an innovative project for applying the proposed annular fuel to SCWR. This reactor will be expected to combine the advantages of SCWR and annular fuel, and greatly improve the economy and safety performance. In order to analyze the thermal-hydraulic characteristics of the annular fuel assembly used in SCWR, we give a reference SCWR design. The operating parameters of SCWR loaded with annular fuel are shown in Table 1.

In this paper, we will make thermal-hydraulics analysis for annular assembly used in SCWR with single channel model and subchannel model.

Table 1 Summary table for the SCWR reference design

Parameter	Value
Thermal power	3022MW
Operating pressure	25MPa
Reactor inlet/outlet temp	280/500℃
Reactor flow rate	1561kg/s
Fuel pin lattice	Square
Assembly side	292.2mm
Assembly array	17×17
Number of fuel assemblies	121
Heated length	4.27m

2. Single channel analysis

Before making subchannel analysis for fuel assembly, we should carry out preliminary thermal hydraulic analysis for single annular fuel element, namely single channel model analysis. With single channel model, we can calculate rapidly the parameters of the channel, such as coolant temperature, density, fuel pellet and cladding surface temperature. The single channel model is composed of annular fuel rod, inner and outer coolant channel.[5] Its dimensions and configuration are shown in Figure 1.

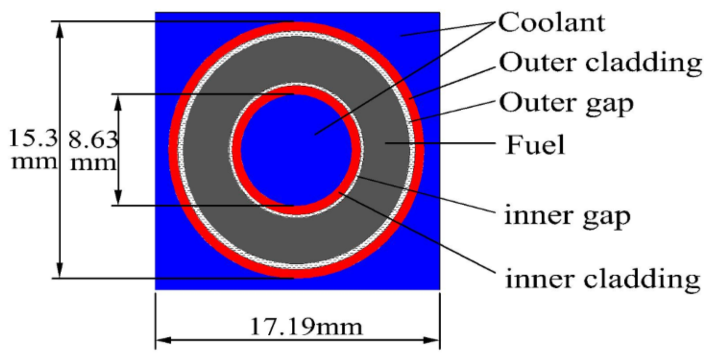

Figure 1 Single channel model

Based upon the single channel model, we put forward two cooling schemes. 1) Coolant flows into inner channel and flows out of outer channel, and called in-out type. 2) Coolant flows into outer channel and flows out of inner channel, and called out-in type. The two cooling scheme are shown in Figure 2.

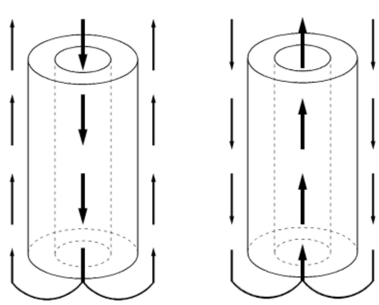

Figure 2 In-out type (left) and out-in type (right)

2.1 1-D conduction model

The conduction model used in single channel model involves a number of assumptions. Important assumptions are as follows. 1) Heat conduction is ignored in the axial and azimuthal direction. 2) The axial power distribution exhibits a symmetric cosine profile. [6]

The cylindrical coordinate 1-D steady-state conduction equation and boundary condition are listed below:

$$\frac{d^2T}{dr^2}+\frac{1}{r}\frac{dT}{dr}+\frac{Q_v}{k}=0 \tag{1}$$

$$BC: at\ r=r_1, -k_c\left.\frac{\partial T_C}{\partial r}\right|_{r=r_1}=H_{wi}\left(T_{cii}-T_{bi}\right) \tag{2}$$

$$at\ r=r_6, -k_c\left.\frac{\partial T_C}{\partial r}\right|_{r=r_6}=H_{wo}\left(T_{coo}-T_{bo}\right)$$

Here, Q_v is volumetric heat generation rate, k is thermal conductivity, r_1 and r_6 are inner and outer radii of the annular fuel rod, respectively. H_{wi} and H_{wo} are heat transfer coefficients of inner and outer channels. T_{bi} and T_{bo} are inner and outer coolant temperature. The heat transfer coefficient between cladding surface and coolant is calculated by Dittus-Boelter correlation.[7]

$$Nu_b=0.0243\,\mathrm{Re}_b^{0.8}\,\mathrm{Pr}_b^{0.4} \tag{3}$$

If T_{bi} and T_{bo} are given, the temperature distribution of cladding, gas gap and fuel pellet can be calculated. A schematic drawing of the temperature distribution and radial node location are presented in Figure 3.[6]

