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Development of the T/H model for Aged core in Wolsong-1 NPP

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1. Introduction

Pressurized Heavy Water Reactors(PHWR) have unique core configuration. The pressure tube is the pressure boundary like the reactor vessel in PWR. It surrounds fuel bundle and coolant and separates coolant from moderator. But as operation time increases, the pressure tube gets more irradiation and begins to creep radially and elongate axially. Pressure tube radius changes due to creep affects core channel flow thermal-hydraulically and softens neutron spectrum physically. Also down stream part of steam generator tube is deposited with magnetite which comes mostly from the outlet feeder pipes. Deposited magnetite increases the surface roughness of channel and is expected to affect flow stream. Feeder orifice degradation is also another aging effect. These aging phenomena are taken into account in thermal hydraulic model for Wolsong 1 NPP

2. Analysis Methods

As operating year is increased, total radiation dose of pressure tube increases under high coolant temperature condition. It is generally known that the diameter of pressure tube is increased due to the high coolant temperature and the accumulated irradiation. The increase of pressure tube diameter makes more coolant bypass fuel channel and makes it harder to transfer heat from the fuel to the coolant locally. This aging mechanism makes the critical channel power (CCP) decrease. Here we quantify the degree of pressure tube diameter increase in terms of creep.

2.1. Pressure tube creep

There have been some selective creep measurements on the pressure tubes at Wolsong-1 NPP. Actually, total 51 channels were measured at four different times; 11, 17, 14 and 9 channels respectably at 1990, 1992, 1994 and 2001. Based on the measured creep data, the diametral creep for each fuel bundle location in all 380 channels is predicted using RC-1980 program. The accuracy of RC-1980 prediction is measured by comparing the prediction value with measured creep at the 51 measured channels. The average and standard deviation of the difference are 2.88% and 9.99% respectively and Fig 1 shows the comparison results. And these creep data for 380 channels are averaged into 7 groups for each different four passes in the safety analysis CATHENA model as in Fig. 2.

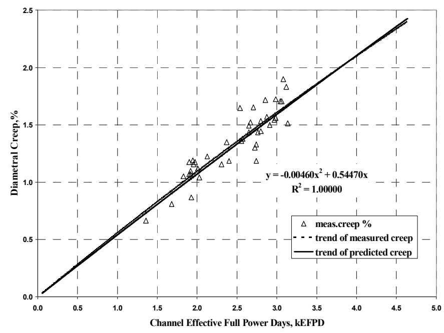

Fig.1 Comparison of measured and predicted creep

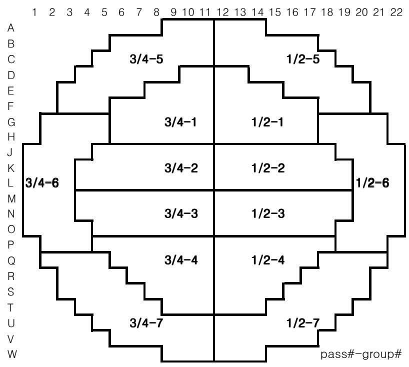

Fig.2 Fuel channel grouping

2.2. Fouling of primary heat transport system(PHTS)

The roughness of the inner surface of all the fuel channels, feeder pipes and the steam generator tubes increasing along the plant operation is modeled in the PHTS thermal-hydraulic NUCIRC model and provided to the CATHENA model. Also, the magnetite deposits reduce the tube inside hydraulic diameter. These effects are dominant along the cold leg portion since the water temperature is colder resulting in a greater deposition rate. The relative roughness is multiplied by the steam generator tube inner diameter for the respective section to obtain the absolute roughness, which is input to CATHENA model.

2.3. Fuel cross section generation

As pressure tube creeps radially, its radius increases resulting in the change of coolant volume and consequently neutron spectrum also changes. The creep rate depends on the location in the core because creep is known to be a function of coolant temperature, fast neutron flux and the accumulated operation time. To take into account of the pressure tube creep effect in physics more accurately, fuel and reflector cross section for different locations is generated using WIMS-AECL.

