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DEVELOPMENT OF A CANDU FUEL CHANNEL MODEL TO ASSESS THE EFFECT OF A PRESSURE TUBE CREEP ON THE SAFETY RELATED PARAMETERS

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Abstract

Recently the effect of pressure tube creep on the reactor safety in CANDUs emerges as an important issue of safety analysis due to a need for an extended operation. The accident analysis for the aged plants needs to incorporate major degradations of the plant performance in the safety analysis. In this paper, a CATHENA fuel channel model for studying the effects of the vertical offset of the fuel bundles in a crept pressure tube on the fuel and pressure tube cooling is developed. The current practice of the CANDU safety analysis assumes that the fuel bundles stay in a manner concentric to the pressure tube centerline even in the crept pressure tubes, whereas in reality the bundles sit at the bottom of the pressure tube. With this point in mind, 37-pin models with and without vertical offset of the bundle in the crept fuel channel are developed and tested for Reactor Outlet Header (ROH) 100% break LOCA accident, and results compared. As a result, it was found that the difference between the uncrept fuel channel model and the two crept fuel channel models, a concentric one and another vertically offset one, is quite significant, whereas the difference between the two crept fuel channel models is insignificant. Therefore it is concluded that the use of the concentric crept fuel channel model for the aged CANDU-6 safety analysis is justifiable for the first 200 sec into an accident.

Introduction

A two-phase heat transfer phenomena in the fuel bundle strings located in a horizontal pressure tube with an axial variation of the radial creep, especially under a low stratified two-phase flow condition such as that encountered in a CANDU reactor under the later stage of a blowdown phase of a LBLOCA, say from 40s to 200s, involves a complex heat transfer nature. This includes a conduction in the fuel rods and pressure tube, convection in the vapor and liquid regions, and radiation among the fuel rods and those portions of the pressure tube exposed in the steam, pressure tube, and calandria tube. As these three modes of heat transfer has to be treated in a combined way, modeling the heat transfer phenomena inside the fuel bundle under the stratified flow occurring during the later stage of LBLOCA blowdown has been one of the most challenging tasks in the CANDU safety analyses. The main reason for this attention to the channel aging is that it is closely related to the integrity of the pressure tube. In this study a fuel channel model for handling this aged situation was developed and implemented, and is under preliminary testing of the analysis results. The analysis results up to now are encouraging and the validation of the developed model is ongoing. The major motivation of this study is to assess the conservatism of the current CANDU safety analysis methodology for a fuel channel with an axial variation of the radial creep of the pressure tube as usually experienced in an aged CANDU plant, as the current practice of a CANDU safety analysis assumes that the fuel bundles stay in a manner concentric to the pressure tube centerline even in crept pressure tubes.

1. Major Improvement of the Model

The backbone of the fuel channel heat transfer modelling of CATHENA[1] for a stratified flow is (a) to find the height of the water level using a void fraction, (b) to find the convective heat transfer coefficient on the surfaces of the heat structures depending on the fraction exposed to steam, and (c) to calculate the view factors among the segmented parts of the heat structures and use them to find the radiation heat transfer coefficient. Combining all these three modes of heat transfer for the same heat transfer surfaces without a priori knowledge of the flow regime and heat transfer topology requires a careful and detailed segmentation of the heat structures. To carry out these tasks, a computer program for computing the relative location of each segment of the heat structures such as the fuel rod and pressure tube with respect to the centerline of the crept pressure tube and the water level is developed. A way to compute the fraction of the submerged part of each segment of the heat structures is then developed, and validated against benchmark results whose accuracy are already proved. Based on these results, the CATHENA crept fuel channel model input is prepared, i.e., the interlinked network among all the segments of the involved heat structures are set up, and the necessary inputs for the combined heat transfer calculation are prepared. Once these preliminary works are completed, the heat transfer model calculates the heat fluxes for different modes on the same surface in such a way that heat balance is kept. The thermalhydraulic boundary conditions imposed are the channel inlet and exit pressures, the enthalpy and void fraction of each phase of the inlet flows, and the cooling condition of the calandria tube outside. The convective heat exchange by the annulus gas flow through the annulus gap between the pressure and calandria tubes is assumed to be negligible. As for the modeling of the fuel bundle strings, developed is a 37-pin model where the 37 rods are treated individually. Axially, the whole fuel channel of about 6 m is divided into 12 segments both hydraulically and heat structure-wise. The configuration of the 28 element fuel bundle, pressure tube, and calandria tube are schematically depicted in Figure 1 to show what condition is considered in this study as an illustration only.

