USING ENVIRONMENT DEPENDENT CELL PROPERTIES FOR ACR-1000 CORE CALCULATIONS

J-S. Duquette, G. Marleau, R. Chambon

Institut de génie nucléaire, École Polytechnique de Montréal,
C.P. 6079, succ. Centre-ville, Montréal, Québec, Canada, H3C 3A7

ABSTRACT

The Advanced CANDU®[1] Reactor (ACR-1000®) is a generation III+ heavy water moderated and light water cooled reactor. It is designed to meet very strict requirements related to safety and competitive cost. The differences between the CANDU-6 and the ACR include the use of light water coolant, low-enriched uranium fuel and a major reduction in the cell lattice pitch. In fact, reducing of the lattice pitch should have a large impact on the neutrons behavior in the reactor because the cells will be more strongly coupled. The purpose of the article is to evaluate the impact of using environment dependent cell cross sections and leakage coefficient on a full core simulation of the ACR-1000. The DRAGON code will be used to obtain two-group nuclear properties. Those properties will then be used to perform a full core simulation of the ACR-1000 with the DONJON code.

1. INTRODUCTION

The Advanced CANDU Reactor (ACR-1000) is being developed by Atomic Energy of Canada Limited (AECL).[1] Amongst the ACR-1000 design parameters that differ from the CANDU-6, the reduced lattice pitches and the use of light water coolant and enriched fuel are the three most important. The ACR-1000 core contains 520 fuel channels having a 24 cm spacing. The fuel bundle consists of 42 elements of low enriched uranium surrounding a central poison pin without uranium. It is located inside the light water coolant in a pressure tube and a calandria tube, which is surrounded by heavy water moderator. All those features modify the behavior of the neutrons in the ACR compared to the CANDU-6. The impact of the tight lattice is that a cell is more strongly coupled to its neighbor. As a result, the environment in which the cell is placed will affect its behavior and the variation in the two group cell properties should have an impact on a full core behavior. This has required the implementation of a multi-cell capability in the lattice cell code (WIMS) being used for the design of the ACR-1000 at AECL.[2]

This paper describes the impact of the cell environment on the properties of that cell. Two main situations will be exposed. First of all, we will study the effect of the reactor reflector. We will also consider the effect of differential burnup in a heterogeneous multicell pattern. We then obtain, using the lattice code DRAGON[3], the specific properties for that cell. Those properties are used in DONJON[4] for full core simulations.

[1] CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL). ACR® (Advanced CANDU Reactor®) is a trademark of AECL.

In section 2, we will present the models that are used for the transport calculation with DRAGON. In section 3, we analyze the results of the transport calculation. In section 4, we will perform full core simulations with properties generated using the models described in section 2. Finally, in section 5, we will conclude.

2. ACR-1000 MODELS FOR DRAGON

The ACR-1000 cell geometry is composed of a fuel bundle that has 42 fuel pins surrounded by a pressure tube in the calandria tube. The center of the fuel bundle is occupied by a poisoned pin without fuel. As we already said, the lattice pitch is 24 cm.

The first step of a transport calculation consists in selecting the right geometry. For our calculations we will consider the following three geometries:

- Single cell
- Muticell geometry representing the core interior
- Muticell geometry at the core periphery

2.1. SINGLE CELL

The reference properties are obtained from a single cell geometry. White reflection boundary conditions are used along the x axis and white transmission for the y axis. As observed on Figure 1 and Figure 2, the spatial mesh selected is finer for the transport calculation than for the self shielding calculations. The effect of mesh discretization will be discussed later in the paper.

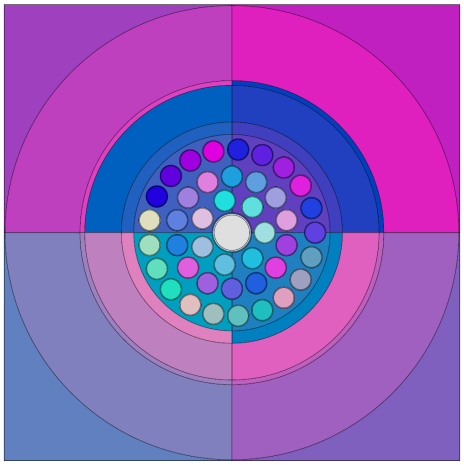

Figure 1 DRAGON single cell geometry for self shielding.

