Review Of The CANDU® Time-Average Model And Calculations

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Abstract

This is a review paper. It provides a review of the CANDU time-average model. This is a reactor physicist's design tool, useful in designing an average picture of the reactor operation. It provides to the reactor operator targets of 3-dimensional core flux, power, burnup, and refuelling-frequency distributions to be followed for operating the reactor. The time-average model accounts for the discrete nature of refuelling, the refuelling scheme used, and the variation of lattice properties from bundle to bundle. This paper describes the CANDU time-average model, the analytical basis of the time-average model, and how time-average calculations are done.

1. Introduction

The CANDU time-average model is a computational device used in the design and analysis of a CANDU core. It gives an average picture of the reactor operation, and it provides to the reactor operator targets of 3-dimensional core flux, power, burnup, and refuelling-frequency distributions to be followed for operating the reactor. This paper is a review paper. It describes the CANDU time-average model and explains how time-average calculations are done. The analytical basis of the time-average model is described in detail, as are the degrees of freedom available to the reactor-core designer.

CANDU reactors are refuelled on-line. Refuelling operations are typically performed every day, or a few times per week. A very small fraction of the fuel (typically something of the order of 0.1-0.2%) of the fuel in core is replaced at each refuelling operation. Also, because refuelling is performed on a quasi-continuous basis, the excess reactivity of the core is always very low (except at the very start of the core life, or after a long shutdown).

On account of these features, the overall global power distribution in a CANDU core operated in a well-thought-out manner remains largely constant in time, with relatively small variations ("power ripples") superimposed locally as individual bundles and channels go through their normal burnup cycles from one of their refuellings to the next. Thus, we can think of this relatively constant power distribution as the "time-averaged" core power shape.

2. Direct averaging over time?

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A priori, we might well think that the easiest and most straightforward determination of the relatively constant core power distribution over time is to follow in detail a long period of operating history (actual or simulated), and then simply take the average at each point in the core over time. While this conceptually seems to make a lot of sense, it is in fact not possible *ab initio*, and not only on account of the extremely long time which would be required.

The point here is that to operate a CANDU over a long period of time, or to simulate a period of operating history, it is preferable to have a **target** power distribution as a starting point. This is because while there is a lot of freedom in selecting the sequence of channels to refuel, operating (or simulating) a period of history in a well-coordinated way requires following some basis or target, otherwise it could simply degenerate into a haphazard sequence of disconnected operations and could lead to a less than desirable power shape (e.g., too much or not enough radial flattening).

3. Degrees of freedom in CANDU operation

Let us look at the breadth of freedom in operating a CANDU reactor. The following are "degrees of freedom" available to the operator:

- The fuel type used (e.g., the fuel enrichment or the fuel-bundle geometry, that is 28-element, 37-element, etc.)
- Which channel to refuel on any given day
- Which channel to refuel after the previous one refuelled, and which channel to refuel after that one (i.e., ultimately, the sequence of refuellings)
- The "axial" refuelling scheme to be used (i.e., the number of bundles to change at each channel refuelling), e.g., 8-bundle shift, 4-bundle shift, 2-bundle shift, or other, more complicated scheme. In addition, this axial refuelling scheme may vary from core region to core region, or even in principle from channel to channel. **[Note that refuelling schemes where bundles are reshuffled from one channel to another are not considered in the current discussion.]**
- The refuelling frequency of each channel – the average frequency will depend on the fuel type(s), on the refuelling scheme, and on the overall core-reactivity decay rate.
- The desired discharge (exit) burnup for the fuel, as an overall average and even by core region
- The power desired for each core zone or region.

These degrees of freedom are not all independent of one another: refuelling frequencies, discharge burnup, and region powers are all interrelated. Nonetheless, it is clear that there is a very broad range of possibilities available to the operator, even if a single fuel type is used. Therefore, rational and orderly operation of the reactor really requires following guidelines in terms of the average refuelling frequency of each channel.