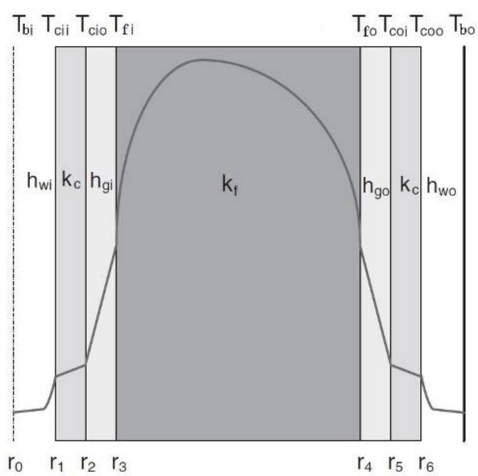

Figure 3 Radial temperature profile and node location

2.2 Results and Discussion

The axial distributions of coolant temperature, density and cladding surface temperature are calculated with the single channel model, and shown in Figure 4, Figure 5 and Figure 6. In these figures, the axial zero location represents coolant flow inlet and outlet.

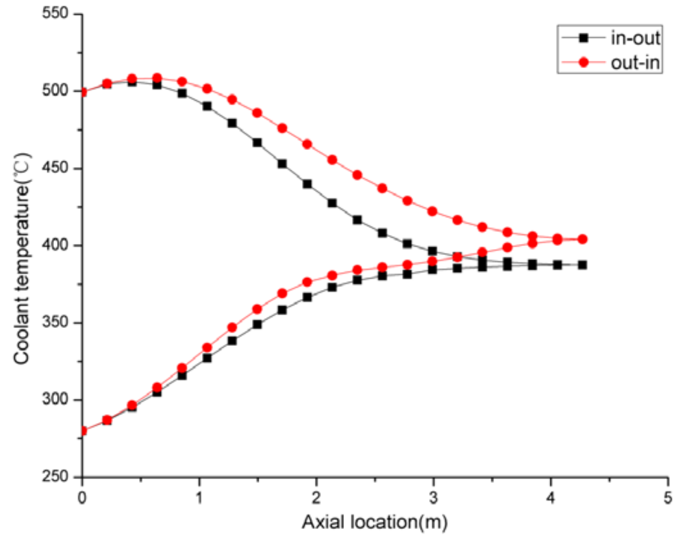

Figure 4 Axial distribution of coolant temperature

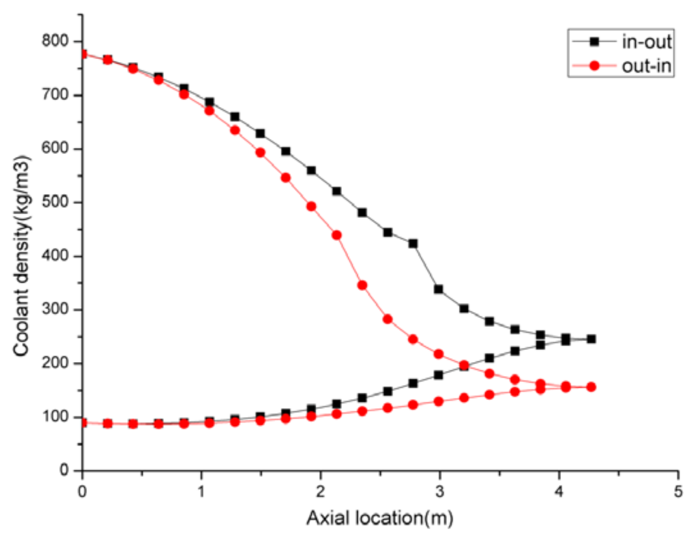

Figure 5 Axial distribution of coolant density

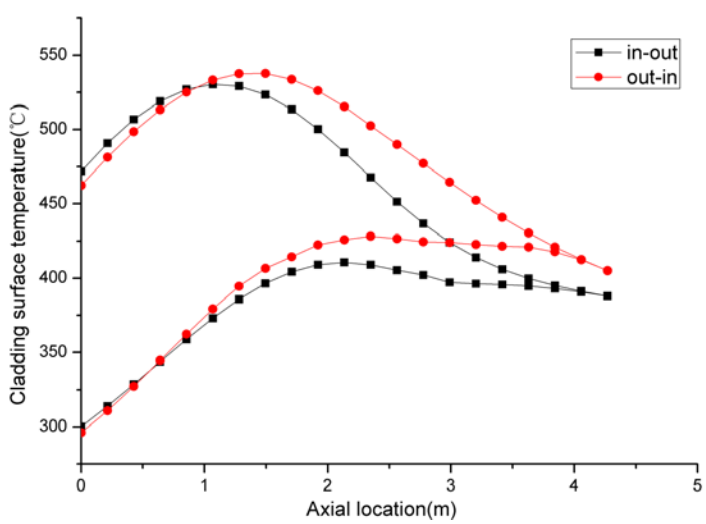

Figure 6 Axial distribution of cladding surface temperature

As can be seen in Figure 4, in in-out scheme, the inner channel has lower coolant temperature than the outer channel. This is because coolant flows into inner channel firstly and flows out of outer channel finally. In out-in scheme, the situation is on the contrary. From the aspect of thermal-hydraulics, if the inner coolant temperature is higher than the outer, the maximum cladding surface temperature (MCST) will occur in the inner channel. However, the inner channel is independent channel in fuel assembly, and it doesn't have mass and energy exchange with other channels. So the cladding is easier to burn out in out-in scheme. If the maximum cladding surface temperature (MCST) occurs in the outer channel, the MCST can be decreased by mixing coolant among channels. As can be seen in Figure 5, there is a kink at the axial location for two cooling schemes. It's because that the