For shelf-shielding calculations, three mixtures are selected. One for each ring except for the central ring for which there is no shelf-shielding calculation (the E6MLIB library selected for

our analysis has no resonance integral tabulations for the isotopes used as poison). Transport calculations with burnup will be considered over a 600 days period at a fixed power of 42.88 kW/kg(U).

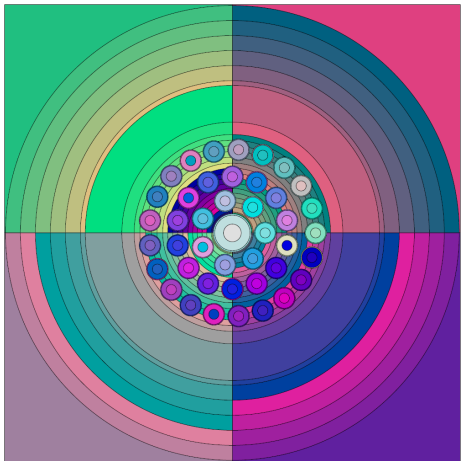

Figure 2 DRAGON single cell geometry for transport calculations.

In case of burnup calculations, each pin of fuel is assumed to burn as an independent mixture.

2.2. MUTICELL GEOMETRY OF THE CORE INTERIOR

The next geometry is used to simulate the evolution of a cell located at the center of the core (see Figure 3 and Figure 4). Again the transport calculations with burnup will be performed for a period of 600 days at fixed power (same as that used for the single cell calculation). The cell from which the two groups homogenized properties will be extracted corresponds to the bottom-middle cell that will be named CenterCell in the text. The choice of mixtures for the shelf-shielding calculations is the same as for the single cell geometry.

