Coupling of Reactor Physics and Thermal-hydraulics Codes for CANDU Analysis

Benoît Dionne,
Jean Koclas
Peter Tye
dionneb@meca.polymtl.ca, koclas@meca.polymtl.ca, peter@meca.polymtl.ca
Institut de Génie Nucléaire
École Polytechnique de Montréal
C.P. 6079, Succ. Centre-ville
Montréal (Québec), Canada H3C 3A7

ABSTRACT

A coupled thermal-hydraulics and reactor physics simulation procedure was recently developed. It uses ASSERT-PV and the reactor core code chain DRAGON/DONJON/NDF developed at IGN. Parts of these programs where adapted to be run within a coupling procedure written in the GAN generalized driver environment. As a first test, the process was used to simulate one channel (channel N11) in a nominal reactor core similar to a CANDU-6. The coupled results are compared with a regular stand alone reactor physics calculation of the same core.

INTRODUCTION

To improve the analysis of a reactor core, both thermal-hydraulics and neutronics effects must be taken into account. With the substantial increases in the memory and the speed of computers, it has become possible to do such coupled simulations in a reasonable amount of simulation time.
As a first step, the coupling was performed for a single, typical high power channel near the centre of the CANDU-6 reactor core, namely channel N11. To perform these calculations, an iterative loop was used. At first, a static physics calculation was performed with DONJON/NDF, without thermal-hydraulics effects, to produce a bundle average heat flux distribution that is then fed into ASSERT-PV [1]. This program uses a detailed representation of the channel, and a detailed fuel model to calculate the thermal-hydraulics parameters of interest, namely the fuel temperature, the coolant temperature and density. These values are then used to recalculate the macroscopic cross-sections that NDF needs, to complete the cycle.
This procedure is continued until the effective multiplication factor of NDF is converged. A comparison is then made between this coupled simulation and a uncoupled one.

THEORY

A detailed calculation of the power distribution in a reactor core requires an accurate knowledge of the local parameters. For the purpose of modelling the feedback of the thermal-hydraulics effects, a choice of parameters must be made. Four parameters where chosen based on their impact on the power distribution and on their availability in the thermal-hydraulics simulation.

The fuel temperature is responsible for the Doppler effect that widens the resonance absorption peaks of the microscopic cross-sections. This effect changes the absorption and the fission cross-sections.
The coolant is responsible for a small part of the moderation process which tends to slow the fast neutrons just enough to fall into the resonance absorption region. A change in density affects the number of neutrons available for the fast fission of U-238. It also changes the number of fast neutrons that escape from the fuel into the moderator region. As a smaller third effect, the density of the coolant affects the absorption rate of the thermal neutrons. The theory predict that the coolant density will have the greatest impact on the bundle power distribution.
The coolant temperature has an effect on the energy spectrum of the neutrons. Since the coolant temperature is higher than the moderator temperature, it can up-scatter the thermalized neutrons and thus change the effective fission cross-section.
The power distribution is an important parameter since it affects the values of these three previous parameters.
Finally, we must not forget that the extent of these effects depend on the burnup and on the burnup history of the fuel bundles.

Somes studies [2] suggest that the thermal-hydraulics/neutronics coupling is rather weak for a static reactor core. It suggests a decrease of almost 2% in the channel powers near the centre of the core.

CODES USED

The codes set used in this analysis to perform the coupled simulation are grouped in what we call the NEUTHE environment. It is the result of the coupling of the reactor core code chain, DRAGON/DONJON/NDF, with the thermal-hydraulics code, ASSERT-PV. This coupling is carried out using the GAN generalized driver and the CLE-2000 control language. Before going into the details of each step of the control loop illustrated in Fig. 1, a short review of each code is in order.

GAN DRIVER/ CLE-2000

The GAN generalized driver is a kernel that interprets the CLE-2000 control language to perform complex calculations [3]. It is a collection of functions to manage/manipulate data structures and application-dependent modules. It also manages the data exchange between each step of the control loop.

