POWER COEFFICIENT CALCULATION OF A CANDU REACTOR

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ABSTRACT

A coupling calculation between the neutronic and thermal-hydraulic codes has been established for the estimation of the power coefficient using the lattice parameters generated by WIMS-AECL code. In order to utilize the existing capability of RFSP code, a few modifications were made to the RFSP and an interface program between RFSP and NUCIRC was written. The power coefficients were calculated for natural uranium and DUPIC fuel cores. The simulation 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 fuel temperature and less coolant density feedback effects. However, this study has also shown that the fuel temperature feedback effect predicted by lattice codes needs to be validated, especially for the irradiated natural uranium fuel.

1. INTRODUCTION

For the inherent safety of reactor operation, it is important to estimate the reactivity feedback due to power level change. The reactor power level change affects the reactivity due to the fuel temperature change, coolant density change and concentration of major fission products. The temperature feedback is due to the resonance broadening of fertile isotopes when the fuel temperature changes and is regarded as a prompt response to the power change. The coolant density feedback occurs when the coolant temperature changes due to the perturbation of heat transfer from the fuel. Because the time constant of the heat transfer to the coolant is relatively large ($\sim$10 secs) compared with the fuel temperature feedback, the coolant density feedback is regarded as a delayed response to the power change. Furthermore, if we consider a steady-state power level change for an even longer time period, the effect of certain neutron absorbers in the fuel (e.g., xenon, samarium and rhodium) on the core reactivity is not negligible.

Currently, the power coefficient calculation for a CANDU reactor is well-established using the lattice code POWDERPUF-V (PPV),[1] core simulation code RFSP[2] and thermal-hydraulic code NUCIRC.[3] The recent development of advanced CANDU fuels, such as slightly enriched uranium, mixed oxide or spent pressurized water reactor fuel, 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.[4] However, the introduction of such a lattice code to the power coefficient calculation has not been well-established because of the difficulty in coupling the neutronic and thermal-hydraulic parameters. In this study we have established the coupling calculation between RFSP and NUCIRC with the lattice parameters provided by WIMS-AECL utilizing the existing capability of RFSP code in full strength.

2. ANALYSIS TOOLS AND MODEL DESCRIPTION

The neutronic and thermal-hydraulic codes are externally coupled so that almost no changes in each code are needed. However, in order to facilitate the coupling, a few modifications were made to RFSP code and a program was written to transfer data from one code to another. This section describes the computer codes and data used for the power coefficient calculation.

2.1 WIMS-AECL

WIMS-AECL is a transport code used to generate physics parameters including the composition of the fuel. The lattice parameters for the RFSP code are prepared in two forms: standard and perturbed cross-section tables.

2.1.1 Standard Cross-section Table

The standard cross-section table is typically used for the steady-state core calculations such as time-average and refueling simulations. The table contains eight homogenized cell parameters as a function of fuel burnup to characterize the properties of an individual fuel bundle, which are the fast neutron diffusion coefficient, thermal neutron diffusion coefficient, fast neutron absorption cross-section, thermal neutron absorption cross-section, effective thermal neutron fission yield cross-section, effective fast neutron moderation cross-section, effective fission energy cross-section, and ratio of the thermal neutron flux in the fuel over the cell-average thermal neutron flux.

The cross-section table also includes cell parameters for the reflector material, which is heavy water for the CANDU reactor. Because there is no fission reaction in the reflector, the cross-sections are given for the diffusion, absorption and scattering reactions which are not dependent on fuel burnup. In the WIMS-AECL lattice calculation, the cross-sections of the reflector material are obtained from the heavy water moderator.

The xenon properties are also produced from the WIMS-AECL lattice calculation when the standard cross-sections are generated. In the CANDU core analysis code RFSP, the existence of ^{135}Xe is described by an incremental cross-section $\Delta\Sigma_a^{Xe}$ which is additive to the reference thermal absorption cross-section with equilibrium ^{135}Xe concentration. The standard cross-section table contains three xenon property constants such as the relative iodine yield, the reference xenon concentration, and the proportional constant as a function of fuel burnup.

