# RECENT IMPROVEMENT OF ATHAS CODE FOR SCWR

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## Abstract

ATHAS code has been successfully applied in SCWR subchannel analysis. The present paper gives the recent achievements of the code: (1) Subchannel analysis of CANDU-SCWR sliding pressure startup procedure: The subchannel analysis was performed for startup procedure, which covers the pressure range from 8MPa to 25MPa, flow pattern covers two phase flow and supercritical condition. The results show that proposed startup procedures can meet thermal-hydraulics criteria; the high sheath temperature should be carefully analyzed with PDO heat transfer model at pressure near critical point; (2) Subchannel analysis of CANDU-SCWR reheat bundle: The inlet condition for reheat bundle is different from that for normal bundle, such as superheated steam, subcritical pressure condition. The results show that the bundle can meet thermal-hydraulics criteria, and the pressure drop is much higher than normal bundle, which should be carefully designed; (3) Subchannel analysis of bundle with wire wrap: Wire wrap model was developed and added in the ATHAS code.

## 1. Introduction

The supercritical water coolant remains in single phase at all operating conditions of the SCWR. Therefore, the traditional limiting criteria based on either the dryout or the burnout phenomenon is not applicable. In turn, maximum cladding surface temperature (MCST) and peak fuel centerline temperature have been adopted as design criterion for the SCWR. The highly heterogeneous SCWR reactor core in radial and axial directions limits the application of the traditional single-channel thermal-hydraulic analyses to predict accurately the maximum cladding surface temperature in support of the fuel and core designs. Preliminary analyses of the cladding temperature of bundles in a fuel channel have been performed using subchannel codes.

ATHAS is a subchannel code developed in Xi'an Jiaotong University. The code has been designed to be general enough to accommodate itself to calculate with different geometries and orientations. These include single subchannels of different shapes, and multiple subchannels of CANDU PHWR, PWR and BWR designs, in both vertical and horizontal orientations. The operating conditions can be either supercritical pressure or subcritical pressure condition.

The subchannel analysis model of ATHAS is described by Shan et al[1,2]. . The present paper describes the recent improvement of ATHAS code and its application in SCWR analysis.

## 2. Analysis of CANDU-SCWR sliding pressure startup

Similar to the FPPs, there exist two startup strategies for SCWRs[3] . One is constant pressure startup, and the other one is sliding pressure startup (as shown in Fig. 1). During constant pressure startup procedure, the reactor starts at a supercritical pressure, and it requires a flash tank and pressure-reducing valves components, which can produce steam when the reactor power is low and the outlet temperature is lower than pseudo-critical point. During sliding pressure startup procedure, the reactor starts at a subcritical pressure and it requires a steam-water separator and a drain tank for two-phase flow. It must be assured that the maximum cladding surface temperature is below the criterion during startup, especially when CHF occurs at subcritical pressures. Compared with the constant pressure startup procedure, the advantage of the sliding pressure startup system is that complicated valve operations are not needed, and the pumping power consumption is decreased, which results in a higher efficiency during low-load operations. Therefore, the sliding pressure startup procedure is usually applied in SCWR system.

The main issue of sliding pressure startup is to limit the peak cladding temperature under the limiting criterion when CHF occurs. To achieve this simulation capability in subchannel code, the selection of heat transfer correlations should consider the wide range of application, because the startup procedure covers from 8MPa to 25MPa. Table 1 shows the heat transfer correlations in the analysis. CHF lookup table and PDO heat transfer lookup table are selected due to the following advantages over other prediction methods: (i) simple to use, (ii) no iteration required, (iii) wide range of application and (iv) high prediction accuracy. The effect of horizontal flow on CHF is also considered. To consider the effect of horizontal flow on CHF and PDO, a correction factor is selected.

