Critical Power Ratio with New Pressure Tube Creep Data of a CANDU-6

J. H. Park[a], J. Y Jung[a] and G. G. Lee[a]
[a]*Korea Atomic Energy Research Institute, 989-111 Daedukdaero, Yuseong-gu, Taejon, 34057, Korea*
jhpark@kaeri.re.kr

1. Introduction

A CANDU-6 reactor core has 380 fuel channels and each fuel channel is loaded with 12 fuel bundles into a pressure tube that has an independent pressure boundary. It is known that the pressure tube is the most important component of a CANDU reactor and can be degraded by creep phenomena during long term reactor operation. Creep of a pressure tube may be as high as 5.1% at the end of the tube's life time. When the reactor becomes older, the top open gap between a horizontal pressure tube and an inside fuel bundle becomes wider because it is expanding radially as well as axially during its life time, as a result of the creep of the pressure tube. This allows a by-pass flow on the top section inside the pressure tube. Hence, the crept pressure tube deteriorates the CHF (Critical Heat Flux) of the fuel channel and finally reduces the reactor operating performance and derating of the reactor power to keep the operational margin. In order to assess thermal-hydraulic characteristics of a fuel channel, the creep rate of a pressure tube should be known.

In particular, to evaluate the CPR (Critical Power Ratio) of a CANDU reactor, the pressure tube creep data generated by the RC1980 model have been used worldwide until recently [1] because the creep mechanism of the pressure tube during the reactor operating is complex and other creep models of the pressure tube have not been well developed and verified with measurement data. Recently, KAERI has developed a new creep model (hereafter referred to as the Lee model) [2] and generated the pressure tube creep data of Wolsong Units according to the reactor burnup.

This paper introduces the assessment of the effect of the pressure tube creep on the CPR using new pressure tube creep data from W-4 (Wolsong-4). The new pressure tube creep data from W-4 were generated by the Lee model or a new program that is being developed based on Bayesian Statistics and MCMC (Markov Chain Monte Carlo) methodology at KAERI [2]. NUCIRC code [3] was used to evaluate the CPR characteristics of W-4. The results were compared with previous results.

2. NUCIRC modelling for CCP evaluation

The PHTS of CANDU-6 reactor can be analyzed by a system thermal-hydraulics code, NUCIRC [3]. NUCIRC code is a one dimensional and lumped parameter code system. It mainly has two different functions, channel wise calculation and circuit analysis among the 9 ITYPE options. The present study selected the ITYPE 2 option for channel wise calculation to evaluate the critical channel power using header-to-header boundary conditions. The ITYPE 2 option is used to evaluate the conditions under which the critical heat flux of the channel occurs at a given pressure boundary condition. Iteration under the header to header pressure boundary conditions can be continued until the critical heat flux occurs in the fuel channel. When the critical heat flux is reached at any axial location of the fuel channel as the channel power is increased, it is noted that the fuel channel becomes burnout or damaged and the channel power at CHF occurrence is called the critical channel power (CCP). The critical channel power ratio, CPR, which is defined as the maximum channel power divided by the given channel power, should always be kept higher than 1 so that the fuel surface will not be damaged. This condition is the most important criterion for evaluating the reactor safety. Also, when the fuel channel power increases, the enthalpy of the cooling water flowing into the fuel channel increases in the single-phase liquid flow. The abnormal flow region, such as a two phase flow, consequently becomes wider and eventually a critical heat flux state occurs in which no liquid exists on the surface of the fuel rod. In the case of the same header to header pressure boundary conditions, the flow rate of the cooling water decreases gradually as the cooling water enthalpy increases to satisfy the pressure drop of the header to header boundary condition. In general, when the critical heat flux occurs, the flow rates in the channel of a CANDU-6 reactor are distributed between about 9 kg/s and 22 kg/s.

Among the input data of NUCIRC code, the past pressure tube creep data of the RC1980 model were replaced with the pressure tube creep data generated by the Lee model. The reactor burnup of 3650 FPD, which is equivalent to the middle burnup of a reactor lifetime was selected.

3. New pressure tube creep data

Fig. 1 shows the flow direction of the cooling water of the 380 pressure tubes connected to the loop by color and the cooling water is designed to flow in the opposite direction from one channel and the adjacent channel such that the positions of the inlet and the outlet of the fuel channel are opposite to each other direction. The point where the cooling water of low temperature is supplied is the inlet portion and the portion where the cooling water of high temperature is discharged is the outlet. During the statistical processing of the pressure tube measurement data, the first string of fuel bundles in the fuel channel is always located on the exit side of the twelfth bundle on the inlet side.

In the case of W-4, all 380 nuclear fuel channels are installed in a BEI (Backend Inlet) system. However, the RC1980 model is calculated based on BEO (Backend Outlet). The Lee model predicts BEI or BEO differently providing more precise prediction than the RC1980 model, which simulates the diameter of a pressure tube without considering BEI and BEO. In the case of the RC1980 model, the maximum diameter of the diameter measured in the axial direction of the pressure tube is well predicted. Similarly, the value predicted by the Lee model agrees well with the measured maximum value. The predicted value of the pressure tube diameter can vary due to the differences in characteristics of BEO/BEI.

Fig. 2 shows the typical axial power and pressure tube creep profiles for the O-08 fuel channel which is the highest quality channel. The axial pressure tube creep profile is a downward cosine type and the peak creep rate was located at the 9th or 10th fuel bundles as shown in Fig. 2.