2.1.2 Perturbed Cross-section Table

The RFSP code accepts the perturbed cross-sections to be used for the kinetic calculation. The perturbed cross-sections are grid-based, which is interpolated through the *WIMSTABLES option of RFSP. 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 for the variation of composition 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. The burnup-dependent composition of fuel material is calculated by WIMS-AECL with all state variables at nominal values for the reactor of interest such that the material

compositions are obtained as a function of fuel burnup. Then, the burnup-dependent material compositions are converted into a material library which is directly accessible by subsequent WIMS-AECL perturbation calculations.

2.2 RFSP

The 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 the processing code WIMTAB by interpolating them for specific data points of the fuel burnup, fuel temperature, coolant temperature and coolant density.

2.2.1 Interpolation of State-Dependent Data from WIMS-AECL

The RFSP has the option *WIMSTABLES in the CERBERUS module, enabling state-dependent cross-section generation from the cross-section table prepared by the processing code WIMTAB using WIMS-AECL results. The *WIMSTABLES option allows for lattice parameters to be computed from the WIMS-AECL cross-section table rather than from PPV calculation. The *WIMSTABLES card assumes that lattice properties have been read into the appropriate index and records of the RFSP direct-access file for all fuel types under consideration, and must be used in every case of the transient being simulated with WIMS-AECL lattice parameters. The interpolations required to produce lattice parameters for use in the CERBERUS module of RFSP are performed using the adaptive Lagrange method. The lattice parameters are treated through successive one-dimensional interpolations, though they actually vary as three-dimensional functions of the perturbation variables.

2.2.2 Modification of CATHENA Option

Currently, the RFSP static calculation is linked with the thermal-hydraulic calculation by NUCIRC. However, the *WIMSTABLES option is prepared for the kinetic calculation by the CERBERUS routine of RFSP because the perturbed cross-sections are typically required for the dynamic calculations. In the CERBERUS routine, the thermal-hydraulic data is read through the CATHENA format and, therefore, the RFSP was modified to accommodate the full core channel model, and an interface program was written for the format change of thermal-hydraulic data and the prediction of average fuel temperature.

2.3 NUCIRC

NUCIRC is a thermal-hydraulic code used to get the coolant condition of a fuel channel. The NUCIRC code was originally developed for the 37-element fuel bundle which is the standard fuel type of the CANDU-6 reactor. When DUPIC fuel is loaded in a fuel channel, the bundle radial power distribution and the channel axial power shape will be different from those of a natural uranium core, which will affect the critical heat flux condition. However, because NUCIRC adopts a single channel model for the fuel channel analysis, it accounts for the average property of the fuel bundle radial power profile. Therefore, it is expected that the effect of the axial power profile will be greater than that of the bundle radial power profile when the DUPIC fuel is analyzed, which indicates that the use of NUCIRC for the thermal-hydraulic analysis of the DUPIC fuel core is reasonable too.

2.4 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. The cross-sections are read by RFSP using the *READ CARD option. 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 based on the time-average core calculation. 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 the thermal-hydraulic coupling calculation. Typically, the thermal-hydraulic coupling calculation is performed by the kinetic routine CERBERUS of RFSP code, which is used for the safety analysis, too. The CERBERUS 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, the RFSP again calculates the bundle power distribution again to be used for NUCIRC. At present, because the CATHENA format is used to transfer thermal-hydraulic data, a utility program is used to convert the data format to make it suitable for both RFSP and NUCIRC codes.

3. CORE CALCULATION

The power coefficient was calculated by incorporating the fuel temperature and coolant density feedbacks. However, in order to see the effect of xenon distribution, the power coefficients were calculated for three different conditions, such as:

i) equilibrium xenon concentration (burnup-dependent xenon concentration),
ii) distributed xenon at a full power level, and
iii) distributed xenon at a specified power level.