temperature at this point reaches pseudo-critical temperature, the coolant density changes greatly. In addition, the in-out scheme has much higher inner coolant density than the out-in scheme. From the aspect of physics, the supercritical water of inner channel in in-out scheme will have better moderating performance. As can be seen in Figure 6, the in-out scheme has lower MSCT than out-in scheme. In conclusion, the in-out scheme will be the better choice.

3. Subchannel analysis

3.1 Coolant flow distribution

In the selected in-out scheme, all coolant flow averagely into inner subchannels of fuel assembly firstly. After flowing out of inner subchannels and mixed in the lower plenum, the coolant flow should be distributed into each outer subchannel in the manner of equaling pressure drops. [8]

The friction pressure drop is calculated by

$$\Delta P = \left(\frac{f\,\Delta z}{D_e} + k \right) \frac{\rho}{2} u^2 \tag{4}$$

Where f is the friction coefficient and k is the friction coefficient at the spacer grid. The f is calculated by Blasius' correlation.

$$f = 0.3164 \cdot \mathrm{Re}^{-0.25} \tag{5}$$

The k-factor is calculated using Weisbach's formula.[9]

$$k = \left(\frac{1}{C_c \varepsilon} - 1 \right)^2 \tag{6}$$

Where ε is the obstruction area ratio, C_c is a coefficient.

$$\varepsilon = \frac{A_s - A_k}{A_s} \tag{7}$$

$$C_c = 0.6079 + 0.1739\varepsilon - 0.3382\varepsilon^2 + 0.5544\varepsilon^3 \tag{8}$$

Here A_k is the projected area of a grid spacer in the sub-channel and A_s is the flow area of the sub-channel considered.