On the other hand, twelve fuel bundles are loaded horizontally inside the pressure tube, and the distribution of neutrons is different for each position of the reactor core. As a result, the amount of pressure tube creep varies depending on the axial position. Fig. 2 shows the change of the creep rate of the pressure tube according to the axial position. The creep rate is most affected by neutron irradiation, so the creep rate of the pressure tube irradiated by high neutron flux is high. As shown in Fig. 2, the creep is low due to the low bundle power at the inlet, and as the bundle power increases, the increase in the pressure tube creep rate becomes larger as the bundle becomes closer to the downstream, and the increase in the creep rate of the pressure tube has a parabolic shape that is generally tilted downward due to the flow effect.

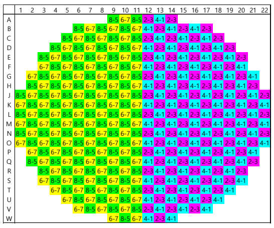

Fig. 1 Core fuel channel distribution of loop path of reactor core

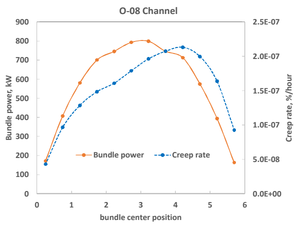

Fig. 2 Typical axial distribution of PT creep rates and bundle power

Next, the axial pressure tube diameter predictions of the fuel channel by the RC1980 and Lee models were compared and plotted in Figs. 3 and 4. In the case of 3650 FPD at the middle of the reactor operation period, the axial distributions of the pressure tube creep ratios (Lee model/RC1980 model) are shown for each fuel channel according to the fuel channel powers.

As can be seen from these figures, the pressure tube creep rates by the Lee model at the inlet and exit regions of the fuel channel is higher than the RC1980 model data and the ratios of the creep rates at the axial central region are close to 1. In the case of the high power channels (see Fig. 4), the pressure tube creep rates given by the Lee model are much higher than the RC1980 model data. The reason for this is that all 380 nuclear fuel channels of W-4 are installed as BEI, whereas the RC1980 model is calculated based on BEO. However, the Lee model predicts BEI or BEO differently, and its prediction may be deviated from the RC1980 data.

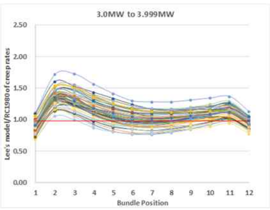

Fig. 3 PT creep ratio (Lee model/ RC1980) between 3.0 MW and 3.99 MW of channel power at 3650FPD

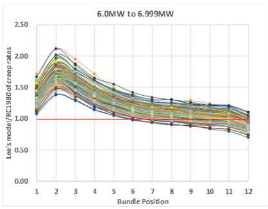

Fig. 4 PT creep ratio (Lee model/ RC1980) above 6.0 MW of channel power at 3650FPD

4. Results and Discussion

Using new pressure tube creep data generated by the Lee model, the CPR was calculated for W-4 by the NUCIRC code to examine the feasibility to apply new pressure tube creep data to the Wolsong CANDU units. The reactor burnup considered in the present study was 3650 FPD, which is approximately equivalent to the burnup at the middle of the lifetime of the reactor operation period. The 380 fuel channel power profiles are shown in Figs. 5 and 6. The maximum and minimum channel powers were found to be 6945 kW of O18 channel and 2344 kW for V17 channel, respectively, as shown in Fig. 5.

Using these data and boundary conditions, the critical power ratios for the 380 fuel channels of W-4 were calculated and the results are summarized in Fig. 7. As shown in Fig. 7, the CPRs were distributed in a lower in the inner region and a higher range in the center and outer regions as expected. The minimum CPR was 1.253 for the M19 channel.

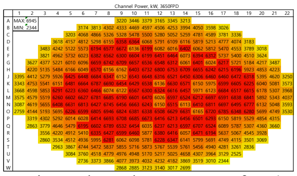

Fig. 5 Channel power map of W-4

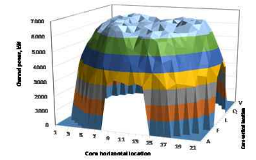

Fig. 6 3D Channel power of W-4

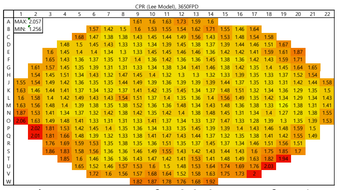

Fig. 7 CPR map for 3650 FPD of W-4

The CPRs predicted by the Lee model were compared with those given by the RC1980 model and the ratios of Lee data vs. RC1980 data for the 380 fuel channels are plotted in Fig. 8. The minimum and maximum ratios were 0.983 and 1.012. This means that the maximum and minimum differences of the two models for the CPRs of the all channels were within ±2%.

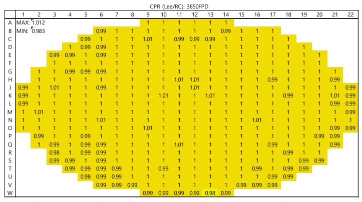

Fig. 8 CPR ratio (Lee data/RC1980 data) map for 3650 FPD of W-4

5. Conclusion

From the present study, new creep data of the pressure tube diameter generated by the Lee model was applied to W-4 in order to investigate whether the new model is feasible to evaluate the CPR of the Wolsong CANDU reactors.

It was found that the CPR predicted by the Lee model is very close to the CPR predicted by the past model referred to as RC1980 within ±2% differences. It is noted that the new model of pressure tube creep can successfully substitute for the RC1980 model. Further study will be necessary for the higher creep rates of the pressure tubes at the end of reactor lifetime.

Acknowledgement

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