4. Time-average model to establish targets

The time-average-model methodology developed for CANDU reactors is designed to calculate the target core power distribution, channel refuelling frequencies, and fuel discharge burnup in the reactor.

Because this model does not involve actual averaging of quantities over a period of time, we will distinguish it from one doing actual averaging by calling it the **"time-average"** model, instead of "time-averaged" model, i.e., without affixing a "d" to "average".

Once the time-average calculation is done, the result can then serve as the target to be used in actual operation. And if the target is indeed followed in operation, then actual averaging over a long period of time can in principle validate the outputs of the time-average model, in terms of average discharge burnup attained, average power distribution attained by region and channel, etc.
The next section will develop the mathematical basis of the time-average model. The process will be illustrated, where required, using the CANDU-6 reactor as an example.

5. Technical basis of the time-average model

5.1 Computer programs

We review here the standard CANDU reactor-physics calculations. These are performed in three stages:

- A lattice or cell code provides the nuclear properties of the basic lattice cell of a CANDU reactor, consisting of fuel, coolant, pressure and calandria tubes and insulating gap between them, and the appropriate amount of moderator surrounding the calandria tube. The lattice code must also be able to follow the evolution of the lattice properties as the fuel irradiation or burnup increases. The lattice code used in the design of the original CANDU reactors was POWDERPUFS-V [1], a semi-empirical code based on measurements in heavy-water-moderated lattices. This has now been replaced by WIMS-IST [2], a code with a firm foundation in neutron-transport theory in 2 dimensions.
- A reactivity-device code provides the "incremental" properties (nuclear cross sections) for reactivity devices, to be added to the basic-lattice properties over an appropriate region around each device. The current Industry Standard reactivity-device code is DRAGON-IST [3], developed at École Polytechnique de Montréal. It can solve the neutron-transport equation in 3 dimensions - important, since CANDU reactivity devices are perpendicular to the fuel channels.
- A finite-core code combines the basic-lattice cross sections and reactivity-device incremental cross sections to calculate the neutronics over the entire CANDU core. RFSP-IST [4] performs full-core calculations by solving the time-independent finite-difference form of the neutron-diffusion equation in 3 spatial dimensions and 2 energy groups:

$$-\vec{\nabla}\bullet \mathrm{D}_1(\vec{\mathrm{r}})\,\vec{\nabla}\phi_1(\vec{\mathrm{r}}) + (\Sigma_{\mathrm{a1}}(\vec{\mathrm{r}}) + \Sigma_{1\to 2}(\vec{\mathrm{r}}))\,\phi_1(\vec{\mathrm{r}}) - \left(\Sigma_{2\to 1}(\vec{r}) + \frac{\nu\Sigma_{f2}(\vec{r})}{k_{eff}}\right)\phi_2(\vec{\mathrm{r}}) - \frac{\nu\Sigma_{\mathrm{f1}}(\vec{r})}{k_{eff}}\phi_1(\vec{r}) = 0$$

$$-\vec{\nabla} \bullet D_2(\vec{r})\,\vec{\nabla}\phi_2(\vec{r}) + \left(\Sigma_{a2}(\vec{r}) + \Sigma_{2\to 1}(\vec{r})\right)\phi_2(\vec{r}) - \Sigma_{1\to 2}(\vec{r})\,\phi_1(\vec{r}) = 0$$

The time-average calculations proper, described in this document, are carried out in the *TIME-AVER module of RFSP-IST. However, as explained above, the basic-lattice cross sections and device incremental cross sections must be available from the lattice and reactivity-device codes.

5.2 Basic idea of the time-average model

The basic idea of the time-average model is to calculate lattice properties, at each bundle location in core, which are averages over the residence time of the fuel at that location. **The proper definition of the average properties is the one which reproduces the number of reactions.** Then these "time-average" properties are used in the neutron diffusion equation to obtain the time-average core distributions of flux, power, burnup, and refuelling frequencies.