3.2 2-D heat conduction model

The annular fuel assembly has inner and outer coolant channels. Only part of the heat rate generated by fuel pellet will be transferred to the external coolant in the open fuel pin array, and the remaining part will be transferred to the internal coolant flowing through the inner channels. To obtain the detailed temperature distribution in the fuel rod and more accurate results, the 2-D annular fuel heat conduction model is needed. The conduction equation in the cylinder is solved for the annular fuel rod using the thermo-physical properties for cladding, gas gap and pellet respectively. The cylindrical coordinate 2-D conduction equation and boundary condition are listed below:

$$(\rho C_P)_r \frac{\partial T}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}(k \cdot r \frac{\partial T}{\partial r}) + \frac{1}{r^2}\frac{\partial}{\partial \theta}(k \frac{\partial T}{\partial \theta}) + Q_v \quad (9)$$

$$BC: at\ r = r_1, -k_c \left.\frac{\partial T_C}{\partial r}\right|_{r=r_1} = H_{wi}\left(T_{cii} - T_{bi}\right) \quad (10)$$

$$at\ r = r_6, -k_c \left.\frac{\partial T_C}{\partial r}\right|_{r=r_6} = H_{wo}\left(T_{coo} - (T_{bo})_i\right)$$

Here, $(T_{bo})_i$ is the coolant temperature of outer subchannel, where each numerical index corresponds to each outer subchannel around the annular fuel rod.

3.3 Calculation scheme of modified ATHAS

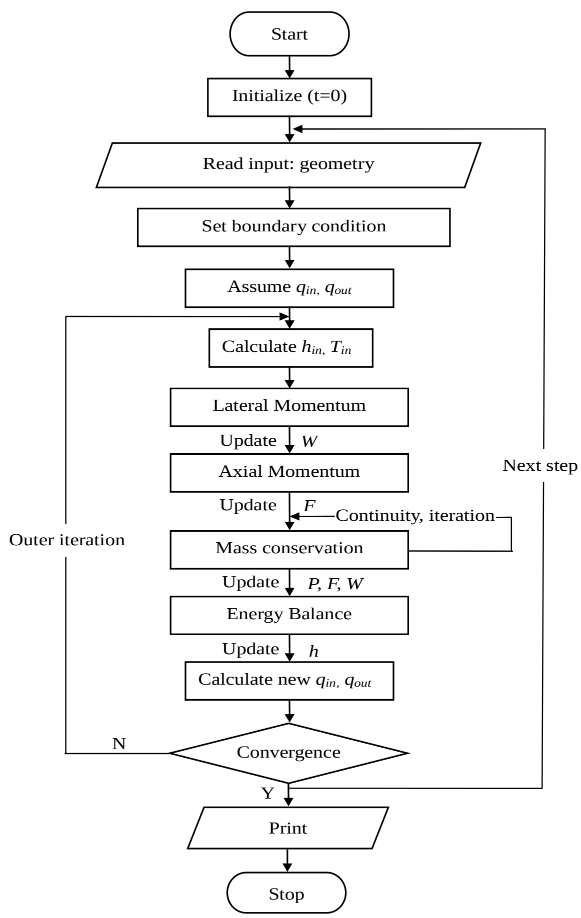

Figure 7 Calculation scheme of modified ATHAS

ATHAS is a subchannel code developed by Xi'an Jiaotong University. It can calculate the thermal-hydraulic conditions of traditional solid fuel assembly.[10] After added the 2-D heat conduction model of annular fuel, the modified ATHAS has the capability to model annular fuel assembly. Figure 7 illustrates the calculation scheme of this code.

This code is developed on the basis of mass, energy, and momentum (lateral and axial directions) conservative equations. After initializing of all parameters, the code reads in the subchannel configuration of the assembly, initial and boundary conditions (i.e., pressure, mass flow rate, coolant inlet temperature, and power), power distributions, and assumed heat fluxes of inner and outer subchannels. And then the calculation starts with the outer iteration to solve the momentum, mass, and energy based on assumed flows in all subchannels. Separate iterations are required to solve the mass and energy equations. Before each end of outer iteration, new heat fluxes of inner and outer subchannels should be calculated with the heat conduction model of annular fuel. The outer iteration is considered terminate after both the axial flow and energy convergence. Calculated results are outputted and the calculation terminates in steady-state analyses or proceeds to the next time step in transient analyses.

3.4 Fuel assembly configuration

Figure 8 illustrates the 17×17 annular fuel assembly configuration. A fuel assembly is comprised of 289 fuel rods, and the rods are arranged with square array (*P/D*=1.12). 10 grids are arranged along axial assembly location.

