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POWER COEFFICIENT CALCULATION OF A CANDU REACTOR

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ABSTRACT

The power coefficient of a CANDU reactor was calculated for natural uranium and DUPIC fuel cores using lattice parameters generated by WIMS-AECL code. The simulation was performed for the natural uranium core first and the result was compared with that of the CANDU-6 physics design manual. It was found that the power coefficient based on WIMS-AECL lattice parameters is consistently larger in magnitude compared with that of the physics design manual. The same calculation for a CANDU core loaded with the DUPIC fuel has shown that the power coefficient of the time-average DUPIC fuel core is more negative compared with that of the natural uranium core, which could be attributed to more xenon effect, less coolant density feedback and more Doppler effect.

I. INTRODUCTION

At present, the neutronic and thermal-hydraulic coupling calculation for a Canada deuterium uranium (CANDU) reactor is well-established using the lattice code POWDERPUF-V,[1] core simulation code RFSP[2] and thermal-hydraulic analysis code NUCIRC.[3] In fact, the lattice code is implemented in the core simulation code such that it provides lattice parameters interactively during the core simulation. Therefore the lattice conditions including the fuel composition, fission products concentration, fuel temperature, coolant density and moderator density are simultaneously considered during the core simulation. However, the disadvantage of this practice is that the application of the lattice code is limited to the natural uranium fuel only because of the empirical correlations implemented.

The recent development of advanced CANDU fuels, such as slightly enriched uranium (SEU),[4] mixed oxide (MOX)[5] or direct use of spent pressurized water reactor fuel in CANDU reactors (DUPIC),[6] also requires an estimation of the stability of the core against the feedback due to thermal-hydraulic and neutronic perturbations. Such a requirement of new fuel development necessitates the use of a general-purpose lattice code such as WIMS-AECL.[7] However, the introduction of such a lattice code to the coupling calculation has not been well-established. In this study, the coupling calculation between RFSP and NUCIRC codes are described and the results

of power coefficient calculation for the natural uranium and DUPIC fuel cores are presented.

II. ANALYSIS TOOLS AND MODEL DESCRIPTION

The computer codes used for the coupling calculation are externally linked so that almost no change of each code is needed. However, in order to facilitate the coupling of full core neutronic and thermal-hydraulic calculations, a few modifications were made to RFSP code and a program was written to transfer data from one code to another.

II.1 WIMS-AECL

WIMS-AECL is a transport code used to generate physics parameters including the composition of the fuel, flux distribution, multiplication factor and reaction rates. Based on the results of WIMS-AECL calculation, the cross-section table is prepared in two forms: a standard and a perturbed cross-section set. The perturbed cross-sections are grid-based, which are used for kinetic calculations of RFSP code. The first approximation to be used in the calculation of grid-based cross-section data from WIMS-AECL is that the fuel composition is a function of fuel thermal neutron irradiation only, and no account will be taken of the variation in compositions that would result from the varying coolant conditions and power levels that different fuel bundles would actually experience during their individual histories of irradiation in a reactor.

II.2 RFSP

RFSP is a three-dimensional diffusion code used for CANDU core analysis. It performs a wide variety of calculations such as time-average, instantaneous and refueling simulations based on the solution of the finite-difference form of the neutron diffusion equation in two energy groups. The RFSP incorporates the lattice parameters generated by WIMS-AECL by interpolating them for specific data points of the fuel burnup, fuel temperature, coolant temperature and coolant density. The interpolations are actually performed using the adaptive Lagrange method. Although the lattice parameters actually vary as three-dimensional functions for the perturbations currently treated, they are treated through successive one-dimensional interpolations.

II.3 NUCIRC

NUCIRC is a thermal-hydraulic code used to get the coolant condition such as pressure, temperature and quality distribution, which are used to determine the critical channel power. Normally the coolant boiling occurs in many of the fuel channels in the reactor at its full power condition. In the NUCIRC code, it is considered that the boiling takes place when the enthalpy exceeds the saturation point. For the single phase region, the density of the coolant is calculated from the enthalpy and the pressure, where the enthalpy of the coolant is determined by the heat balance between the fuel and coolant. On the other hand, the density of the two-phase mixture for the boiling region can be determined by the saturation properties of each phase and the quality, where the saturation properties are found from the steam table correlation.