For the first case, the equilibrium xenon concentration which depends on the fuel burnup only is used for the power coefficient calculation. The second model calculates the distributed xenon concentration for the full power level, and the xenon distribution is used for the different power levels assuming that the xenon concentration change is delayed appreciably compared with the fuel temperature and coolant density changes. For comparison, the third model calculates the power coefficient using the distributed xenon concentrations which are obtained for the specified power levels.

3.1 Natural Uranium Core

The results of these calculations are given in Table I. The variation of k_{eff} is shown in Fig. 1, where the k_{eff} of the full power core is normalized to 1.0. The power coefficient is composed of the fuel temperature and coolant density terms, which were calculated in parallel, as shown in Table II, along with the xenon effect. This calculation has shown that the power coefficient predicted by WIMS-AECL and RFSP is somewhat different from that given in the CANDU-6 physics design manual.[5] For the power reduction from 100% to 50%, the Doppler effect provides 0.31mk, while the coolant density change does -0.45mk, resulting in a net decrease in the core reactivity.

The effects of fuel temperature and coolant density were studied more using the lattice

code WIMS-AECL. In Fig. 2, the variations of k_{inf} are shown for natural uranium fuels in fresh and irradiated (3980 MWd/T) conditions. For the irradiated fuel, the temperature effect is relatively smaller compared with the fresh fuel because of transuranic isotopes which have resonances for fission reactions. It should also be noted that there is an inherent difference between WIMS-AECL and PPV for eigenvalue calculations. For example, the hot shutdown reactivity for the irradiated fuel is ~1mk and ~3mk by WIMS-AECL and PPV, respectively. Therefore, the Doppler effect for WIMS-based calculation is by 60% smaller than for PPV-based calculation, which is a large enough difference to change the sign of the power coefficient.

3.2 DUPIC Fuel Core

The power coefficients were calculated for the same conditions that were used for the natural uranium core. Table III shows the k_{eff} for different power levels and Fig. 3 shows the variations of k_{eff} which are normalized to the k_{eff} the full power core. It can be seen that the power coefficient of the DUPIC fuel core is slightly more negative than that of the natural uranium core, which is summarized in Table IV for the fuel temperature and coolant density terms. Compared with natural uranium fuel, DUPIC fuel contains a poison material in the center of the fuel bundle such that the slope of reactivity increase upon coolant voiding is the same for both natural uranium and DUPIC fuel cores at the equilibrium condition. This constraint actually results in the coolant void reactivity of the DUPIC fuel (~12mk) being smaller than that of the natural uranium fuel (~14mk). The isotopic composition of the equilibrium DUPIC fuel also provides a greater fuel temperature effect than that of the equilibrium natural uranium fuel, as shown in Fig. 4. As a result, the DUPIC fuel core possesses a more negative power coefficient compared with the natural uranium core, as summarized in Table V. In addition, it is understood that the flatter power distribution of the DUPIC fuel core enhances a greater xenon effect when the power level changes, compared with the natural uranium core which has a center-peaked power shape.

4. CONCLUSION

The comparison of power coefficients between natural uranium and DUPIC fuel cores has shown that the DUPIC fuel core possesses a more negative coefficient compared with the natural uranium core, which could be attributed to the greater Doppler and lesser coolant density feedback effects. However, the CANDU-6 core simulation has shown that the power coefficient based on WIMS-AECL lattice parameters is not consistent with that of the physics design manual. There is a large difference in the fuel temperature coefficient estimation between the physics design code PPV and the general-purpose transport code WIMS-AECL. Therefore, the validation calculation of WIMS-AECL for predicting key safety parameters such as the coolant void reactivity and fuel temperature coefficient should be done in the near future to establish the credibility of the lattice parameter generation of the abnormal lattices by WIMS-AECL.