ASSERT-PV

ASSERT-PV is a subchannel code able to model two-phase flows and heat transfer in horizontal channels. It uses the drift flux model and a drift flux correlation to account for transverse flows in CANDU fuel channels. ASSERT-PV uses a detailed representation of the channel including the fuel bundles, the bundle end plates, and an accurate fuel model. In the present case, ASSERT-PV was adapted to accept data provided by the generalized driver as a module that can be called from a procedure written in CLE-2000.

DRAGON/DONJON/NDF

The DRAGON/DONJON/NDF chain of codes are a set of calculation procedures designed to simulate a reactor core [4]. It starts from the solution of the transport equations and the homogenisation process and ends up with the solution of the diffusion equations with multiple strategies and also for many geometry types. Moreover, it has functionalities that enable the user to work easily with sets of macroscopic cross-sections. NDF is used to solve the multi-group static diffusion equations. DONJON is responsible for managing the geometry, the macroscopic cross-sections and the control devices while DRAGON was used to generate a database from a feedback model [5]. The DRAGON/DONJON/NDF reactor core code chain extensively uses the generalized driver and the CLE-2000 control language.

METHODOLOGY

1) ASSERT-PV carries out a thermal-hydraulics calculation on a typical channel with an initial power distribution imported from a neutronics simulation. The 1-D fuel model of ASSERT-PV calculates the fuel temperature at the surface and at the centre of each fuel pin. To calculate the average bundle temperature, the hypothesis of a constant thermal conductivity was used. In this way, we simply average the two temperatures for each pin and then calculate the average bundle temperature. In ASSERT-PV, the discretisation is finer than one value for each bundle used in the neutronic simulation. This implies that the average fluid temperature and density over each bundle must be calculated. ASSERT-PV uses the bundle average enthalpy and pressure drop to approximate the fluid temperature.
   An additional test is performed to make sure that the temperature of the coolant does not exceed the saturation temperature of heavy water under the same pressures and enthalpy. When the calculation is done, the ASSERT-PV module exports the useful information in a data structure.

2) The NDFLINK procedure uses the data exported in step 1 to calculate the new sets of macroscopic cross-sections. In the set of cross-sections, a different fuel region was defined for each fuel bundle of the simulated channel. The properties of these regions take into account the changes in the local parameters. The presence of a liquid zone controller or an adjuster rod and the saturated concentration of Xe-135 for a core at

equilibrium are also taken into account. The feedback model database is queried to modify the set of interpolated nuclear properties that will be used in the next step.

3) A static calculation of the reactor core is carried out with the changes in the properties of the investigated channel. A grid of 26x26x12 is used to discretize the reactor core thus providing one calculation node per fuel bundle. The bundle power distribution of the investigated channel is then extracted and put in an another separate data structure to be used by ASRTLINK.

4) The ASRTLINK module reads the power distribution data structure provided by step 3 and converts it into a heat flux distribution. The module then modified the input file of ASSERT-PV.

5) The whole calculation loop is iterated upon until the effective multiplication factor of the NDF calculation is under a precision of 10^{-6} .