II.4 Coupling Model

In this study, the thermal-hydraulic coupling calculation is performed for the time-average core which represents the near equilibrium behavior of a time-dependent core. The time-average core is determined using the standard cross-section set which contains burnup-dependent cross-sections, reflector cross-sections and xenon constants. Because the standard cross-section set can not be used for the interpolation of thermal-hydraulic parameters, a transient core model is reconstructed by RFSP code based on the time-average core model. Therefore the core properties of the transient core are the same as those of the time-average core. However the transient core model reads in the perturbed cross-section table to be used for thermal-hydraulic coupling calculation.

Typically the thermal-hydraulic coupling calculation is performed by a kinetic routine of RFSP code which are used for the safety analysis too. The kinetic routine reads the coolant density and temperature which are calculated by NUCIRC, and the fuel temperature is externally calculated based on the bundle power. Then, using the thermal-hydraulic parameters, RFSP calculates the bundle power distribution again to be used for the thermal-hydraulic calculation. In order to facilitate the data transfer between RFSP and NUCIRC, a utility program is used to convert the data format to be suitable for both codes. The thermal-hydraulic coupling between RFSP and NUCIRC is sketched in Fig. 1.

III. CORE CALCULATION

The CANDU core is divided into two radial regions based on the channel flow rate distribution and the physics analysis model also utilizes the two-region model (inner and outer core) to determine the average discharge burnup for the candidate refueling strategy. The discharge burnups of inner and outer core are determined such that the reactor is critical and the maximum channel and bundle powers are the lowest. During the critical core search, the zone controller water level is typically assumed to be 50%. In a typical CANDU core analysis, all reactivity devices and structural materials are considered in the core model.

III.1 Natural Uranium Core

The reactivity feedback effect of the natural uranium CANDU reactor was calculated for three different effects: the fuel temperature effect, coolant density term and xenon redistribution, which are given in Table 1. It can be seen that the prompt feedback (Doppler effect) is negative for the power level increase. However, if we include the coolant density change, which is a delayed effect due to the heat transfer from the fuel to the coolant, there is a positive contribution to the core excess reactivity. When the time scale is even longer, the redistribution of xenon provides a strong negative reactivity feedback. For example, when the power level increases from 50% to 100% full power, the reactivity feedback is -0.31, 0.45, and -5.03 mk for Doppler, coolant density and xenon effect, respectively. The total reactivity feedback is given in Table 2 and compared with that given in the CANDU-6 physics design manual.[8] Generally the trend of the power coefficient change is similar for PPV-based (physics design model) and WIMS-based calculations, however, there is a difference in magnitude, which could be due to several factors

such as the inherent difference in the time-average core power distribution, fuel burnup distribution, lattice model, etc.

III.2 DUPIC Fuel Core

The reactivity feedback effects were calculated for the time-average DUPIC core using the same conditions as were used for the natural uranium core. The change of each reactivity component is given in Table 3 when the power level changes from 100% full power. The total reactivity change of the DUPIC fuel core is compared with that of natural uranium core in Fig. 2. It can be seen that the power coefficient of the DUPIC fuel core is a slightly more negative number compared with that of natural uranium core, which was already given in Table 2. The more negative value of the power coefficient comes from both the fuel temperature and coolant density terms. For the DUPIC fuel, a poison material is mixed in the center rod of the fuel bundle such that the slope of reactivity increase upon coolant voiding is the same for both the natural uranium and DUPIC fuels of equilibrium burnup. This constraint actually results in a coolant void reactivity of the DUPIC fuel (~12mk) be smaller than that of the natural uranium fuel (~14mk). The isotopic composition of the equilibrium DUPIC fuel also provides more fuel temperature effect than that of the equilibrium natural uranium fuel. As a result, the DUPIC core possesses a more negative power coefficient compared with the natural uranium core.

IV. CONCLUSION

The thermal-hydraulic coupling calculation has been established using the grid-based lattice parameters generated by WIMS-AECL code. The comparison of power coefficient against the physics design manual has shown that the overall trends of power coefficient generally agree with each other. The comparison of power coefficients between natural uranium and DUPIC fuel core has shown that the DUPIC fuel core possesses a more negative value compared with the natural uranium core, which indicates that the DUPIC fuel core is more stable against the power level perturbations.