Acknowledgements

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TABLE I. EFFECT OF POWER LEVEL ON k_{eff} OF NATURAL URANIUM CORE

Power Level (%)	Equilibrium Xenon	Full Power Xenon	Distributed Xenon
130	1.00164	1.00160	1.00079
120	1.00110	1.00105	1.00047
110	1.00075	1.00068	1.00042
100	1.00060	1.00052	1.00065
90	1.00055	1.00047	1.00105
80	1.00052	1.00044	1.00170
70	1.00048	1.00042	1.00257
60	1.00046	1.00039	1.00383
50	1.00045	1.00038	1.00571

TABLE II. FEEDBACK EFFECT OF NATURAL URANIUM CORE

Power Level (%)	Average Property			Reactivity Effect (mk)		
	T_{fuel}	T_{cool}	ρ_{cool}	T_{fuel}	ρ_{cool}	Xenon
130	672.2	296.1	0.69748	-0.11	1.19	-0.94
120	634.8	294.3	0.73656	-0.09	0.62	-0.71
110	598.2	292.3	0.77404	-0.04	0.20	-0.39
100	562.3	289.9	0.80179	0.0	0.0	0.0
90	527.4	287.5	0.81120	0.06	-0.11	0.45
80	493.8	285.1	0.81659	0.12	-0.20	1.13
70	461.4	282.8	0.82178	0.17	-0.27	2.01
60	430.1	280.5	0.82679	0.25	-0.38	3.35
50	400.0	278.2	0.83166	0.31	-0.45	5.17

TABLE III. EFFECT OF POWER LEVEL ON k_{eff} OF DUPIC FUEL CORE

Power Level (%)	Equilibrium Xenon	Full Power Xenon	Distributed Xenon
130	1.000760	1.000720	0.99937
120	1.000320	1.000280	0.99934
110	1.000040	0.999999	0.99957
100	0.999919	0.999895	1.00012
90	0.999913	0.999891	1.00088
80	0.999922	0.999906	1.00203
70	0.999939	0.999913	1.00351
60	0.999951	0.999927	1.00547
50	0.999974	0.999945	1.00815

TABLE IV. FEEDBACK EFFECT OF DUPIC FUEL CORE

Power Level (%)	Average Property			Reactivity Effect (mk)		
	T_{fuel}	T_{cool}	ρ_{cool}	T_{fuel}	ρ_{cool}	Xenon
130	669.5	298.2	0.68200	-0.29	1.11	-1.57
120	632.7	296.3	0.72407	-0.19	0.57	-1.16
110	596.5	294.1	0.76430	-0.10	0.20	-0.65
100	561.0	291.5	0.79576	0.0	0.0	0.0
90	526.5	288.9	0.80798	0.10	-0.10	0.76
80	493.3	286.4	0.81389	0.19	-0.18	1.90
70	461.1	283.9	0.81947	0.27	-0.25	3.36
60	430.0	281.5	0.82485	0.37	-0.34	5.29
50	400.0	279.0	0.83008	0.45	-0.40	7.91

TABLE V. COMPARISON OF POWER COEFFICIENT (WIMS-BASED)

Power Level (%)	Natural uranium core	DUPIC core
125	0.055*	0.044
115	0.037	0.028
105	0.016	0.010
95	0.005	0.000