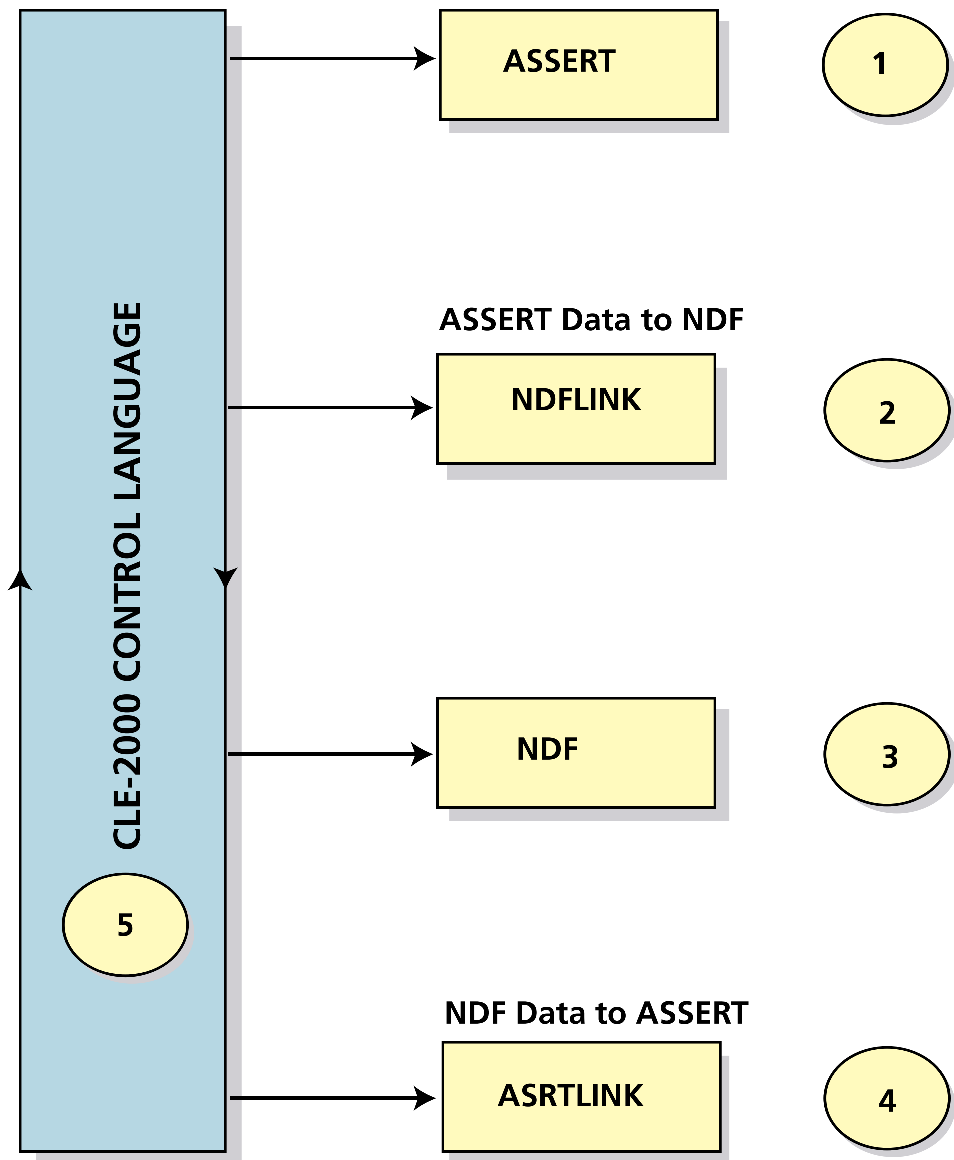

Fig.1 Schematic representation of the control loop with step identifications

APPLICATION

The simulation of a single high power channel will serve as the first application of this procedure. The channel N11 was chosen because it is a typical channel near the centre of the core. In the thermal-hydraulics part of the numerical simulation, a typical eccentric geometry similar to a Bruce channel was used. It includes the representation of the K-factors for the end-plates, the bearings pads and inter-elements spacers, and provides for the fact that some end-plates are miss-aligned. As a reference case, an uncoupled simulation of a reactor core in a nominal state was performed with the characteristics shown in Table 1 and 2. The resulting uncoupled bundle power distribution is shown in Fig. 2 , and this initial bundle power distribution is quite typical for a channel near the centre of the core.