Since an "average" picture is desired, the time-average model typically includes reactivity devices in their positions for normal reactor operation: adjusters in core, shutoff rods and mechanical control absorbers out of core, and zone-control-compartment fills at mid-range (typically 50% or so).

5.3 Definitions

Let us start with some notation and some definitions:

- Let c be a label for channel number, c = 1 to M, where M is the total number of channels in core
- Let b be a label for a bundle-position number within a channel, b = 1 to N_c, where N_c is the total number of bundles per channel (12 or 13)
- Then we can use cb as a label for a specific individual bundle position in core
- Let x be a label for a type of nuclear cross section, so, for example:
  - $x = a_1$: Σ_{a1} = fast-absorption cross section,
  - $x = a_2$: Σ_{a2} = thermal-absorption cross section,
  - $x = m$: Σ_m = moderation (down-scattering) cross section, etc.
- Let $\phi_{cb}(t)$ and $\Phi_{cb}(t)$ be respectively the **fuel flux** and the **cell flux** at position cb at some time t. These fluxes are actually multigroup quantities (2-group in the current RFSP-IST). $\phi_{cb}(t)$ and $\Phi_{cb}(t)$ are related by the so-called "F-factor", ratio of flux in fuel to average flux in cell; this is calculated by the lattice code.
- Let $\phi_{tav,cb}$ and $\Phi_{tav,cb}$ be the corresponding **fuel and cell fluxes** in the time-average model. Because the time-average model does not have time as an independent parameter, these fluxes are not functions of time.
- Let D_c be the average time interval between refuellings of channel c; this is also known as the **dwell time** for channel c. [Note: This is not the same as the average residence time of a bundle in core, since in most refuelling schemes some bundles remain in core for more than 1 cycle.]
- Let $\omega_{entr,cb}$ be the fuel irradiation at the time of entrance of the fuel at position cb, and
- Let $\omega_{exit,cb}$ be the fuel irradiation at the time of exit of the fuel from position cb.
- Also, let E_c be the total number of bundles which leave channel c on a refuelling operation; these bundles will be labelled β, β = 1 to E_c. For example, in the 8-bundle-shift (8-bs) refuelling scheme, we have E_c = 8, and the bundles β are bundles 5 to 12 from the channel inlet end.
- Let $\omega_{exit,c}$ be the **channel exit irradiation** of the fuel from channel c, i.e., it is the average of the exit burnup over the bundles β = 1 to E_c which leave the core on a refuelling.

- Let $\Sigma_{x,cb}(\omega)$ be the macroscopic cross section for cell *cb* for reaction type *x*. This is a function of the fuel irradiation ω.
- Then let $\Sigma_{x,cb,tav}$ denote the time-average cell cross section for cell *cb* and reaction type *x*; this will be calculated in the time-average model.

5.4 The selectable quantities

The channel exit-irradiation values $\omega_{exit,c}$ values will be dependent on the dwell times D_c, and vice versa. Both of course will depend also on the fuel-flux distribution $\phi_{tav,cb}$, which depends on the core flux distribution $\Phi_{tav,cb}$, which will be determined as the solution of the neutron diffusion equation. The relationship between the fuel and cell fluxes comes from the lattice (or cell) calculation.

The $\omega_{exit,c}$ or the D_c are the essential degrees of freedom in the time-average model, i.e., they are the values available to be selected by the user as the basic quantities on which the solution of the time-average model rests, as we shall see. In addition, the axial refuelling scheme is also a degree of freedom, for each fuel channel in core.