* mk/%power

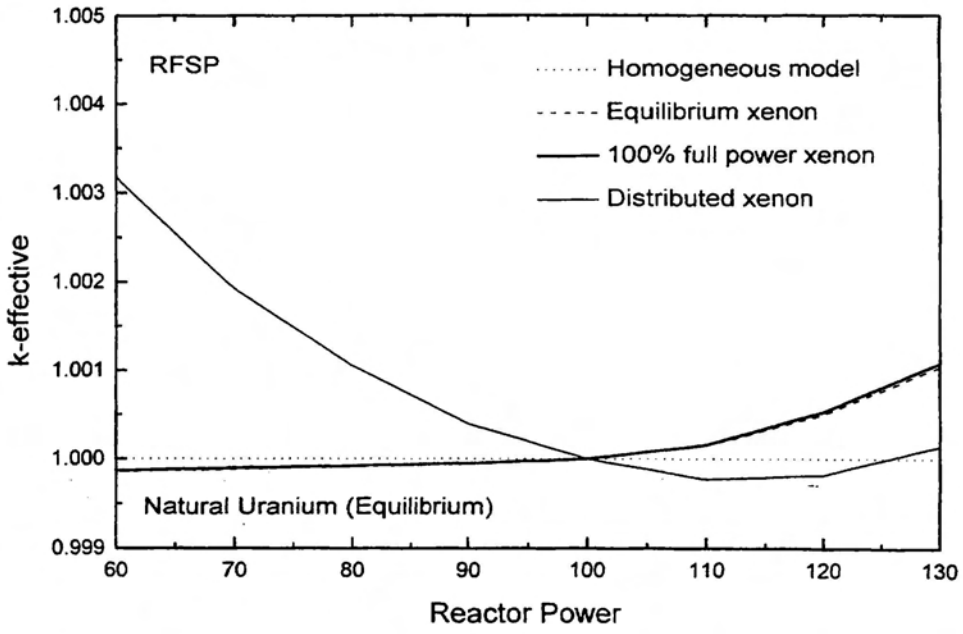

FIG. 1 CORE MULTIPLICATION FACTORS VS. POWER LEVEL (NATURAL URANIUM CORE)