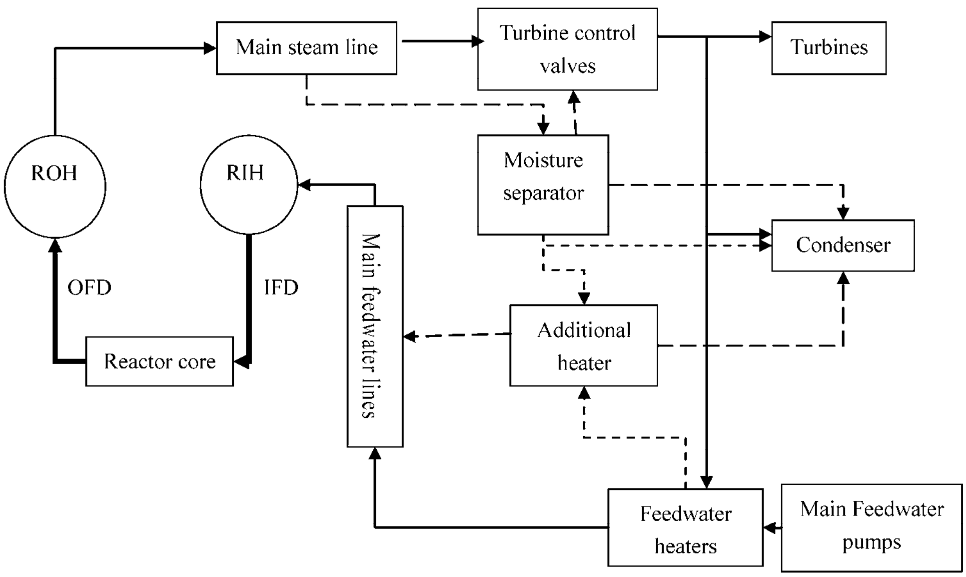



Fig. 1 Sliding pressure startup system for CANDU-SCWR

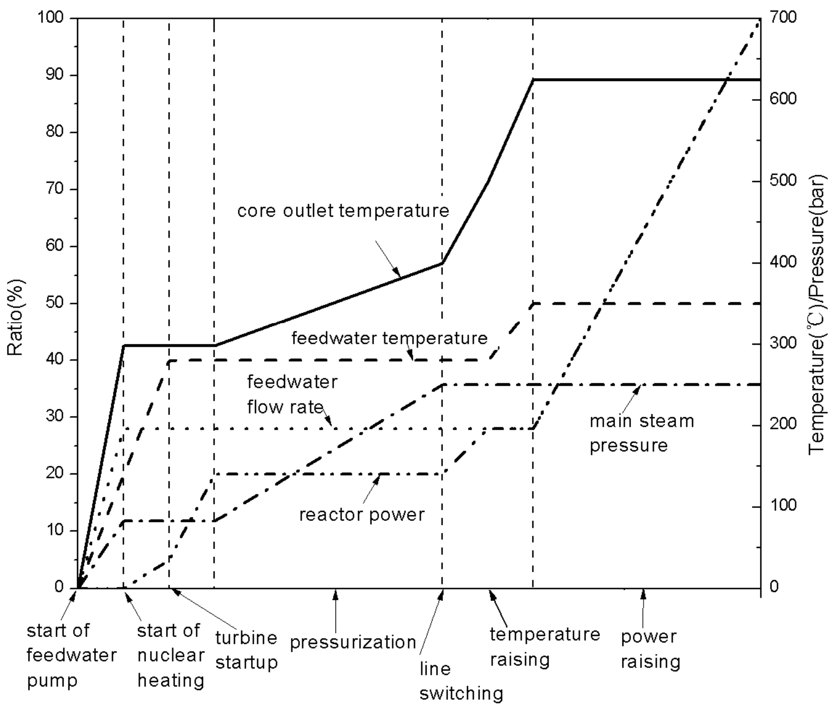
Table 1 List of heat transfer correlation

| Items | Correlations |
|---|---|
| Onset of nucleate boiling | $T_w > T_{sat}$ |
| Single-Phase Heat Transfer | Dittus-Boelter[4] |
| CHF | 2006 Lookup table [5] |
| Pre-CHF Two-Phase Heat Transfer | Chen [6] |
| Post CHF heat transfer | 2003 Look up table [7] |

| | |
|---|---|
| Supercritical pressure condition heat transfer | Swenson et al. [8] |