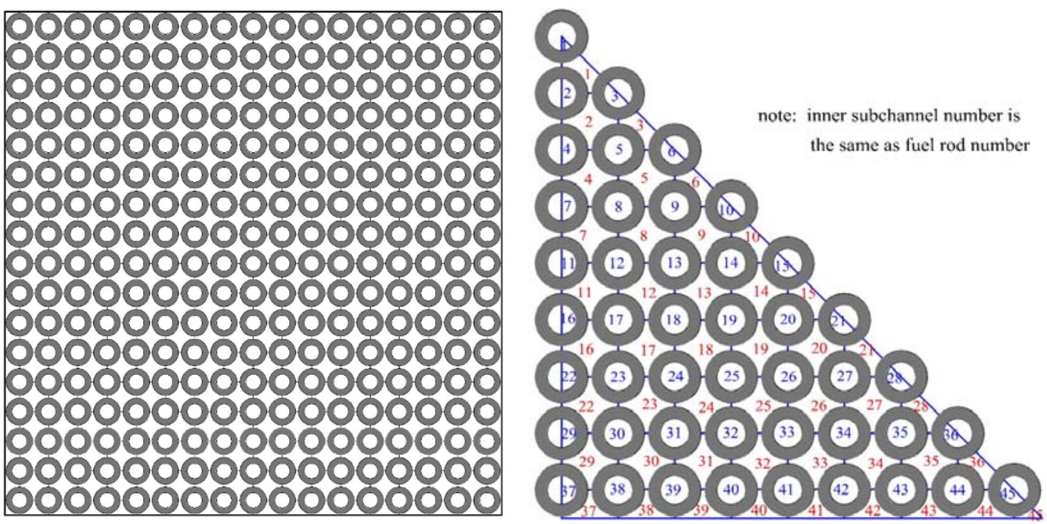

Figure 8 17×17 fuel assembly and subchannel identification in a 1/8 assembly

The assembly exhibits a symmetric cosine axial-power profile and a uniform radial power profile. According to the symmetry of square fuel assembly, we just select one-eighth of assembly to improve computational efficiency. The one-eighth of assembly including the rod and subchannel identifications is shown in Figure 8. As long as the one-eighth of fuel assembly is calculated, we can get thermal-hydraulic characteristics of the entire fuel assembly.

3.5 Results and Discussion

Figure 9 and Figure 10 show the outlet coolant temperature of inner and outer subchannels. The outlet coolant temperature difference among inner subchannels is small. This is mainly attributed to the same coolant flow in the inner subchannels. In outer subchannels, the subchannels close to edge and corner have higher outlet coolant temperature than others.