b)

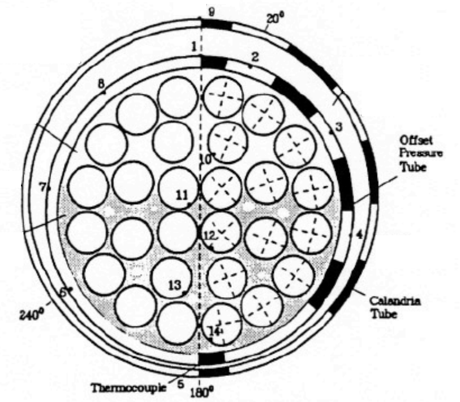


Figure 1 CATHENA Solid Structure Model and Radiation Model for Standard 28-Pin Fuel.

1.1 Surface-to-Surface Radiation Model

As the heat transfer due to conduction and convection in a two phase flow is cumbersome, only that for the radiation heat transfer will be described here. The computation method for radiation heat transfer is a lumped-system approximation for gray diffuse surfaces contained in an enclosure. The assumptions of this method are as follows:

- The fluid in the enclosure neither emits nor absorbs radiant thermal energy.
- Reflectance from a surface is not a function of incident, reflected direction, nor radiation frequency.
- Temperature, reflectance, and radiosity are constant over each surface.

The view factors from each surface to all other surfaces must add up to one. Also, to conserve energy, the sum of the area of surface i times the view factor for each surface i to any other surface j over all j must equal the sum of the area of surface j times the view factor from surface j to surface i. The view factor was calculated using a MATRIX code[2], which was validated and is widely used for CATHENA applications.

2. CATHENA Simulation and Discussion

2.1 37-pin model of 37-rod bundle

2.1.1 CATHENA Simulation for Steady State of Stratified Flow Condition

A simple test problem is formulated and tested. Initially the W-10 channel of 4.0 MWth was simulated at a full power steady state condition. And In order to make the final steady state condition as a stratified two phase flow condition, the inlet pressure was then decreased very close to the exit header pressure of 10.0 MPa and 10.22MPa, and the channel power was also decreased to 60% of the initial full power from t= 1500sec, and kept the same until the final steady condition was reached.

The profile of the radial creep of the pressure tube was assumed to have a center-peaked cosine shape, and the typical axial and radial power shape of a relatively low powered, W10 channel of CANDU-6 was applied. As a result, a steady state two phase condition was formed with the quality gradually rising along the axial direction, and a partially stratified flow formed at the last three fuel bundle locations. The radiation heat transfer model was utilized to simulate the two-phase heat transfer among the fuel rods and pressure tube at this partially stratified flow condition. The view factors among these heat structures were calculated separately using the MATRIX code developed for the CATHENA code. A typical cross sectional view of the fuel channel under consideration for this study is shown in Figure 2, and the steady state result at the 12th axial bundle location from the inlet out of the 12 bundles in a channel is depicted in Figure 3.

Incorporation of a full-fledged 37 element CANDU standard fuel with 2 segments for the complex heat transfer under a blowdown phase two-phase flow including a stratified flow condition seems satisfactory as a fuel temperature distribution with a given void fraction corresponding to the water level at the lower segments of 13th and 14th fuel elements and upper half of the 27th and 31st fuel elements shown in Figure 2, which meets common sense physics. However, further detailed validation of the results against the applicable experimental results for the aged fuel channels seems necessary. Also, the necessity of further detailed modeling of the circumferential variation of the pressure tube temperatures needs to be checked for the best practical use of this model for safety analyses at a massive quantity.

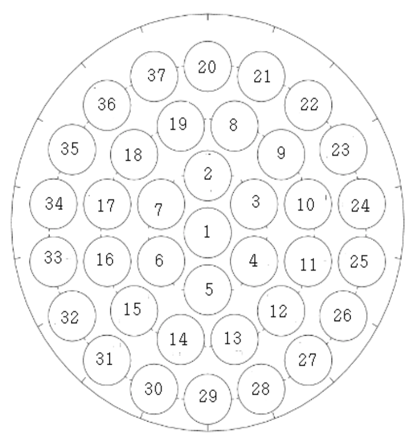


Figure 2 Numbering of the Fuel Elements in the Crept Pressure Tube Fuel Channel Model for Aged Plant Simulation.