Table 1 Region-wise creep rate

Region	Number of channels	Creep(%)	Inner Radii(cm)	Outer Radii(cm)	Volume
Region A(bundle 6-11)	192	2.3072	5.3179	5.7410	38.8353
Region B(bundle 1-5,12)	192	1.1594	5.2583	5.6858	36.8529
Region C(bundle 6-11)	188	1.8851	5.2960	5.7207	38.1037
Region D(bundle 1-5,12)	188	0.9402	5.2469	5.6752	36.4768

Creep rate for each bundle location is calculated based on measurement data and experimental data (RC-1980). For practical points of view, creep rate for each bundle position is averaged over inner region(A,B) and outer region(C,D) each for axial and radial direction. Axial region grouping is done according to the magnitude of irradiation. For each channel there are 12 bundles and bundle location 1-5 and 12 is grouped as low creep rate region (region B and D) and bundle location 6-11 is grouped as high creep rate region (region A and C). Table 1 shows region-wise creep rates. The creep rates of region A and C are higher than those of region B and D because neutron flux distribution is much higher in the bundle location 6-11. Four unique cross sections for different regions are made to take into account of different pressure tube creep rates.

Table 2 Void reactivity

Coolant density (g/cm^3)	Aged core reactivity	Fresh core reactivity
0.8296	0.998240	0.996790
0.0001	1.014160	1.012366
Void reactivity	15.7 mk	15.4 mk

Cross section calculation is done with WIMS-AECL instead of POWDERPUFS because experimental correlation used in POWDERPUFS is locked on fresh 37-element natural enrichment fuel. The creep effect on physics is quantified by using measured pressure tube radius. The void reactivity of aged core is compared to that of fresh core. About 0.4mk of void reactivity increase due to creep can be seen as in the table 2.

3. Results

The primary heat transport system model is verified to demonstrate that aging parameters like pressure tube creep and fouling in feeder and steam generator tube used in Wolsong-1 NUCIRC plant model were properly incorporated into the respective CATHENA component models. The model verification is done with CATHENA simulations at 81.59% power level against NUCIRC prediction.

Steady state values of several important primary circuit parameters, as predicted by CATHENA, are shown in Table 3. Most parameter values are in good agreement between CATHENA and NUCIRC model. It can be said that CATHENA reproduces reference plant model (NUCIRC model) well and can be used for the analysis of an aged Wolsong 1 plant.

REFERENCES

[1] Study of Coupling Effects in Steady-state CANDU core for different operating conditions, 23rd Annual conference of Canadian Nuclear Society

[2] NUCIRC Mod2.000 User's Manual,TTR-422,1997,AECL

[3] CATHENA Mod-3.5c Input Reference, 1999, AECL

Table 3 Comparison of CATHENA with NUCIRC

Parameter	CATHENA Prediction	NUCIRC Prediction
P_{RIH2}/ P_{ROH1} (MPa)	11.23 / 10.00	11.19 / 9.95
P_{RIH4}/ P_{ROH3} (MPa)	11.15/ 9.99	11.12 / 9.95
P_{RIH6}/ P_{ROH5} (MPa)	11.22/ 10.00	11.20 / 9.94
P_{RIH8}/ P_{ROH7} (MPa)	11.17/ 9.99	11.13 / 9.93
T_{RIH2}/ T_{ROH1} (°C)	263.99/ 302.50	263.40/ 302.61
T_{RIH4}/ T_{ROH3} (°C)	264.17/ 302.18	263.42/ 303.14
T_{RIH6}/ T_{ROH5} (°C)	264.00/ 303.15	263.45/ 303.45
T_{RIH8}/ T_{ROH7} (°C)	263.95/ 303.09	262.60/ 303.64
Core Pass 23 Flow (kg/s)	2148	2116
Core Pass 41 Flow (kg/s)	2144	2100
Core Pass 67 Flow (kg/s)	2104	2079
Core Pass 85 Flow (kg/s)	2091	2087
Header 23 Pressure drop (MPa)	1.23	1.23
Header 41 Pressure drop (MPa)	1.16	1.17
Header 67 Pressure drop (MPa)	1.22	1.23
Header 85 Pressure drop (MPa)	1.18	1.20