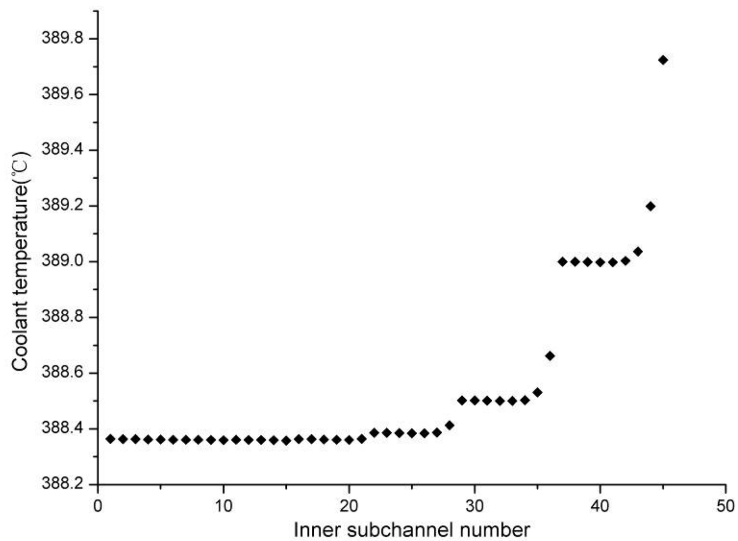

Figure 9 The outlet coolant temperature of inner subchannels

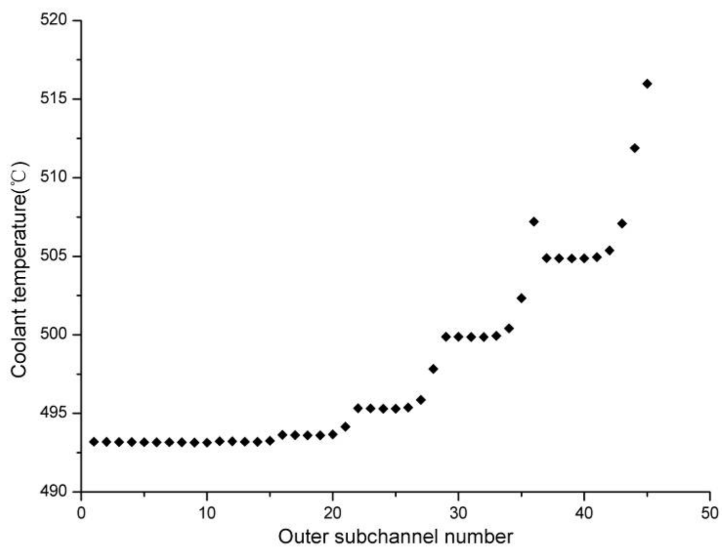

Figure 10 The outlet coolant temperature of outer subchannels

Figure 11 and Figure 12 show the coolant temperature distribution of typical inner and outer subchannels along the axial nodes. The axial coolant temperature distributions of inner subchannels are almost uniform. In the outer subchannels, the temperature difference between the cool and hot subchannels reaches 36℃. The outlet coolant temperature of outer subchannel isn't the highest temperature along the axial nodes, and the highest coolant temperature occurs in an axial node close to the outlet. This is because, at both ends of fuel assembly, the coolant temperature difference between inner and outer subchannels is large and the volumetric heat generation rate is relatively small, the outer coolant will transfer heat to the inner coolant.