5.5 Relationships between bundle and channel irradiations and dwell times

For positions *cb* in core where a refuelling introduces fresh fuel, $\omega_{entr,cb} = 0$ [1]

For positions *b* to which a bundle moves from a position *b'* in the same channel, $\omega_{entr,cb} = \omega_{exit,cb'}$ [2]

Now, since $\omega \equiv$ Irradiation = Flux • Time [3]

then Incremental irradiation accumulated by fuel at *cb* between refuellings = $\phi_{tav,cb} \bullet D_c$ [4]

and we can use this to relate the exit irradiation to the entrance irradiation:

$$\omega_{exit,cb} = \omega_{entr,cb} + \phi_{tav,cb} \bullet D_c \quad [5]$$

We can now relate the average channel exit irradiation $\omega_{exit,c}$ to the fuel flux at the various bundle positions in the channel, as follows.

By definition, $$\omega_{exit,c} = \frac{1}{E_c} \sum_{\beta=1}^{E_c} \omega_{exit,c\beta} \quad [6]$$

and, using Equation [3] $$\omega_{exit,c} = \frac{1}{E_c} \sum_{\beta=1}^{E_c} (\omega_{entr,c\beta} + \phi_{tav,c\beta} \bullet D_c) \quad [7]$$

The bundle positions β are those from which bundles leave the channel on refuelling. Some may at the same time be positions into which bundles enter as fresh bundles, e.g., bundle positions 5 to 8 for an 8-bundle-shift refuelling scheme. For these bundles, the term $\omega_{entr,c\beta}$ is zero and does not contribute in Equation [7].

On the other hand, some β may be positions where bundles come from a previous location in the channel, e.g., bundles 9 to 12 in the channel for the 8-bs refuelling scheme. For these bundles, the term $\omega_{entr,c\beta}$ is **not** zero, but can be equated (see Equation [2]) to $\omega_{exit,c\beta'}$, where β' is the position from which the bundle at β came on the previous refuelling. In this way, it can be seen that the sum of terms in ϕ_{tav} in Equation [7] now extends over bundles β **and** β'.

In fact, by repeating the exercise for bundles β' and examining from which position **they** came, we see that eventually all the terms $\omega_{entr,cb}$ disappear and are replaced by terms in ϕ_{tav}, with the sum in Equation [7] covering **all** bundles b in the channel:

$$\omega_{exit,c} = \frac{1}{E_c} \sum_{b=1}^{N_c} (\phi_{tav,cb} \bullet D_c) \qquad [8]$$

Since D_c is independent of the bundle position b, we can take it out of the sum. We can also turn the equation around to isolate D_c on one side:

$$D_c = \frac{E_c \bullet \omega_{exit,c}}{\sum_{b=1}^{N_c} \phi_{tav,cb}} \qquad [9]$$

Equation [9] links the channel dwell time to the channel average exit irradiation, the refuelling scheme (via E_c), and the total fuel flux in the channel.

5.6 Relationships between time-average cross sections and irradiation values

Let us start with the equation for a reaction rate in cell cb (for any reaction type x)

$$\text{Reaction Rate}_{x,cb} = \Sigma_{x,cb}(\omega) \bullet \phi_{cb}(t) \qquad [10]$$

If we focus on one cycle between successive refuellings of channel c, and integrate over time in that cycle, we get the total reactions in cell cb during the cycle:

$$\text{Reactions}_{x,cb} \text{ Per Cycle Between Refuellings} = \int_0^{D_c} \Sigma_{x,cb}(\omega) \bullet \phi_{cb}(t) dt \qquad [11]$$

We define the time-average cross section for cell cb and reaction type x as the cross-section $\Sigma_{x,cb,tav}$, independent of time, which reproduces the total reactions, i.e.,

$$\text{Reactions}_{x,cb} \text{ Per Cycle Between Refuellings} = \Sigma_{x,cb,tav} \int_0^{D_c} \phi_{cb}(t) dt \qquad [12]$$

Equating Equations [11] and [12], we get

$$\Sigma_{x,cb,tav} = \frac{\int_0^{D_c} \Sigma_{x,cb}(\omega) \bullet \phi_{cb}(t) dt}{\int_0^{D_c} \phi_{cb}(t) dt} \qquad [13]$$

and using Eq. [3], we can rewrite this as $\Sigma_{x,cb,tav} = \frac{\int_{\omega_{entr,cb}}^{\omega_{exit,cb}} \Sigma_{x,cb}(\omega) d\omega}{\int_{\omega_{entr,cb}}^{\omega_{exit,cb}} d\omega} = \frac{\int_{\omega_{entr,cb}}^{\omega_{exit,cb}} \Sigma_{x,cb}(\omega) d\omega}{(\omega_{exit,cb} - \omega_{entr,cb})}$ 14]

The integral in Equation [14] can easily be computed using the cell cross section obtained as a function of irradiation with the cell code.