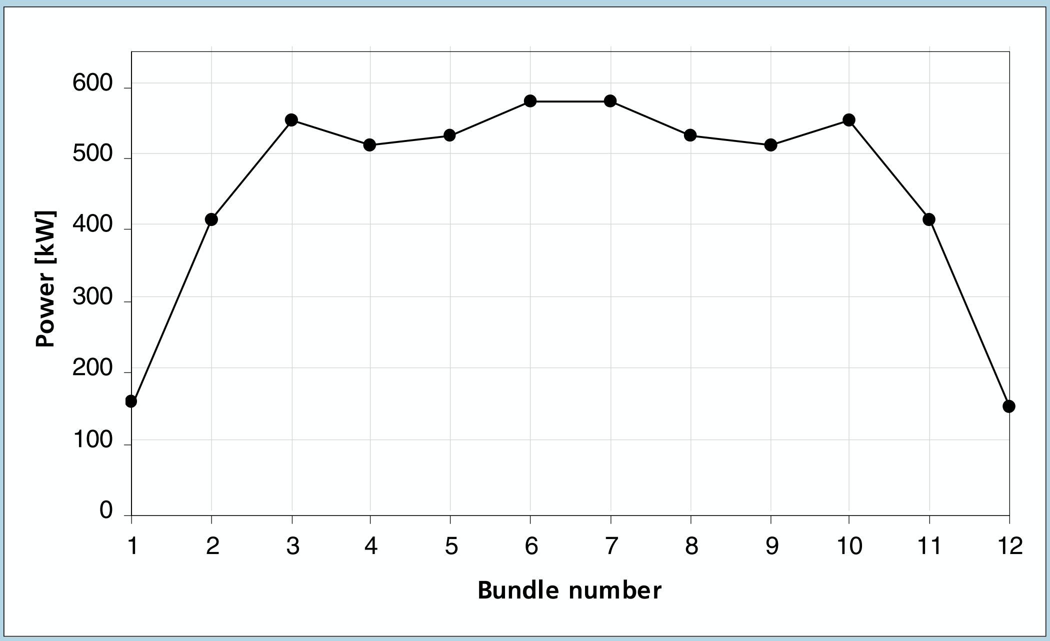

Fig. 2 Uncoupled bundle power distribution in N11 channel

In the following step, this distribution is used to start the coupled simulation of the reactor core in the same nominal state as before. After two iterations (for a simulation time of 308 sec.), the effective multiplication factor is converged to a new value (see Table 3). The local parameters have changed to represent the spatial distribution of the flux (as shown in Fig. 3, 4 and 5) and create a new set of average properties (shown in Table 4). This results in the new bundle power distribution shown in Fig. 6.