Acknowledgements

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Table 1. Reactivity Feedback Effect (mk) for Natural Uranium Core

Power Level (%)	Perturbation Type			
	T_{fuel}	ρ_{cool}	Xenon	Total
130	-0.11	1.19	0.14	1.22
120	-0.09	0.62	-0.18	0.35
110	-0.04	0.20	-0.23	-0.07
100	0.00	0.00	0.00	0.00
90	0.06	-0.11	0.40	0.35
80	0.12	-0.20	1.05	0.97
70	0.17	-0.27	1.91	1.81
60	0.25	-0.38	3.17	3.04
50	0.31	-0.45	5.03	4.89

Table 2. Comparison of Power Coefficient

Power Level (%)	Natural Uranium (Design Manual)	Natural Uranium (WIMS-based)	DUPIC Fuel (WIMS-based)
125	0.0222*	0.0320	0.0030
115	0.0061	0.0050	-0.0230
105	-0.0081	-0.0230	-0.0550
95	-0.0158	-0.0399	-0.0759

* mk/%power

Table 3. Reactivity Feedback Effect (mk) of DUPIC Fuel Core

Power Level (%)	Perturbation Type			
	T_{fuel}	ρ_{cool}	Xenon	Total
130	-0.29	1.11	-0.75	0.07
120	-0.19	0.57	-0.78	-0.40
110	-0.10	0.20	-0.55	-0.45
100	0.00	0.00	0.00	0.00
90	0.10	-0.10	0.76	0.76
80	0.19	-0.18	1.91	1.92
70	0.27	-0.25	3.38	3.40
60	0.37	-0.34	5.32	5.35
50	0.45	-0.40	7.96	8.01

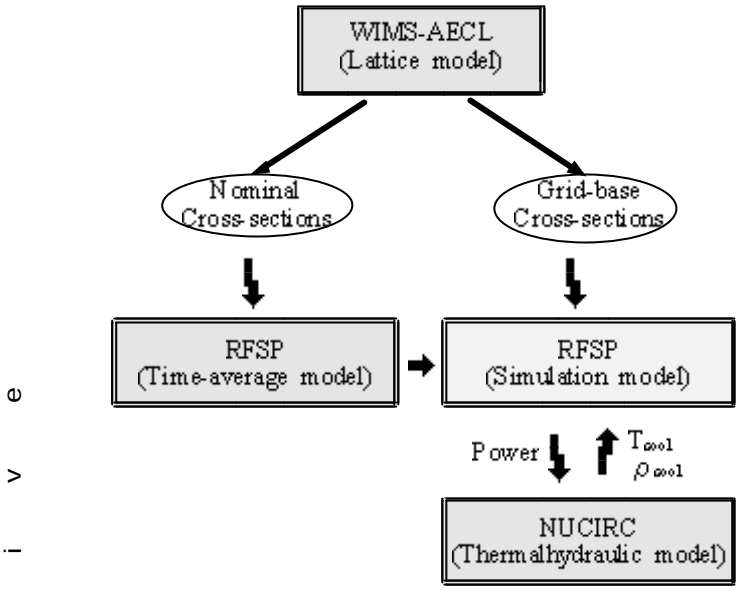

Fig. 1 Coupling Calculation of RFSP and NUCIRC

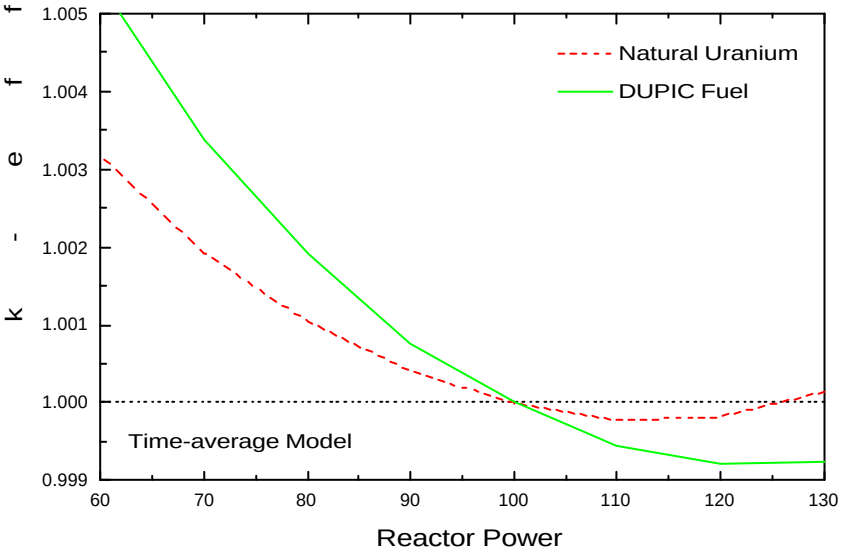

Fig. 2 Comparison of Core Multiplication Factors