Using the values selected for the entrance and exit irradiations at each position, the $\Sigma_{x,cb,tav}$ can be computed for all positions in core. These nuclear properties can then be entered into the RFSP-IST core model, and the neutron-diffusion equation can be solved by the usual means to find the time-average flux distribution $\Phi_{tav,cb}$.

5.7 Summary of computational scheme

The computational scheme for the time-average model is now complete. It consists of the neutron-diffusion equation plus Equations [1], [2], [5], [9], and [14].

Remember that:

- The flux distribution $\Phi_{tav,cb}$ depends on the time-average cross sections $\Sigma_{x,cb,tav}$ through the diffusion equation;
- The time-average cross sections $\Sigma_{x,cb,tav}$ depend on the entrance and exit irradiation values (Equation [14]);
- The entrance and exit irradiation values $\omega_{entr,cb}$ and $\omega_{exit,cb}$ depend on the fuel flux in the channel (which will be calculated from the time-average flux $\Phi_{tav,cb}$) and the dwell-time D_c (Equations [1], [2] and [5]);
- The dwell-time D_c depends on the flux and on the channel-average exit irradiations $\omega_{exit,c}$ (Equation [9]).

It is clear then that this equation set must be solved as a self-consistency problem. An iterative solution scheme is applied until satisfactory self-consistency is attained.

The basic independent data which needs to be input to the time-average model consists of:

a) the axial refuelling scheme for each channel (for instance, 8-bs, 4-bs, 10-bs, etc.). Note that this may vary from channel to channel, i.e., one may choose 8-bs for some channels and 4-bs for others. Typically, however, the axial refuelling scheme must be selected taking into account also non-physics considerations, e.g., fuelling-machine utilization.
b) the channel-average exit irradiations $\omega_{exit,c}$. These are the main degrees of freedom in the problem. While there are in principle as many degrees of freedom as there are channels, the core is usually divided into a small number of relatively large regions, in each of which ω_{exit} is taken as uniform, to reduce the number of degrees of freedom. Figure 1 illustrates a possible time-average model with 8 regions, for fine control over the target power distribution; note in particular the regions at the bottom and far right-hand side, which

represent areas in the calandria with much hardware (to hold device guide tubes in position) and consequently for which different exit irradiations are needed.

A very important point to note is that there are constraints on the degrees of freedom:

- First, reactor criticality is required. That is, the value obtained for the reactor multiplication constant k_{eff} when solving the diffusion equation must be unity (or close to unity, if we know that the model does not include all material in core, for instance detector material, in which case a small bias is required) – otherwise, the results obtained cannot be expected to represent a realistic time-average picture.
- Secondly, the absolute and relative values of the channel-average exit irradiations $\omega_{exit,c}$ have a significant impact on the radial power distribution, and in turn on the refuelling frequencies and the attained discharge burnup. Increasing the exit irradiation in a region tends to decrease the local reactivity of the region, leading to a lower power. Decreasing the exit irradiation has the opposite effect.
- Thus, the channel exit irradiations need to be selected judiciously to achieve both criticality and the desired degree of radial flattening of the power distribution and of consequent burnup and refuelling rates.