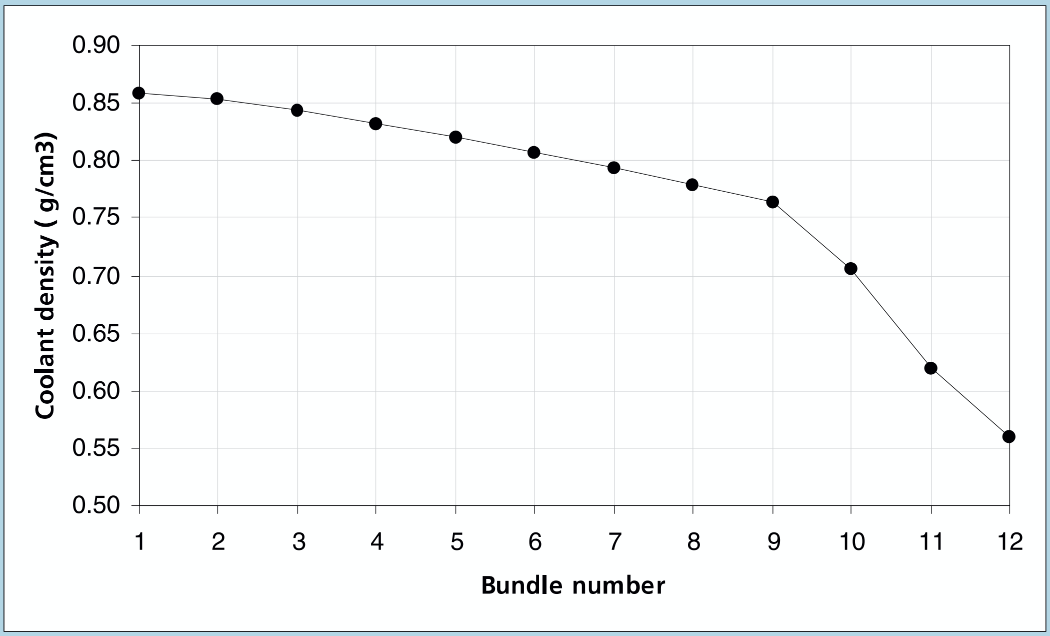

Fig. 3 Coolant density distribution in N11 channel

As it can be seen in Fig. 7, the coupled power is higher than the uncoupled power by about 2.5%. The decrease in the average coolant density is responsible for this effect as predicted by the theory. The spatial distribution of the coolant deviates mostly from the average value used in the uncoupled calculation, at the beginning and at the end of the channel. This is reflected in the higher power difference for the channel extremities. The other effects mentioned in the theory section are not clearly visible in this case because of the large change in the coolant density which dominate the effects on the bundle power distribution. In Fig. 3, we can observe a large decrease in the coolant density for the last three bundles. This can be explained by the boiling as the coolant temperature reaches the saturation temperature (see Fig. 4). The resulting 4.3% quality at the end of the channel is typical for a high power channel.