## 2.1 Startup System and Procedures

In the sliding pressure startup system, the reactor starts at a subcritical pressure. Because of the two-phase flow at subcritical pressures, a startup bypass system including a steam water separator, a drain tank and drain valves is needed, which is similar to the supercritical FPPs. The system for sliding pressure startup of CANDU-SCWR is shown in 错误！未找到引用源。. The sliding pressure startup curves are illustrated in Fig. 2, and they can be divided into five phases, which is similar to the design of SCLWR-H [3] : (1) Start of Reactor Core at Subcritical Pressure; (2) Start of Turbine; (3) Pressurization Phase; (4) Feedwater Temperature Increasing Phase; and (5) Core Power Increasing Phase. The detail analysis was listed by Shan etl al.[2]. The present paper just shows the result in Phase (3)-(5).



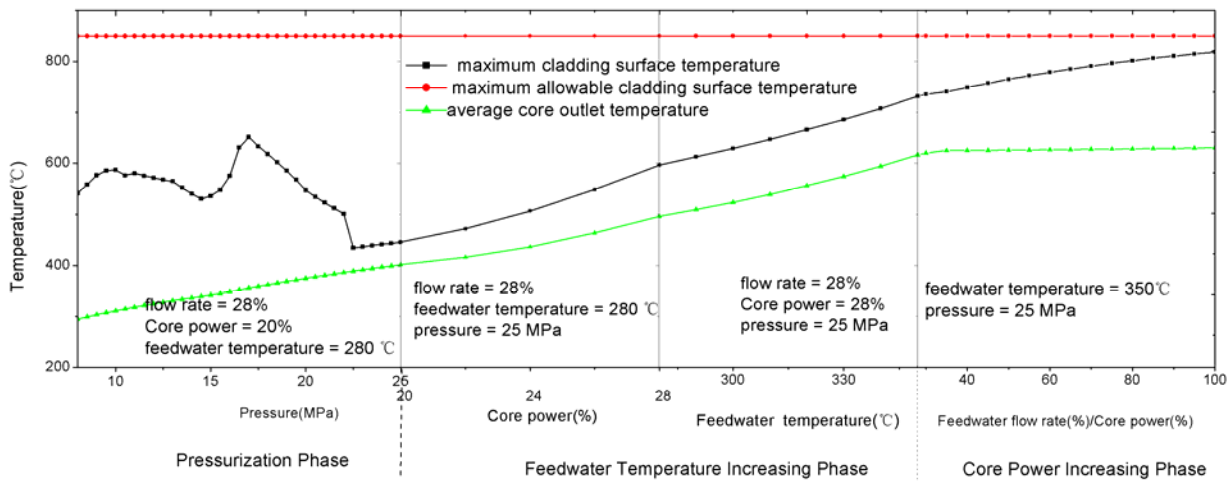



Fig. 2 Sliding pressure startup curve and peak cladding temperature

The core is pressurized from 8.3 MPa to 25 MPa after the turbine startup. The feedwater flow rate is kept at 28%, and the core power is kept at about 20%. The main steam temperature is equal to the saturation temperature at subcritical pressure, since the outlet coolant is two-phase flow, and its temperature is increased to 401 °C when the reactor pressure rises up to 25 MPa.

The dryout will occur during the whole pressurization phase. This is because the CHF in a horizontal channel is much lower than that in a vertical one. So care must be taken to make sure that the maximum cladding surface temperature will not pass the rated value throughout the pressurization, especially at subcritical pressures when dryout occurs. The maximum cladding surface temperature (653 °C) is well below the criterion of 850 °C. The plant is shifted from the startup bypass mode to the once-through normal operation mode after pressure is raised to 25 MPa.

After line switching, the core power is raised from 20% to 28%, then the inlet temperature is increased to 350 °C, while the core pressure is kept constant at 25 MPa. The core outlet temperature of coolant is finally increased to 625 °C. The maximum cladding surface temperature is 732 °C.

The core power and the feedwater flow rate are increased from about 28% to 100% after the main steam temperature increases to 625 °C. The core inlet and outlet temperatures are kept constant accordingly. The maximum cladding surface temperature is 818.7 °C.