The time-average equation set is solved in the *TIME-AVER module of RFSP-IST. The flow chart for this iterative calculation is shown in Figure 2. **Note: When running the *TIME-AVER module of RFSP-IST for this purpose (we will encounter another purpose later), it is necessary here to set the input parameter IPRESERV to the value 0, otherwise the iterative scheme will not be invoked.**

First, the user selects the axial refuelling scheme(s) and the core regions with the corresponding values of ω_{exit}. Using an initial guess for the flux distribution, the program determines the channel dwell times D_c (Equation [9]) and the entrance and exit values of irradiation $\omega_{entr,cb}$ and $\omega_{exit,cb}$ (Equations [1], [2] and [5]). It then determines the time-average cross sections $\Sigma_{x,cb,tav}$ (Equation [14]). The program then proceeds to solve the neutron diffusion equation. The flux distribution thus obtained is used to repeat the cycle, until consistency is attained in all the parameters.

At that point, the user must intervene (outside the code) and examine the results obtained for k_{eff} and for the power distribution. If k_{eff} is not close to unity, the exit irradiations selected must be adjusted. A standard way of doing this is to **increase or decrease all the $\omega_{exit,c}$ values uniformly, depending on whether the k_{eff} is too high or too low**. Also, the radial power distribution must be examined (this can be done at this time, or after another cycle of iterations with the new $\omega_{exit,c}$ values).

If the radial power distribution is not satisfactory, e.g., if the degree of radial flattening is not the desired one , or if peak channel powers are too high (or too low), or if the rate of refuelling in certain regions is too high, then the regions and/or exit irradiation values must be “manually” adjusted to move the results in the desired direction. Then another cycle of iterations is started.

This process is repeated until a critical reactor (or the required bias) and a satisfactory power distribution are obtained, at which point the time-average model has been established.

5.8 Outputs of the time-average model

The calculations described above provide a number of time-average outputs, which are useful as design outputs and also as targets for the reactor operator. These outputs include:

- The reactor multiplication constant (k_{eff})
- The 3-dimensional time-average neutron-flux distribution. From this, the time-average zone fluxes can be obtained, for use as the target distribution by the Reactor Regulating System.
- The channel and bundle powers. Note that these are the time-average values; they do not include the instantaneous ripple corresponding to the stage in the refuelling cycle. The maximum time-average channel power in a CANDU-6 reactor is typically about 6.7 MW.
- The entrance and exit irradiation values for each bundle position in core.
- The channel **dwell times** (intervals between refuellings of the individual channels). shows dwell times from a CANDU-6 time-average calculation. Dwell times (1-cycle residence time) for bundles in a CANDU-6 reactor typically range between 160 and 360 full-power days (FPD).
- The refuelling rates in the various irradiation regions (actually calculated with the *SUMMARY module).
- The reactivity decay rate in mk/Full-Power Day (actually calculated with the *SUMMARY module). This is useful in gauging the expected variation in zone-control-compartment fill per FPD if there were no refuelling.

Table 1 shows an example of summary results from a CANDU-6 time-average calculation.

5.9 Perturbation calculations with the time-average flux distribution

Once the time-average model has been established, "perturbation" calculations can be performed with it in RFSP-IST. That is, the effect of perturbations acting on the time-average flux shape can be evaluated. The following are examples of such perturbation calculations:

- Reactivity worth of devices (moving a device in or out of the time-average flux shape)
- Reactivity coefficients (derivatives of reactivity with respect to a change in a lattice parameter, such as a temperature, a poison concentration, etc.)
- Coolant-void reactivity (reactivity on complete loss of coolant).

This type of calculation can also be performed in the *TIME-AVER module of RFSP-IST. It is important to note that in this type of calculation, the **pre-perturbation** lattice properties are supposed to be the time-average cross sections, i.e., the $\Sigma_{x,cb,tav}$. While these will be changed in the calculation, one must ensure that this is due simply to the superposition of the assumed perturbation, **not** to any change in the irradiation distributions $\omega_{entr,cb}$ and $\omega_{exit,cb}$. These are not changed, because the reactor is **not** operated steadily with the perturbation being studied. **Therefore, for these perturbation calculations we must bypass the iterative scheme of solution, and preserve the previously calculated $\omega_{entr,cb}$ and $\omega_{exit,cb}$ to compute the $\Sigma^{perturbed}_{x,cb,tav}$,**

in a similar way to [Eq. 14]. The bypassing of the iterative scheme is accomplished in RFSP-IST by setting the input parameter IPRESERV to 1.