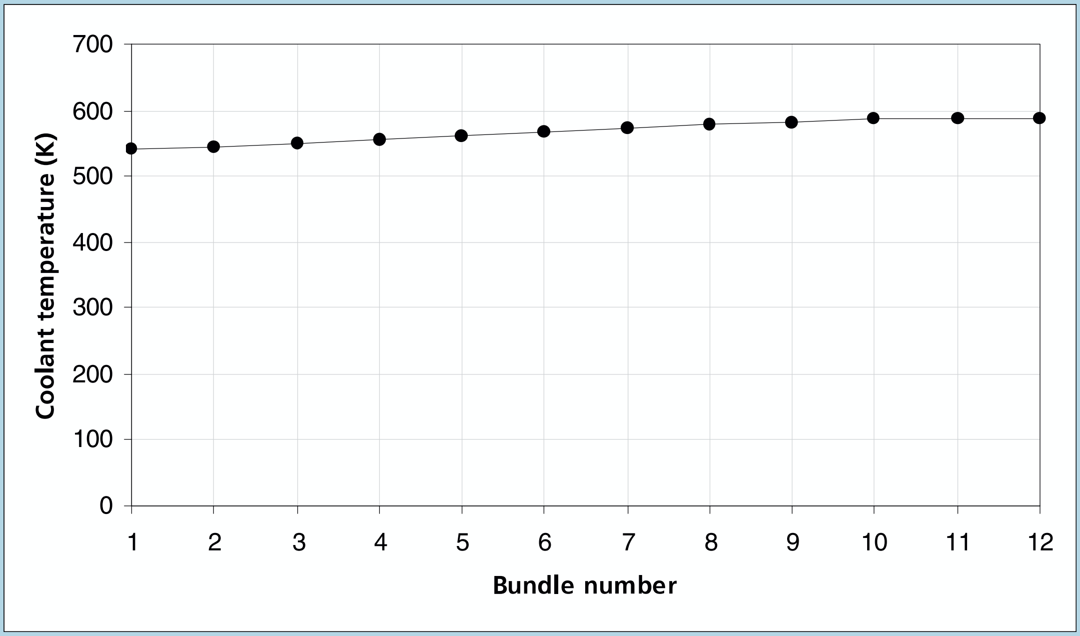

Fig. 4 Coolant temperature distribution in N11 channel

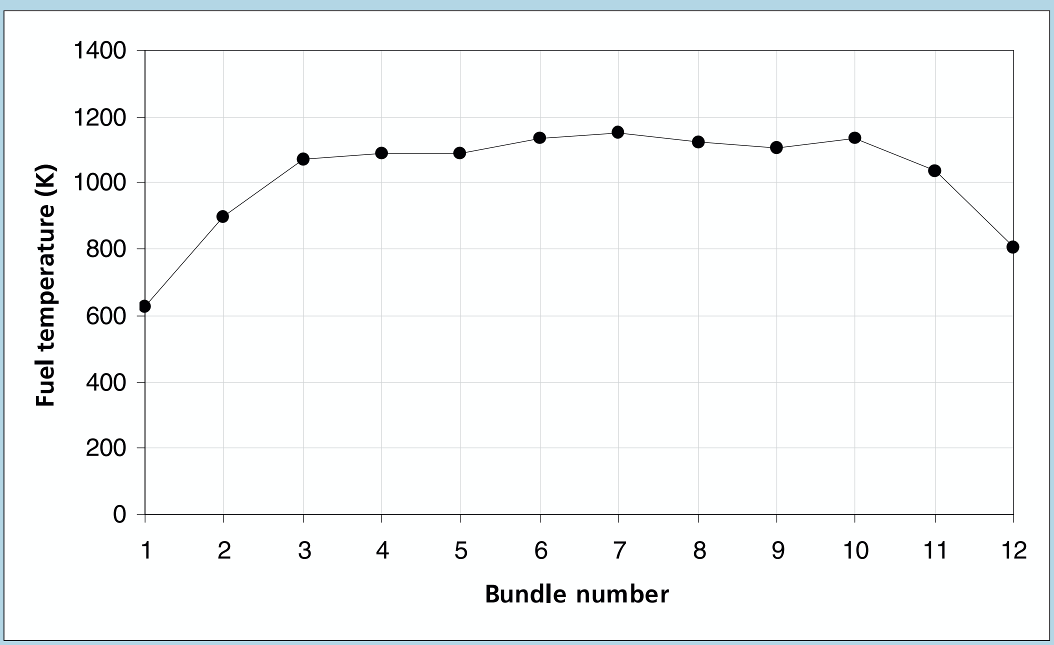

Fig. 5 Fuel temperature distribution in N11 channel

In Fig. 5, the fuel temperature distribution follows the trend of the bundle power distribution. The fuel bundle temperature is higher at the end of the channel than at the beginning. This effect is also typical for a channel near the centre of the core.