## 3. Analysis of wire wrapped SCWR assembly

Application of wire wrap spacers in SCWR can reduce pressure drop and obtain better mixing capability. As a consequence, the required coolant pumping power is decreased and the coolant temperature profile inside the fuel bundle is flattened which will obviously decrease the peak cladding temperature. The distributed resistance model for wire wrap was developed and implemented in ATHAS subchannel analysis code. The HPLWR wire wrapped assembly (as shown in Fig. 3) was analysed.

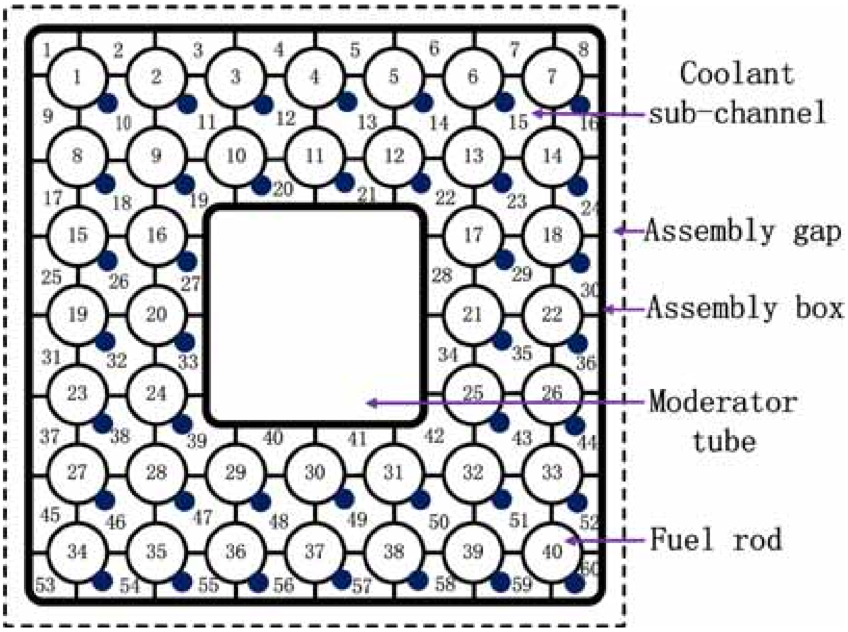



Fig. 3 Fuel assembly design and subchannel identification

Table 2 gives the operating parameters. The relative axial power distribution is from the result of neutronic/ thermalhydraulics coupling analysis[9]. The radial power distribution is assumed as uniform.

Table 2 Reactor operation parameters and correlations options in Current Analysis

| Parameters | Values |
|---|---|
| Coolant inlet pressure | 25 MPa |
| Coolant inlet temperature | 280°C |
| Coolant exit temperature | 500°C |
| Average heat flux | 0.6142 MW/m$^2$ |
| Mass Flux | 780 kg/(m$^2$·s) |
| Heat transfer coefficient | Jackson Correlation |
| Friction coefficient | Blasius Correlation |
| Turbulent Mixing | Rowe and Angle correlation |

### 3.1 Subchannel analysis results with wire wrapped and grid spaced assemblies

To compare with the behavior of wire wrap in the SCWR assembly, a subchannel analysis of assembly with grid space is also taken. The parameters of grid spacers are given by Cheng et al[10] . There are 15 grid spacers along the axial lengths.

The comparison of axial cladding temperature profiles in the hot channel between wire wrapped and grid spaced assemblies is shown in Fig. 4. The peak cladding temperature is 587.6°C for wire wrapped assembly and 603.5°C for grid spaced assembly respectively. The location of peak cladding temperature is at subchannel 43, rod 25 and 3.96m at axial position. The reason for the temperature difference is that more forced cross flow between subchannels in wire wrapped assembly, resulting in a more uniform coolant temperature profile.  Fig. 5 shows the coolant temperature profile in the subchannels at the axial position where the peak cladding temperature occurs. The difference of coolant temperature between the hottest and coldest channels is 25.3°C for the wire wrap and 40.2°C for the grid spacer respectively. Fig. 6 shows the pressure drop in the two kinds of assembly, it can be observed that the pressure loss in the wire wrapped assembly is less than that in the grid spaced assembly, which consequently can reduce the required pump power.