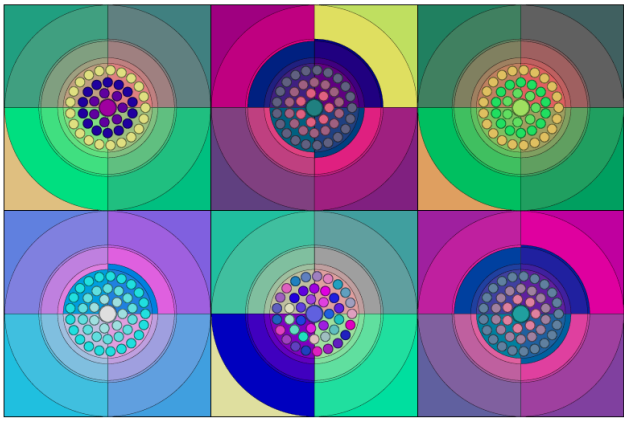

Figure 3 DRAGON core center muticell geometry for self shielding

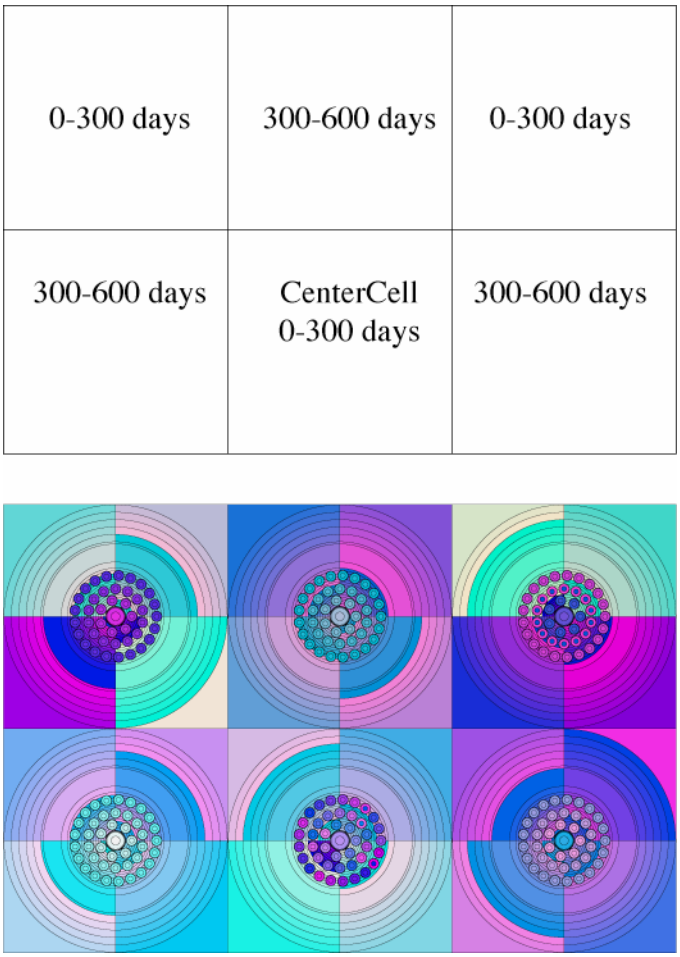

Figure 4 DRAGON core center muticell geometry for transport calculation

To maintain the coherence in our models, each pin of the bottom-middle cell (CenterCell) is also burned independently (see section 2.1). As for the other cells we considered one mixture for each ring of fuel for each cell.

2.3. MUTICELL GEOMETRY OF THE CORE PERIPHERY

Here (see Figure 5), we represent the geometry that will be used to simulate the cell located at the core periphery. Starting from the left, two plates of cadmium are in place to simulate the void condition because the transport calculation using B_1 heterogeneous model in DRAGON is only possible if full reflection or transmission boundary condition are used at the limit of the assembly.[3,5,6] By inserting the plates of cadmium, neutron leakage is simulated by loss of neutron due to absorption. As a result, the diffusion coefficient in the cell containing the fuel (in an heterogeneous leakage model) should be independent of the absorbing material used to simulate leakage.

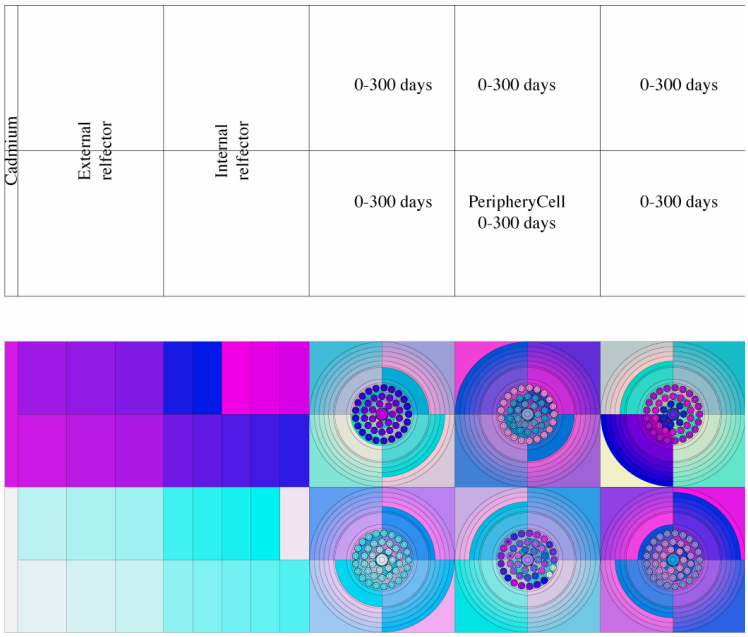

Figure 5 DRAGON periphery multicell geometry for burnup

Following the cadmium, there are four (2X2) 24 per 24 cm reflector regions made up of heavy water. Next to the reflector, the same heterogeneous geometry pattern as that presented in Section 2.2 is reproduced. Once again, the nuclear properties are extracted from the bottom-center cell that contains fuel that will be named PeripheryCell in the text.