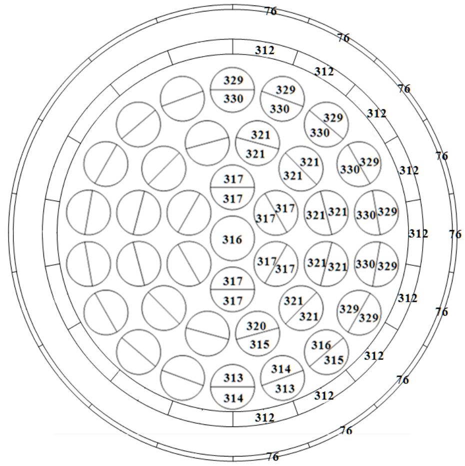


Figure 3 Predicted Fuel and PT/CT Temperatures at the 12th Bundle Location for W10 Channel under the Stratified Two-Phase Flow Condition.

2.1.2 CATHENA Simulation of the Reactor Outlet Header (ROH) 100% Break Analysis

A more elaborate model for the same purpose has been developed, as shown in Figure 4, where the bundle is rotated by 30o in counter-clockwise, and the pressure tube is divided into 36 segments rather than 18 of the previous one for steady state. This model is the same fuel channel model as that used for the safety analysis for the Wolsong Unit 1 refurbishment plant. This model is symmetric along the vertical line passing through the center line of the center element as shown in Figure 4.

Three different 37 fuel models are considered and their results for 100% ROH break were compared. They are uncrept fuel channel model, concentric crept fuel channel model and the vertically offset crept fuel channel model. Both the concentric crept fuel channel model and vertically offset crept fuel channel model consider the radial expansion of the pressure tube i.e., 0.0, 1.5, 2.0, 2.5, 3.0, 3.0, 2.5, 2.0, 1.5, 0.0 % in diameter, but the concentric model assumes that the center line of the fuel bundle is located at the center line of the crept pressure tube, whereas the vertically offset model assumes that the fuel bundle sits at the bottom of the pressure tube due to gravity. The analysis was performed using the same thermal and hydraulic boundary conditions of the fuel bundles, inlet and outlet headers for 200 sec after the accident initiation. The initial steady state results showed a 9% increase of the channel flow rate for the two crept channel models as compared with the uncrept channel model, which is consistent with author's expectation as the flow area increased as a result of the pressure tube creep. The initial fuel surface temperatures of the center pin, inner ring pin, middle ring pin, and the outer ring pin were almost the same with only a 0.14 to 0.32% difference. However, the results of the 100% ROH break analyses revealed some differences between the uncrept channel model and the crept channel models, but no significant difference between the two crept channel models. As can be seen in the analyses results, the channel flows shown in Figures 5-7 quickly drops to 0 kg/s in about 25 sec since when the fuel and pressure tube temperatures continue to rise. During 25 to 50 sec, where the flow is nearly zero, fairly good fuel cooling seems to persist, and the temperatures of all the fuels rise in a similar fashion for all three models in Figures. 8-10, whereas the pressure tube temperature for the uncrept model rises more sharply than those for the deformed models, which is thought to be due to the decrease of the annulus gap between pressure tube and calandria tube, and an increase of the distance between the outer fuel bundle and pressure tube, which affect the radiation heat transfer rate. Another noticeable difference is that the rate of fuel temperature rise between the uncrept model and the two crept models shows a significant difference from t= 170 sec into the accident. This is thought to be caused by the contribution of the radiation heat transfer to the total heat transfer becoming dominant after this point. However again, the difference between the concentric model and the vertically offset model turns out to be insignificant, as both results in Figures 11-12 show almost the same trends.

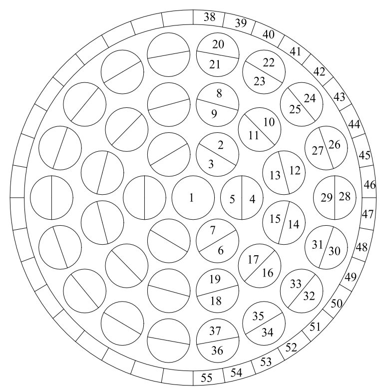


Figure 4 Heat Structure Numbering System of the More Elaborated Channel Model.