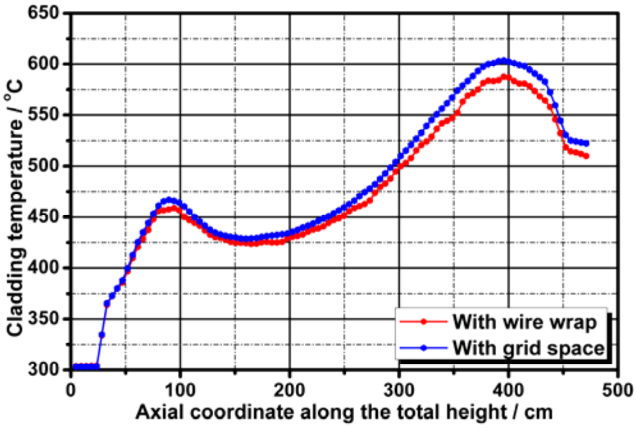



Fig. 4 Axial cladding temperature profile in the hot channel

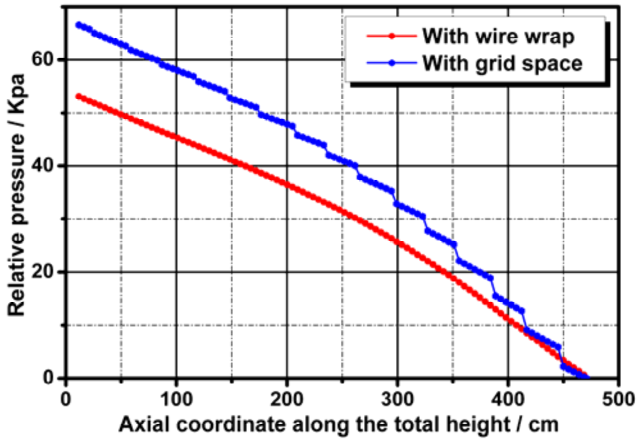



Fig. 6 Pressure drop of two kinds of assembly

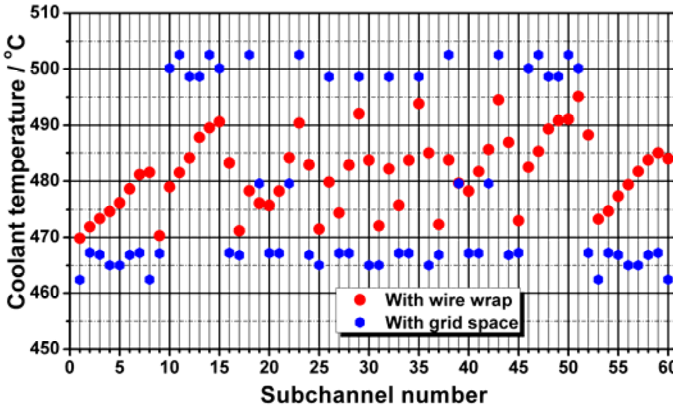



Fig. 5 Coolant temperature profile at axial position where the peak cladding temperature occurs

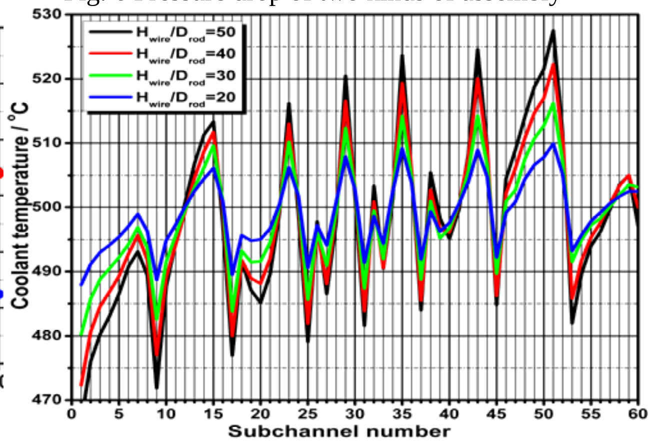



Fig. 7 Exit cooling temperature profile with different $H_{wire}/D_{rod}$ values

### 3.2 Sensitivity analysis of wire wrap pitch

Sensitivity analysis of wire wrap pitch was also studied. The pitch is selected according to the criteria determined by Diller (2004), $H_{wire}/D_{rod}<50$. So four $H_{wire}/D_{rod}$ values have been selected for analysis, which are $H_{wire}/D_{rod}$ =20, 30, 40 and 50 respectively.