3. Transport calculation

The microscopic cross section library we selected is E6MLIB, an 89 groups WIMS-AECL format library.[7] Note that for this library the depletion chains associated with the poison isotopes contained in the central pins are incomplete. In addition, as we mentioned before, no resonance integral tables for the poison isotopes are avaible.[8] It has also been showed that increasing the angular quadrature over a certain value has no impact on k_∞ and accordingly a tracking with $N_\varphi = 9$ and $d = 39$ was considered adequate.

The choice of the spatial mesh for the transport calculation is also very important. We have subdivided in 2 the x and y Cartesian mesh that is superimposed over the annular mesh. The additional uniform volume annular discretization for the flux calculation is summarized in

Table 1 Annular cell discretization. As one can see, the coolant and moderator are discretized using the same factor even if the volume and composition of each region is very different.

Table 1 Annular cell discretization

	Moderator	Coolant	Fuel
Discretization factor	5	5	2

The main modules used for the transport calculation are:

- The self-shielding calculation that are performed using the improved Stamm'ler method programmed in the `SHI:` module of DRAGON.[9]

- The group dependent collision probability matrices are calculated using the `ASM:` module with the option `PIJK` activated to specify that the standard and the directional collision probability matrices are to be computed.

- The resolution of the multigroup collision probability system is performed using the `FLU:` module. The option `B1` is used to specify that the leakage coefficients are calculated using the B_1 model. Also, the option `HETE` is activated so the leakage and anisotropic effects will be taken into account with the TIBÈRE model.[5]

- The isotropic densities and macroscopic cross sections are updated with the module `EVO:`. at fixed power for increasing long period of time.

- The nuclear properties such as the average and condensed cross sections and the reaction rates are computed and then stored with the use of `EDI:` module. For the full core calculations, the reactor cross-section databases are generated with the `CPO:` module.

Even if the center cell assembly is made up with cells that have identical geometries, the burnup is different for each cell (see Figure 4). The fuel in the upper left assembly corresponds to fresh fuel. On the same row, its neighbor is 300 days old. The pattern is repeated for the last cell of the row. In the second row, the left cell is 300 days old. The fuel age then alternates as in the first row.

Table 2 Nuclear properties after 300 days

T=300 days	$\Sigma_{t,1}$ (cm^{-1})	$\Sigma_{t,2}$ (cm^{-1})	$\Sigma_{f,1}$ (cm^{-1})	$\Sigma_{f,2}$ (cm^{-1})	d_1	d_2
Single cell	0,3191	0,4470	0,00111	0,00888	1,54997	1,04772
CenterCell	0,3189	0,4475	0,00110	0,00848	1,55037	1,04237
PeripheryCell	0,3189	0,4472	0,00112	0,00881	1,54805	0,98741

If we take a look at the reference cell after the center cell assembly is burned during 300 days, we then have a reference cell with a age of 300 days. The nuclear properties will be taken over that cell so we will be able to make a consistent comparison with the properties of single cell problem.

For the assembly at the core periphery, the fuel has reached a burnup corresponding to 300 days. The nuclear properties will also be taken from the bottom center fuel cell. We then obtain three sets of properties that we are able to compare.

As we see in Table 2, the variations of the macroscopic total cross section are very small (of the order of 10^{-3}). The macroscopic fission cross section has the largest variation. If we take a look at the second group, we find a variation of 4,5 % between the two assembly calculations.

Table 3 Value of k_∞ over the reference cells

T=300 days	k_∞	Δk_∞(mk)
Single cell	0,951549	
CenterCell	0,931275	-20

On table 3, we obtain k_∞ over the reference cells only. The variation between k_∞ of Single Cell and CenterCell is -20 mk. This can be explain by the fact that the burnup power was imposed on the assemblies. Since the CenterCell (Figure 4) is surrounded by older fuel, the younger fuel of CenterCell will have to produce more neutrons to compensate the lack of neutron produced in the surrounding cells.