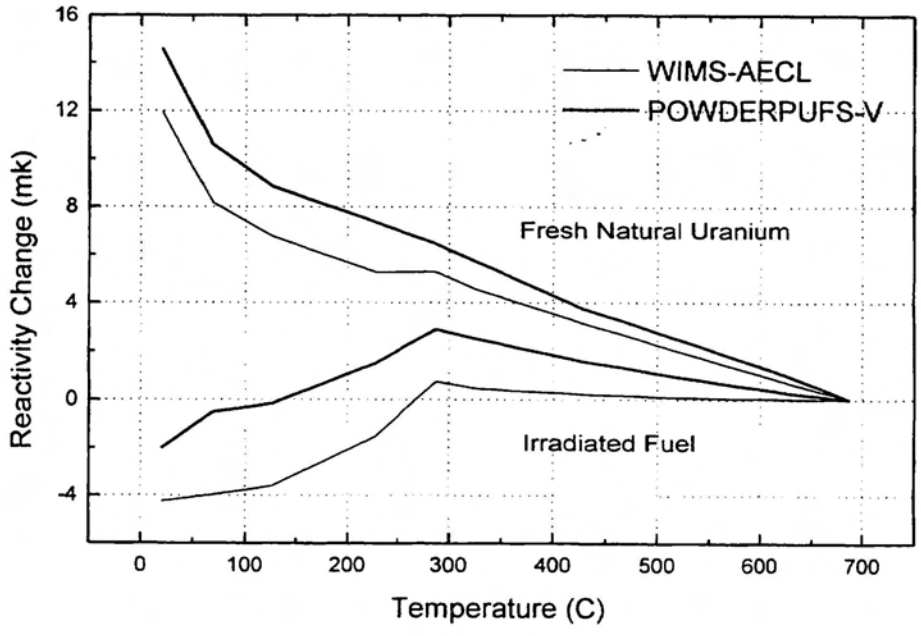

FIG. 2 CANDU LATTICE REACTIVITY VS. SYSTEM TEMPERATURE

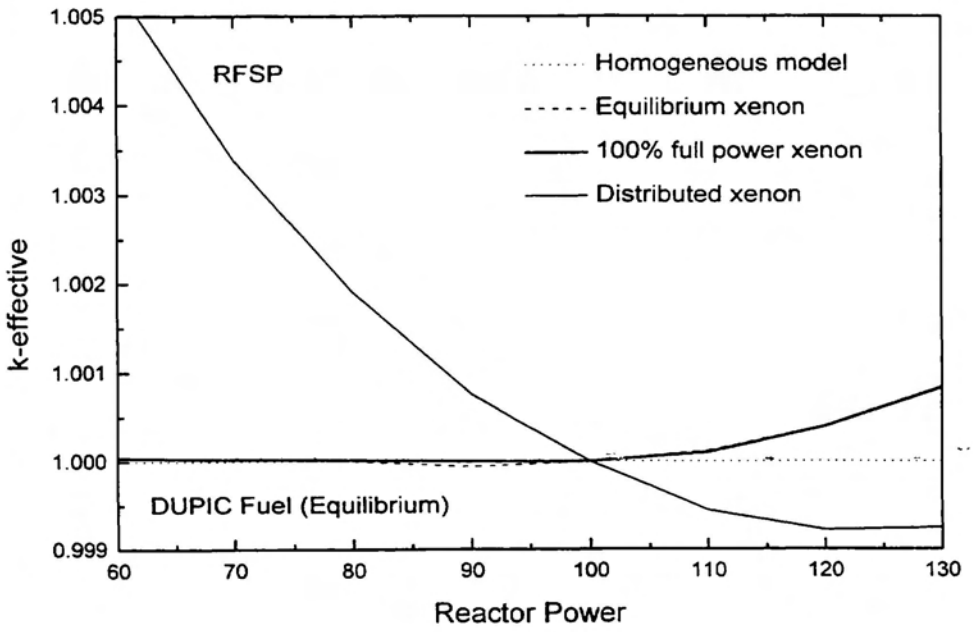

FIG. 3 CORE MULTIPLICATION FACTOR VS. POWER LEVEL (DUPIC FUEL CORE)

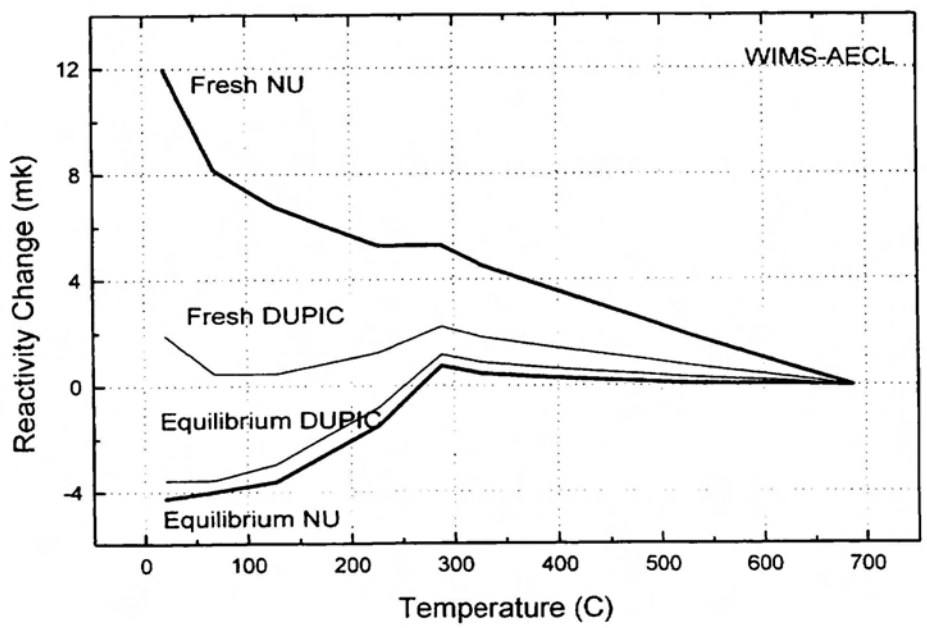

FIG. 4 COMPARISON OF CANDU LATTICE REACTIVITY CHANGES