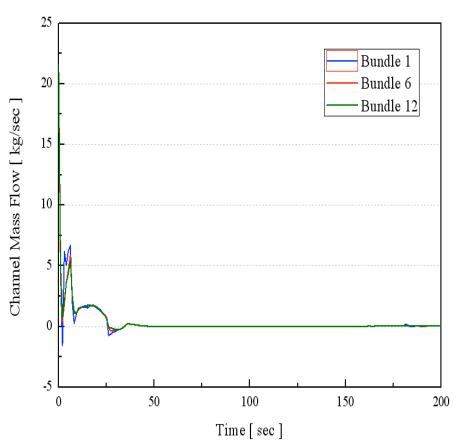


Figure 5 O-06 Channel Flow for the uncrept channel model.

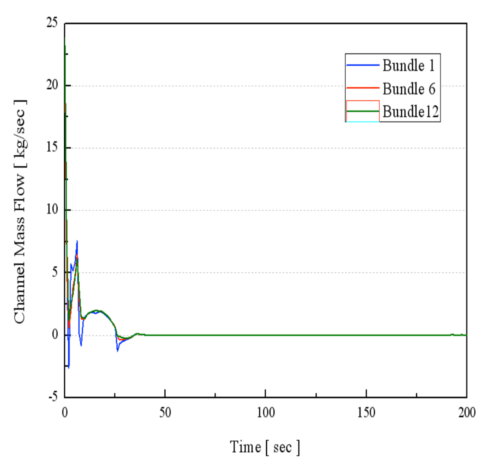
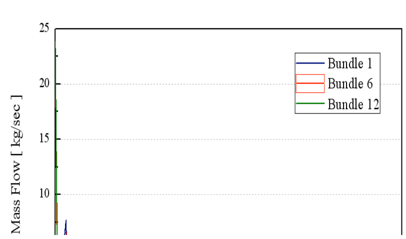
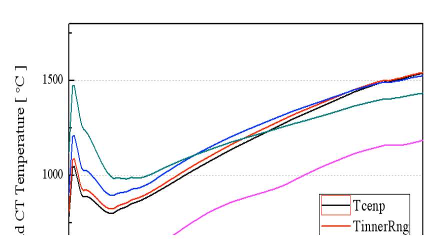


Figure 6 O-06 Channel Flow for the Concentric channel model.





3. Conclusion

The current practice of the CANDU safety analysis assumes that the fuel bundles stay in a manner concentric to the pressure tube centerline even in the crept pressure tubes, whereas in reality the bundles sits at the bottom of the pressure tube. The major thermal-hydraulic parameters of safety concern are the fuel temperature and heat flux at the outer surface of the calandria tube at the time of PT/CT contact. In the current study, a CATHENA fuel channel model that can account for the effect of the vertical offset of the fuel bundles in a crept pressure tube on the fuel and pressure tube cooling is developed. A more elaborate 37-pin models with and without vertical offset of the bundle due to gravity in the crept fuel channels are developed and tested for ROH 100% break LOCA accident, and compared. As a result, it was found that the difference between the uncrept fuel channel model and the two crept fuel channel models is quite significant, whereas the difference between the two crept fuel channel models is insignificant. Therefore, it is concluded that the use of a concentric crept fuel channel model for an aged CANDU-6 safety analysis is justifiable for the first 200 sec into an accident.

Acknowledgements

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4. References

[1] CATHENA MOD-3.5d Theory Manual, 153-112020-STM-001, Rev.0, December 2005, AECL.

[2] MATRIX 1.05, A Stand-Alone Preprocessor Utility for CATHENA Users, J.B. Hedley, AECL, Oct. 1999.

[3] P.S. Yuen, K.A. Haugen, D.G. Litke, R.G. Moyer and H.E. Rosinger, "The Experimental Measurement of Circumferential Temperature Distributions Developed on Pressure Tubes Under Stratified Two-Phase Flow Conditions: Tests 1 to 5," Proceedings of the 10th Annual CNS Conference, Ottawa, ON, June 1989.