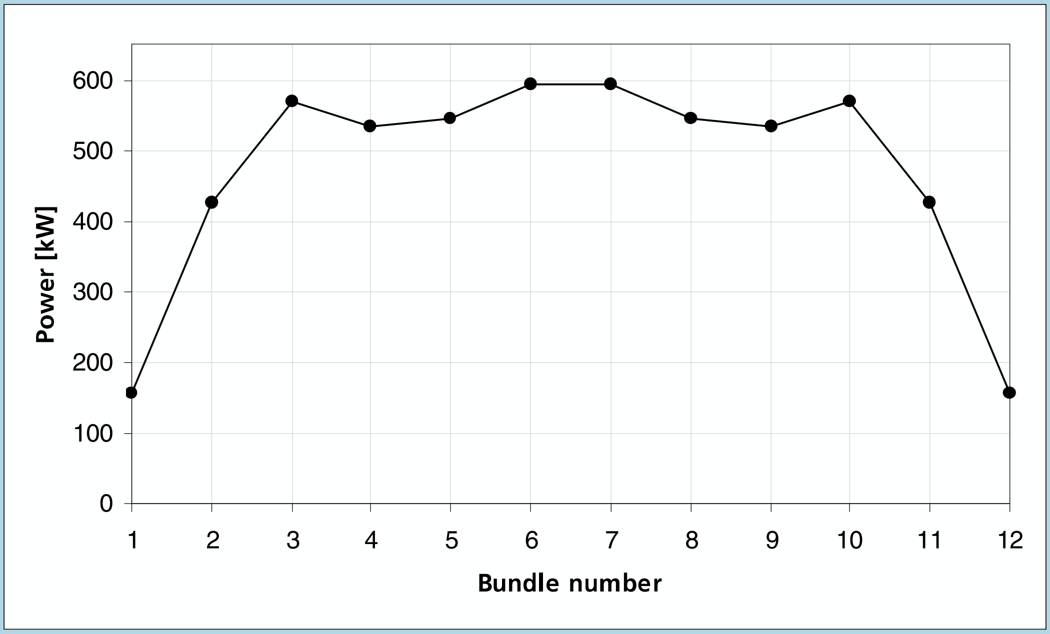

Fig. 6 Coupled bundle power distribution in N11 channel

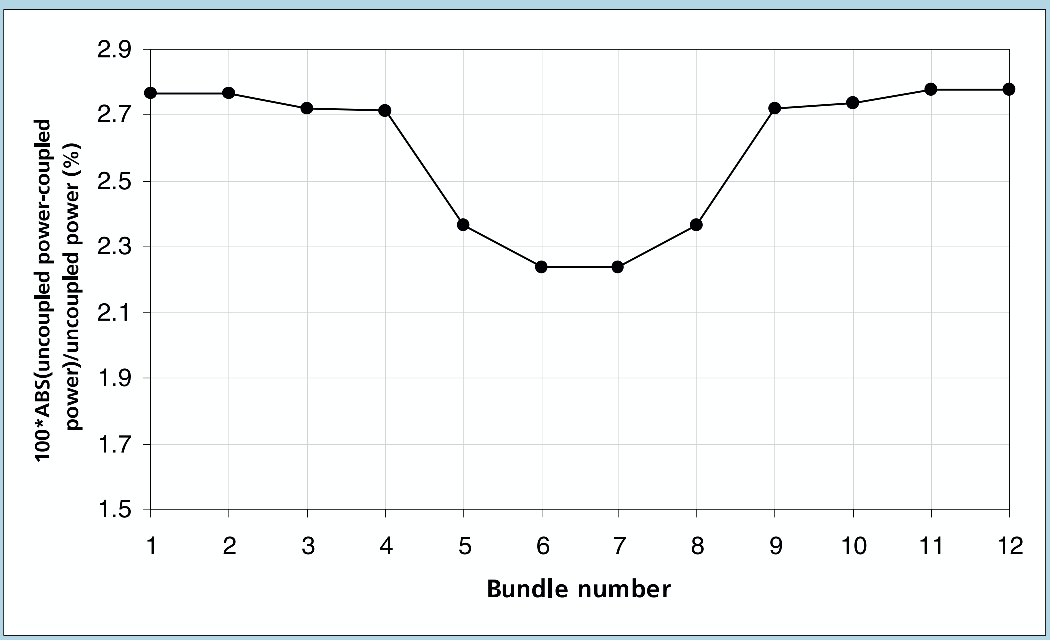

Fig.7 Percentage difference between the coupled and uncoupled simulation

Table 1. Local Parameters of CANDU-6 standard cell used for the uncoupled simulation [6]

Fuel Temperature (°K)	941.3
Coolant Density (g/cm^3)	0.81212
Coolant Temperature (°K)	560.7

Table 2. Nominal properties of the simulated core [7]

Total core power (W)	2.056E+09
Fuel channel inlet enthalpy (kJ/kg)	1132
Fuel channel outlet pressure (Mpa)	10.3
Incoming mass flux ($Mg/s\text{-}m^2$)	5.7
Liquid zone controler (% of full level)	50
Adjuster rod (% of full insertion)	100

Table 3. **Average local parameters resulting from the coupling**

Fuel Temperature (°K)	1022.9
Coolant Density (g/cm^3)	0.76960
Coolant Temperature (°K)	567.5

Table 4. **Effective multiplication factor**

Keff uncoupled	0.99847
Keff coupled	0.99867
mK change	0.2

CONCLUSIONS

The results obtained so far with the current NEUTHE procedure for the coupling of the reactor physics chain DRAGON/DONJON/NDF and the thermal-hydraulics code ASSERT-PV are encouraging. They do not seem to contradict the predicted behaviour obtained from uncoupled calculations.

Our aim is to eventually perform coupled calculations for as many channels as possible with the computer resources available. Experience at this point suggests that about one hundred channels seems to be a reasonable number. This procedure will eventually be extended to do transient calculations. A parallel approach is well suited to such problems, and is also being considered as a future possibility.

ACKNOWLEDGEMENTS

Part of this work was supported by a grant from Natural Sciences and Engineering Research Canada.
ASSERT-PV was made available to École Polytechnique by agreement with the CANDU OWNWERS GROUP, Fuel Channel Critical Power Working Party.

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