It is possible to write the heterogeneous leakage coefficient as a function of the fundamental flux and fundamental current. The heterogeneous leakage coefficient is given by

$$d_{j,g} = \frac{1}{B}\frac{iJ_{j,g}(E)}{\varphi_{j,g}} \tag{1.1}$$

where

$$\varphi_{j,g} = \frac{1}{V_j}\int_{V_j} d^3r\, \varphi_g(\vec{r}) \tag{1.2}$$

and

$$i\frac{J_{j,g}}{B} = \frac{1}{V_j}\int_{V_j} i\frac{J_g}{B} \tag{1.3}$$

The variation between the single cell leakage coefficient and those obtain from the cells extracted from an assembly calculation is under 1%, except for the thermal leakage coefficient of the periphery assembly where the difference reaches 5.7 %. Unfortunately, thermal neutrons are

better absorbed than fast neutron by cadmium.[6] This could explain the fact that the difference for the fast leakage coefficient is small

Table 4 Variation of the leakage coefficient when cadmium and reflector are added

T=0 days	d_1	d_2
CenterCell	1,55037	1,04237
PeripheryCell	1,54777	0,99467
PeripheryCellW	1,55228	0,96351

The next step is to analyze independently the contribution of the moderator and the cadmium on periphery assembly. A transport calculation is performed on a periphery assembly without cadmium (PeripheryCellW). The nuclear properties are extracted from the bottom-center cell that contains fuel.

The variation of the fast diffusion coefficient d_1 is very small, under 0, 3 %, even if the cadmium is taken away (see Table 4). The presence of the moderator decreases the thermal leakage coefficient. In fact, there is a 7.6 % difference between the central core cell (CenterCell) and the periphery assembly cell without cadmium (PeripheryCellW). Adding cadmium to periphery assembly increases the thermal diffusion coefficient (PeripheryCell).

4. ACR-1000 full core calculation

In this section, we proceed to a full core ACR-1000 calculation. First, let's give the main modules that are used to solve the diffusion equation with DONJON[4] :

- The module `TRIVAT:` is used for a TRIVAC-type tracking.
- The module `CRE:` is used to create the MACROLIB containing the nuclear properties.
- The finite element system matrix is computed with the module `TRIVAA:.`
- The solution to the eigenvalue problem is obtained with `FLUD:.`
- The average flux and the axial flux shapes is determined with the use of the module `FLXAXC:.`

The properties are taken from the previous simulation performed with DRAGON. The first full core calculation is performed to see the impact of the reflector on a full core calculation. The geometry definition is presented in Figure 11 of the Appendix. The reflector properties are obtained from the periphery assembly calculation. For the "SingleCellP" calculations, the nuclear properties of all the cells are taken from the DRAGON single cell simulation.

For the "PeriCellP" calculations, the six cells properties near the reflector are extracted from the periphery assembly calculation in DRAGON. When we consider the fact that these cells have evolved in DRAGON and taking to account their location at the core reactor periphery, the result is a diminution of 2.8 mk on the k_{eff} (see **Error! Reference source not found.**). The reactivity devices are at their nominal position. The fuel age is uniform at 300 days. This can explain the low value of k_{eff}.

Table 5 k_{eff} of a periphery full core simulation

T=300 days	k_{eff}
SingleCellP	0,937880
PeriCellP	0,935073

Figure 6 and Figure 7 illustrate the thermal flux in planes #1 and #6 for the SingleCellP and PeriCellP models. We observe that the flux is more localized around the core center for PeriCellP model. Furthermore, the maximal flux amplitude is also larger in this case.

