THERMAL HYDRAULIC CALCULATION OF THE CHANNELS OF A NUCLEAR REACTOR (SCWR)

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It's necessary to study the method of thermo-hydraulic calculation of the core nuclear reactor (a sequence of calculation formulas that made it possible to determine the parameters of the coolant and the surface of the fuel elements along the length of the channel: enthalpy of water, water temperature, the temperature of the cladding of a fuel rod, fuel center temperature, hydraulic resistances). The initial calculation parameters and schematic of circulation circuit is on the basis of PWR. Heat transfer between water and fuel elements in the course of supercritical water flow. Of greatest interest are fast reactors cooled by liquid sodium or lead, and water-cooled reactors at supercritical pressure of 25 MPa (SCWR), which allow combining the design of a reactor with pressurized water (PWR) and a boiling reactor (BWR) in a single concept and increase the efficiency (up to 44% and more).

The flow is super critical, therefore Mokrv et al, correlation is selected for heat transfer calculation.

$$Nu_b = 0.0061 Re_b^{0.904} \overline{Pr}_b^{0.684} \left(\frac{\rho_w}{\rho_b}\right)^{0.564}$$

Fluid bulk properties are known, outside wall temperature of the cladding is assumed and using the Mokrv et al, correlation, fluid properties at wall temperature and bulk temperature, heat transfer coefficient is evaluated. Which is then used to calculate new outside wall temperature of cladding. Percentage difference between the assumed wall temperature and new wall temperature is evaluated. The whole iteration is then repeated with new wall temperature as the assumed wall temperature till the difference between assumed wall temperature and new wall temperature becomes less than 2 percent.

The maximum cladding outside temperature is 388.5 ^{o}C at 0.8 meters above the center of fuel rod. The temperature of cladding is higher than the coolant temperature at all points, it reaches a maximum values and then decrease towards the outlet, this trend is in harmony with that of literature [1]. The maximum temperature value of fuel centerline is 1200 ^{o}C at the center of fuel rod.

The flow is super critical, therefore Mokrv et al, correlation is selected for heat transfer calculation, same as in above section. Fluid bulk properties are known, outside wall temperature of the cladding is assumed and using the Mokrv et al, correlation, fluid properties at wall temperature and bulk temperature, heat transfer coefficient is evaluated. Which is then used to calculate new outside wall temperature of cladding. Percentage difference between the assumed wall temperature and new wall temperature is evaluated. The whole iteration is then repeated with new wall temperature as the assumed wall temperature till the difference between assumed wall temperature and new wall temperature becomes less than 2 percent. The maximum cladding outside temperature is 610 ^{o}C at 0.8 meters above the center of fuel rod.

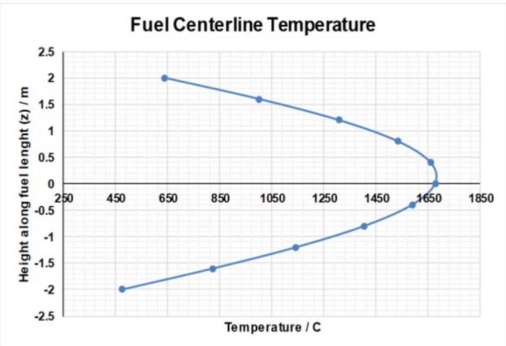

Fig. 1. Distribution of fuel centerline temperature for center

Fuel centerline temperature was evaluated along the length of fuel rod and plotted in figure 1. The maximum temperature value of fuel centerline is 1650 ^{o}C at the center of fuel rod.

The thermal hydraulics calculations for VVER SKD showed that the maximum fuel centerline temperature is 1650 ^{0}C for ascensional section of the core, whereas, for descending section, it is 1200 ^{0}C. These values are very less than the melting temperature of the UO_2 i.e. 2865 ^{0}C, ensuring high safe margin in this reactor. Moreover, the cladding temperature. Moreover, the maximum fuel centerline temperature for VVER SKD is lower as compared with PWR plant of similar power.

REFERENCES

1. A. Glebov and A. Klushin, "Development of Supercritical-Water Cooled Reactors in Russia and Abroad", *Atomic Energy*, vol. 116, no. 5, pp. 320-329, 2014. Available: 10.1007/s10512-014-9860-x [Accessed 2 July 2020].