Fig. 7 shows the comparison of exit cooling temperature profiles with different $H_{wire}/D_{rod}$ values. It is found that lower $H_{wire}/D_{rod}$ will result in a more uniform profile, since the temperature difference between the hottest and coldest channels is 60.9 °C for $H_{wire}/D_{rod}$ =50 and 21.9 °C for $H_{wire}/D_{rod}$ =20. This is understandable because lower $H_{wire}/D_{rod}$ causes more cross flow. However, lower $H_{wire}/D_{rod}$ also leads to a higher pressure drop, as shown in Fig. 8. The pressure drop is increased from 51.8 kPa for $H_{wire}/D_{rod}$ =50 to 56.1 kPa for $H_{wire}/D_{rod}$ =20.

Fig. 9 shows the peak cladding temperature comparison with different $H_{wire}/D_{rod}$ values. As we have discussed above, the peak cladding temperature mainly depends on whether the coolant temperature profile in subchannels is uniform or not. Higher $H_{wire}/D_{rod}$ will result in a higher cladding temperature. As for $H_{wire}/D_{rod}$ =50, the calculated peak cladding temperature is 603.3 °C, nearly the same as grid spaced assembly.

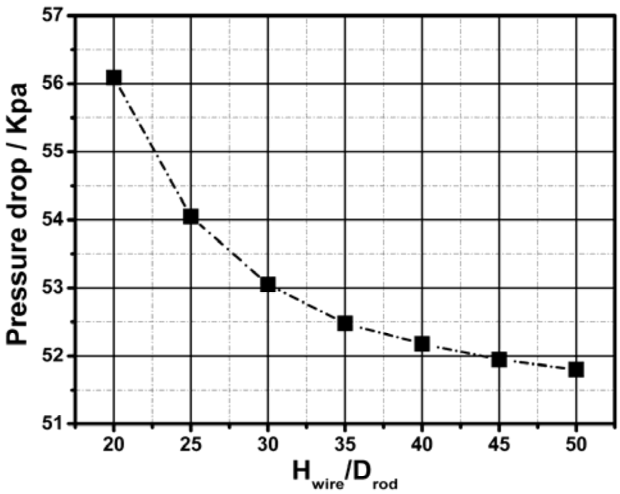



Fig. 8 Pressure drop value with different $H_{wire}/D_{rod}$

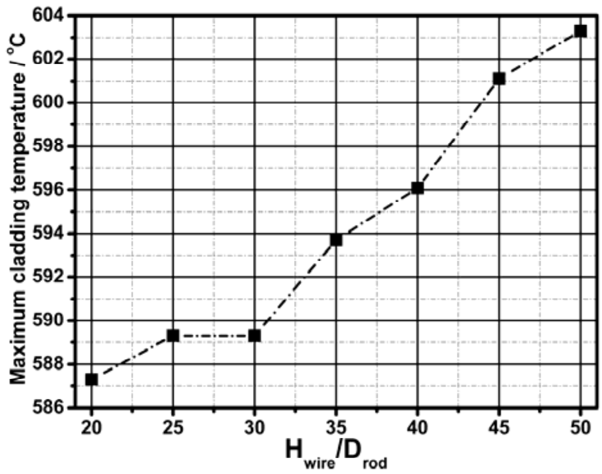



Fig. 9 The peak cladding temperature comparison with different $H_{wire}/D_{rod}$ values

## 4. Analysis of CANDU-SCWR bundle in reheated core

To achieve higher thermal efficiency a nuclear steam reheat has to be introduced inside a reactor. Currently, all supercritical turbines at thermal power plants have a steam-reheat option. In the 1960 and 1970s, Russia, USA and some other countries have developed and implemented the nuclear steam reheat at subcritical-pressure in experimental reactors. There are some papers, mainly published in the open Russian literature, devoted to this important experience. Pressure-tube or pressure-channel SCW nuclear-reactor concepts are being developed in Canada and Russia for some time. It is obvious that implementation of the nuclear steam reheat at subcritical pressures in pressure-tube reactors is easier task than that in pressure-vessel reactors.