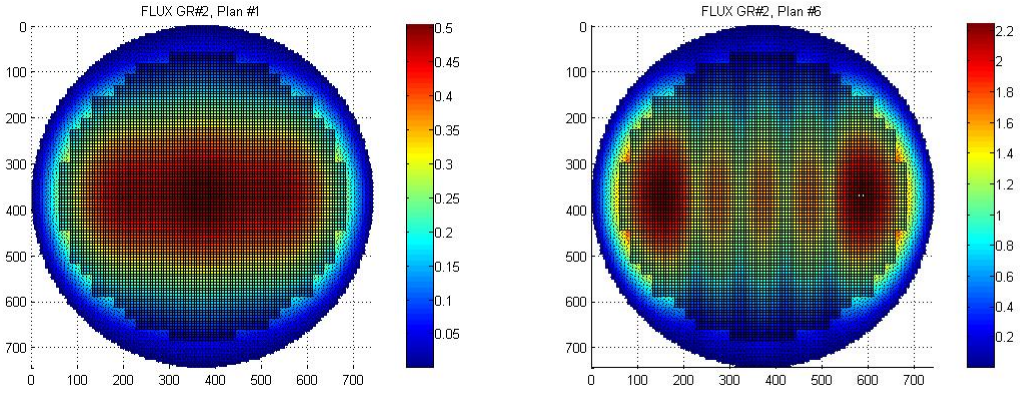

Figure 6 Thermal flux in planes #1 and #6 for SingleCellP model

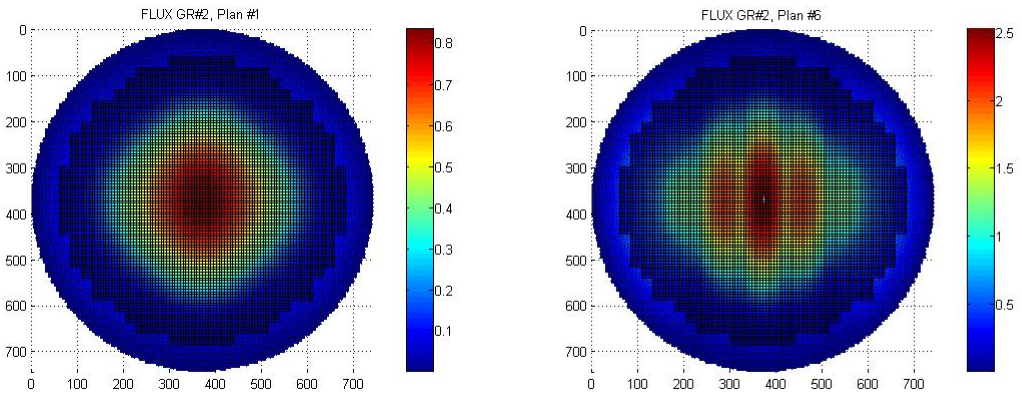

Figure 7 Thermal flux in planes #1 and #6 for PeriCellP model

The second set of full core calculation deals with the environment effect in the center core. The geometry is defined in Figure 12 of the Appendix. The nuclear properties of SingleCellC and CenterCellC models are respectively taken from the DRAGON single cell and central assembly simulations.

Table 5 k_{eff} of a central full core simulation

T=300 &600 days	k_{eff}
SingleCellC	8,59E-01
CenterCellC	8,42E-01

By looking at Table 5, we observe a major drop of k_{eff} when the nuclear properties are extracted from the central assembly DRAGON model. This means that performing an assembly calculation for a cell in an heterogeneous environment (cell with a different age) will have a large impact on a full core calculation. In fact, in that situation, it is a ≈ 17 mk k_{eff} decrease. This large change in k_{eff} can be explained by the fact that the assembly, not each cell within the assembly, is burned at fixed power. As a result, the final burnup of a younger cell will have a tendency to be overestimated (it produces more energy) while that of an older cell is underestimated with respect to the case where a single cell (young or old) is burned at fixed power for the same period of time. Because the two assemblies presented in Table 5 have the same age distribution, a consequence of the above considerations is that the burnup distribution in both assemblies will be different, the more reactive cells (youngest) being effectively more burned in the CenterCellC calculation than in the SingleCellC case. Once again, the reactivity device is fully inserted. The fuel age in the plan x-y alternates from 300 to 600 days. It can explain the low value of k_{eff}.