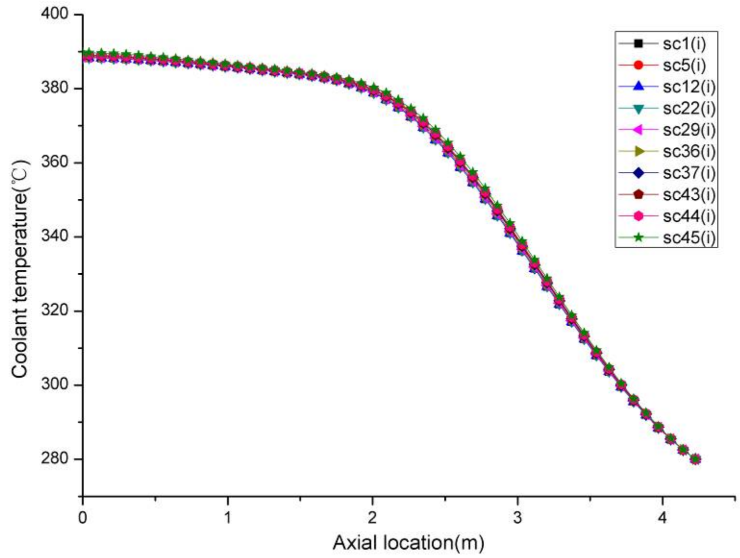

Figure 11 Axial coolant temperature distribution of inner subchannel

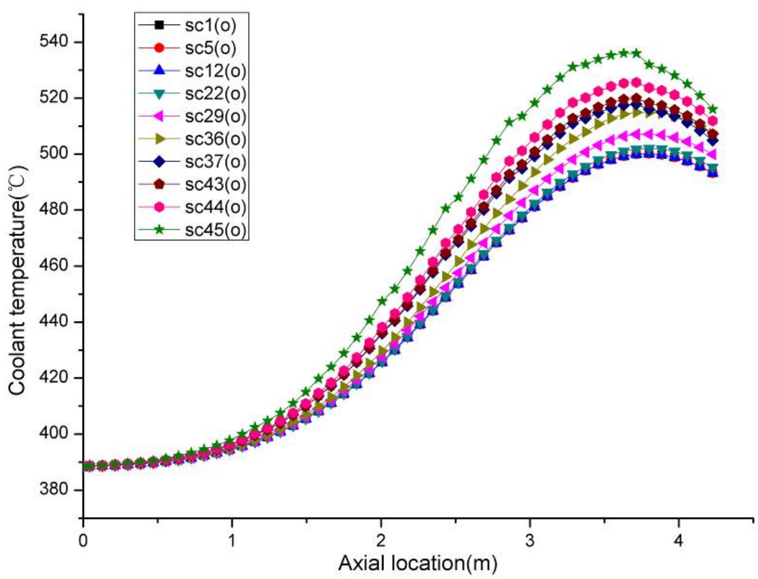

Figure 12 Axial coolant temperature distribution of outer subchannel

Figure 13 illustrates the cladding surface temperature distributions at typical outer subchannels along the assembly. The location *i*-*j* means the rod *i* facing the outer subchannel *j*, for example, 1-1 indicates the cladding rod 1 facing the outer subchannel 1. The cladding surface temperature increases generally along the assembly, reaching a maximum at locations close to the downstream end, and decreases afterward. This maximum temperature location corresponds mainly to the power and flow variations. The maximum predicted surface temperature occurs at rod 45 facing outer subchannel 45. It is 593.4 °C, which is below the design limiting cladding surface temperature 650℃. [11]

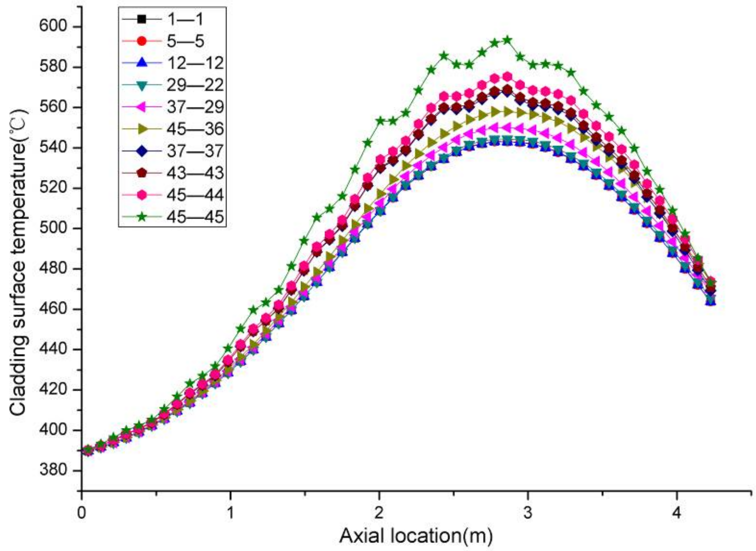

Figure 13 Axial cladding surface temperature of outer subchannel

Figure 14 shows radial fuel temperature distribution at the hotspot. The hotspot locates at rod 45 facing outer subchannel 45. The peak fuel temperature is 668.7℃, which is much lower than the UO_2 fuel melting temperature 2850℃. [12] The annular fuel rod has larger cooling surface than the traditional solid fuel rod, so it can rapidly transfer the heat generated by fuel pellet and greatly reduce the peak fuel pellet temperature. This is one of the advantages of annular fuel.