The simplest single-reheat cycle option of SCW NPP was developed by Naidin et al.[11],. Based on this simple cycle two general schemes of SCW NPP with the single-steam reheat and heat regeneration were developed. In this analysis, the cycle developed by Naidin et al was selected. The CANFLEX bundle is selected as CANDU-SCWR bundle.

Table 3 shows the parameters in the analysis. Fig. 10 shows the pressure drop along the bundle against the mass flux, and it shows that the pressure drop is very high if the mass flux is selected as 2700 kg/m2s, so the mass flux should be reduced to decrease the pressure drop. If the mass flux is reduced to 1600kg/m2s, the pressure drop can be decreased to 1MPa. Fig. 11 shows the peak cladding temperature against mass flux, it shows that the peak cladding temperature increases while mass flux increasing, this is reasonable because increasing of mass flux will result in the power increase to meet fixed outlet coolant temperature. The peak cladding temperature is well below the limiting criterion.

Table 3 the parameters in the reheated channel analysis

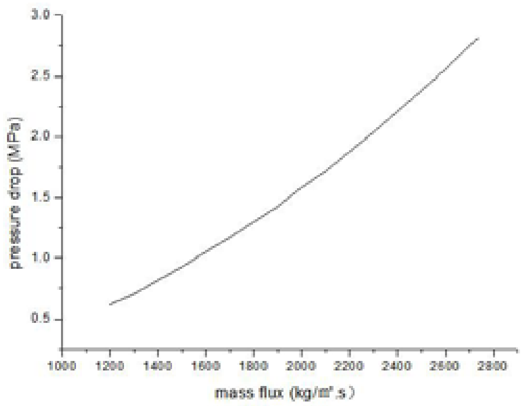



Fig. 10 the pressure drop along the bundle against the mass flux

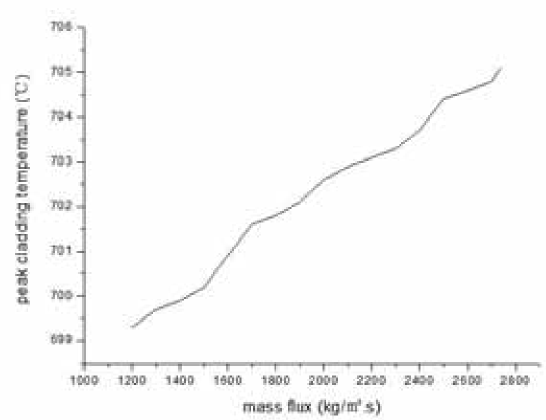



Fig. 11 the peak cladding temperature against mass flux

## 5. Conclusion

ATHAS code has been successfully extended in three fields: (1) Subchannel analysis of CANDU-SCWR sliding pressure startup procedure: The subchannel analysis was performed for startup procedure, which covers the pressure range from 8MPa to 25MPa, flow pattern covers two phase flow and supercritical condition. The results show that proposed startup procedures can meet thermal-hydraulics criteria; the high sheath temperature should be carefully analyzed with PDO heat transfer model at pressure near critical point; (2) Subchannel analysis of CANDU-SCWR reheat bundle: The inlet condition for reheat bundle is different from that for normal bundle, such as superheated steam, subcritical pressure condition. The results show that the bundle can meet thermal-hydraulics criteria, and the pressure drop is much higher than normal bundle, which should be carefully designed; (3) Subchannel analysis of bundle with wire wrap: The results show that wire wrap can result in more uniform temperature profile and lower pressure drop than those in grid spacer Assembly.

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