Figure 8 and Figure 9 present respectively the thermal flux in planes #12 and #6 for SingleCellC and PeriCellC models. We observe that the flux is more localized around the center core for PeriCellP. Once again, when the environment of the cell is taken into account in the transport calculation, the result is a flux maximum around the center.

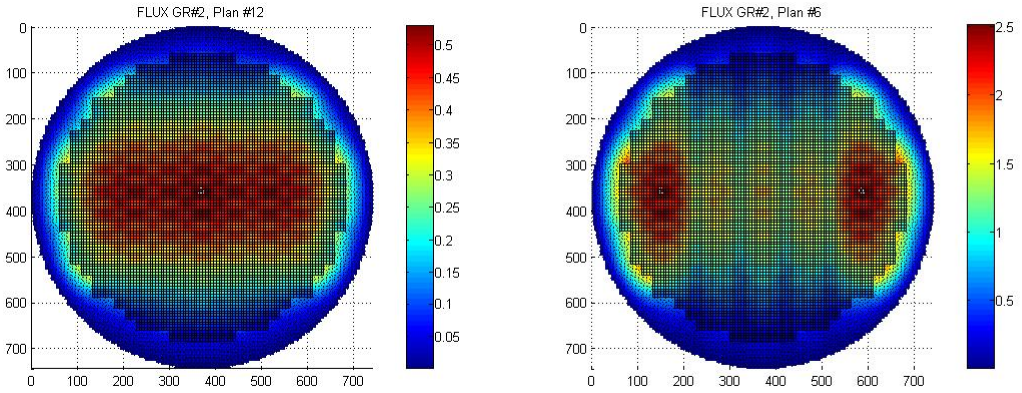

Figure 8 Thermal flux in planes #12 and #6 for SingleCellC model

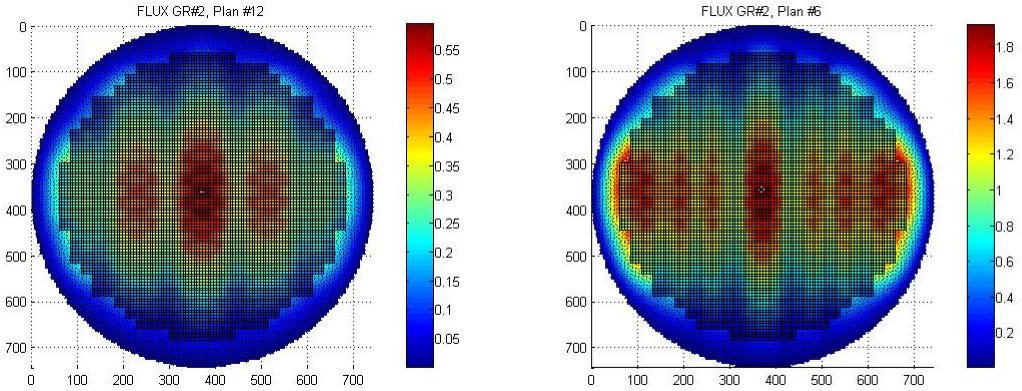

Figure 9 Thermal flux in planes #12 and #6 for CenterCellC model

The fast flux is presented in Figure 10. On the left, the SingleCellC results are presented while one can find the CenterCellC results on the right.