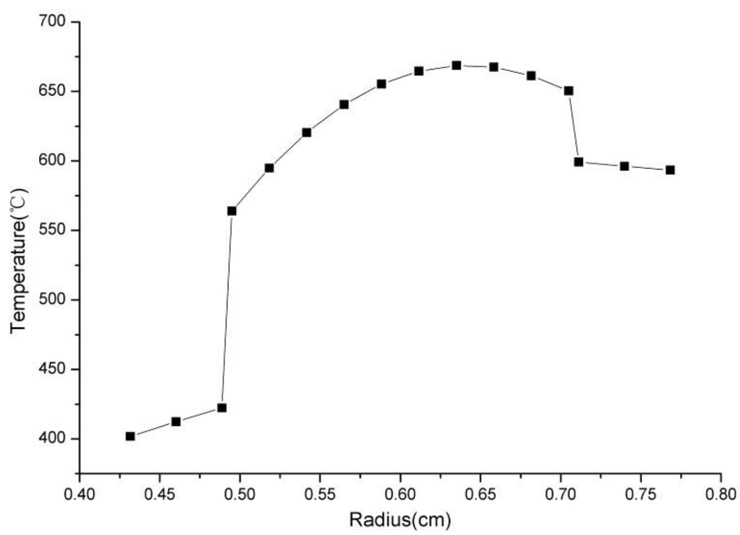

Figure 14 Radial fuel temperature distribution of the hotspot

4. Conclusion and prospect

The 2-D heat conduction model of annular fuel has been developed to evaluate the inner and outer heat fluxes and fuel temperature. After added this model, the ATHAS has the capability to make subchannel analysis for annular fuel assembly. With the in-out cooling scheme, subchannel analysis has been evaluated for a reference SCWR design loaded with annular fuel. The calculation result shows that the

MCST and the fuel pellet temperature were much lower than their design limiting values, especially the latter.

In this paper, we don't consider the inner and outer gap conductance behaviors, and assume their conductance are the same. Actually, the inner and outer gap conductance will change with pellet and cladding deformations, and greatly affect the inner and outer heat fluxes. So the gap conductance model needs to be developed for the exact stimulation of annular fuel in the next step. In addition, if we can adjust the inner and outer gap conductance to control the inner and outer heat fluxes by adding different insulating material to two gaps, we can control the coolant temperature of inner subchannels. If the inner coolant temperature is enough low, the core can get well moderation without extra water rods. However, this assumption needs to be studied further.

5. Reference

[1] Buongiorno, J., 2003. Generation IV R&D activities for the development of the SCWR in the U.S. *Progress report for the FY-03*, INEEL/EXT-03-01210, Idaho National Engineering and Environmental Laboratory.

[2] M. S. Kazimi *et al.*, High Performance Fuel Design for Next Generation PWRs: *Final Report*, MIT-NFC-PR-082, Center for Advanced Nuclear Energy Systems (CANES) (2006).

[3] H. T. Kim *et al.*, "Performance of internally and externally cooled annular fuel in loss of coolant accident'', *Proc. International Congress on Advanced Nuclear Power Plants*, Hollywood, Florida, USA, June 9–13, 2002 (2002).

[4] Edward Lahoda, Herbert Feinroth, Marcelo Salvatore, Diego O. Russo, HollyHamilton.High-Power-Density Annular Fuel: Manufacturing Viability. *Nuclear Technology*, 2007, 160(1): 100-111.

[5] Dandong Feng *et al*., "Thermal-Hydraulic Design of High-Power-Density Annular Fuel in PWRs", *Nuclear Technology*, 2007, 160(1): 16-44.

[6] Yong Sik YANG *et al*., "Evaluation of a Dual-Cooled Annular Fuel Heat Split and Temperature Distribution", *Journal of Nuclear Science and Technology*, Vol. 46, No.8, p. 836–845 (2009).

[7] Pioro, I.L., Khartabil, H.F., Duffey, R.B., 2004b. "Heat transfer to supercritical fluids flowing in channels-empirical correlations". *Nuclear Engineering and Design* 230, 69–91.

[8] Kyu Hyun Han, Soon Heung Chang, "Development of a thermal-hydraulic analysis code for annular fuel assemblies", *Nuclear Engineering and Design* 226 (2003) 267–275.

[9] McCracken, I.K., J.C. Kiteley and M.B. Carver. 1991. ATHAS modelling of pressure drop across Bruce bundle appendages (single-phase flow). *AECL Report*, ARD-TD-297, COG Report, COG-91-115.

[10] Jianqiang Shan *et al*., "SCWR subchannel code ATHAS development and CANDU-SCWR analysis", *Nuclear Engineering and Design* 239 (2009) 1979–1987.

[11] A. Yamaji, Y. Oka, and S. Koshizuka, "Three-dimensional Core Design of High Temperature Supercritical-Pressure Light Water Reactor with Neutronic and Thermal-Hydraulic Coupling", *Journal of Nuclear Science and Technology*, Vol. 42, No.8, (2005).

[12] J. Stuckert, A. Miassoedov *et al*. "ZrO_2 and UO_2 dissolution by molten zircalloy", *Nuclear Energy for New Europe 2002*, Kranjska Gora, Slovenia, September 9-12, 2002.