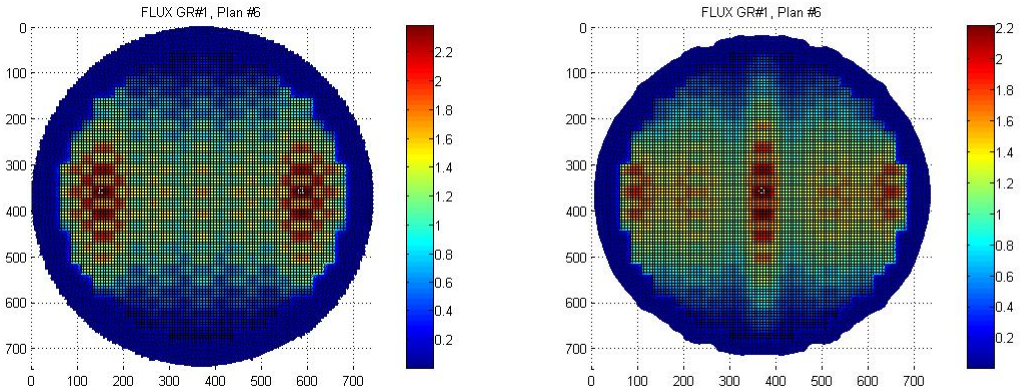

Figure 10 Fast flux in plane #6 for SingleCellC and CenterCellC models

If we compare respectively the thermal and the fast neutron flux for SingleCellC and CenterCellC models, we note that the maximum flux is located in the same area. The fast flux in the reflector is very small as expected.

5. CONCLUSION

The numerical result shows that the highly heterogeneous environment we considered in our calculation seems to have a large effect on a full core calculation. This difference may be attributed in part to the mismatch of local burnup between SingleCell and CenterCell. The transport calculation was executed with the heterogeneous B_1 model. We obtained the smallest leakage coefficient for a cell that was in the muticell peripherical geometry. The presence of moderator tends to reduce dramatically the thermal leakage coefficient. The presence of cadmium slightly increases the thermal leakage coefficient.

The impact of using a periphery assembly to generate the DONJON database from periphery cell is showed. It leads to a reduction of k_{eff} in the full core calculation. In addition, the flux map with the flux was being concentrated around the center for the mixture coming from a periphery cell. Furthermore, the maximum flux amplitude is increased. When DONJON database is created based on center assembly DRAGON calculations, the flux is also more localized around the center core.

ACKOWLEDGEMENT

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Appendix

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Figure 11 Geometry definition for a full core periphery mixture

Mixture identification:

- 6 = Outer reflector
- 5 = Inner reflector
- 4 = Outer periphery fuel
- 3 = Middle periphery fuel
- 2 = Inner periphery fuel
- 1 = Central fuel

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8 8 7 1 1 2 3 3 2 1 1 2 3 3 2 1 1 2 3 3 2 1 1 2 3 3 2 1 1 7 8 8
8 8 7 4 4 5 6 6 5 4 4 5 6 6 5 4 4 5 6 6 5 4 4 5 6 6 5 4 4 7 8 8
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8 8 8 7 4 5 6 6 5 4 4 5 6 6 5 4 4 5 6 6 5 4 4 5 6 6 5 4 7 8 8 8
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8 8 8 7 7 5 6 6 5 4 4 5 6 6 5 4 4 5 6 6 5 4 4 5 6 6 5 7 7 8 8 8
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8 8 8 8 8 7 6 6 5 4 4 5 6 6 5 4 4 5 6 6 5 4 4 5 6 6 7 8 8 8 8 8
8 8 8 8 8 7 7 7 2 1 1 2 3 3 2 1 1 2 3 3 2 1 1 2 7 7 7 8 8 8 8 8
8 8 8 8 8 8 8 7 7 4 4 5 6 6 5 4 4 5 6 6 5 4 4 7 7 8 8 8 8 8 8 8
8 8 8 8 8 8 8 8 7 7 1 2 3 3 2 1 1 2 3 3 2 1 7 7 8 8 8 8 8 8 8 8
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Figure 12 Geometry definition for a full core central mixture

Mixture identification:

- 8 = Outer reflector
- 7 = Inner reflector
- 6 = Bottom-right cell with 600 days old fuel
- 5 = Bottom-center cell 300 days old fuel
- 4 = Bottom-left cell 600 days old fuel
- 3 = Up-right cell 300 days old fuel
- 2 = Up-center cell 600 days old fuel
- 1 = Up-left cell 300 days old fuel