Note that in perturbation calculations, bulk and spatial control can be invoked within RFSP-IST to predict the changes in zone-control-compartment fills consequent to the perturbation.

5.10 Relationship between the time-average model and instantaneous snapshots

The time-average model does not represent the reactor on any given day of operation. Rather, it gives an average picture of the reactor operation, in which the channel and bundle powers do not show the ripples due to the refuelling operations and the passage of individual channels through their cycles between refuellings.
In a CANDU core fuelled with natural-uranium fuel, the typical increase in the power of a channel on refuelling is of the order of 20%. Within the 40-50 FPD following this refuelling, the fresh bundles in the channel will go through their "plutonium peak", and the channel power will normally increase during this period, and thereafter decrease monotonically until the next refuelling of the channel. Thus the power of a channel will go through repeated cycles about the time-average value. Typically, the ratio of the instantaneous power of a (natural-uranium-fuelled) channel to its time-average power can cycle between 0.9 and 1.1 or so. These values, however, will depend on the specific channel and on the refuelling scheme. A refuelling scheme with fewer bundles replaced each time (e.g., a 4-bs) will result in a smaller ripple than one with more bundles replaced (e.g., an 8-bs). The maximum value of this ratio in core (or at least in a predefined high-power region) on a given day is called the Channel Power Peaking Factor (CPPF). The CPPF is an important quantity, the value to which the regional-overpower-protection detectors are calibrated; a high CPPF will then "cut into" operating margins. Poor selections of channel refuellings can lead to larger CPPFs.

6. The time-average-equivalent model

As mentioned earlier, the time-average model does not represent the reactor on any given day, i.e., it is not a **snapshot** of the reactor's operating history. This is clear from the fact that to each bundle location in core there correspond 2 irradiation values (the entrance and exit irradiations), not a single value, as in a snapshot.

However, it is often very useful and convenient to do calculations with a snapshot model. To address this need with something akin to the time-average model, a new module (*TAVEQUIV) was created in RFSP. This module defines a snapshot "time-average-equivalent" model, by defining at each bundle location an "effective" instantaneous irradiation, in the sense that the lattice properties at that irradiation reproduce the local k_∞ from the time-average cross sections at that location. The time-average-equivalent model can then be used as any other snapshot model is used.

7. Summary

The CANDU time-average model is a reactor physicist's design tool. It is useful in designing an average picture of the reactor operation, and in providing realistic target values of flux, power, irradiation (burnup) and refuelling-frequency distributions for use by the reactor operator.

The time-average model accounts for the discrete nature of refuelling, the refuelling scheme used, and the variation of lattice properties from bundle to bundle. Once the time-average flux distribution is obtained, perturbation calculations can be performed to compute typical reactivity device worths or other effects.

The time-average model does **not** represent the reactor on any given day. Instantaneous snapshots differ from the time-average model in that they feature the power ripples which are due to the daily refuelling operations and the burnup and power cycles of each individual channel.

The time-average-equivalent model is a snapshot which reproduces the local reactivity from the time-average model at each bundle location.

8. Acknowledgements

The author is grateful to Dave Buss and Cheryl Gaver for their assistance with figures and references. The time-average model was first developed in the 1970s by A. Wight at AECL in the computer program FMDP, precursor of RFSP. A large number of people have contributed over the past three decades to the further development of FMDP, RFSP and RFSP-IST. Although too numerous to list, they are acknowledged here as a group.

9. References

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Figure 1
Possible Time-Average Model with Eight Irradiation Regions

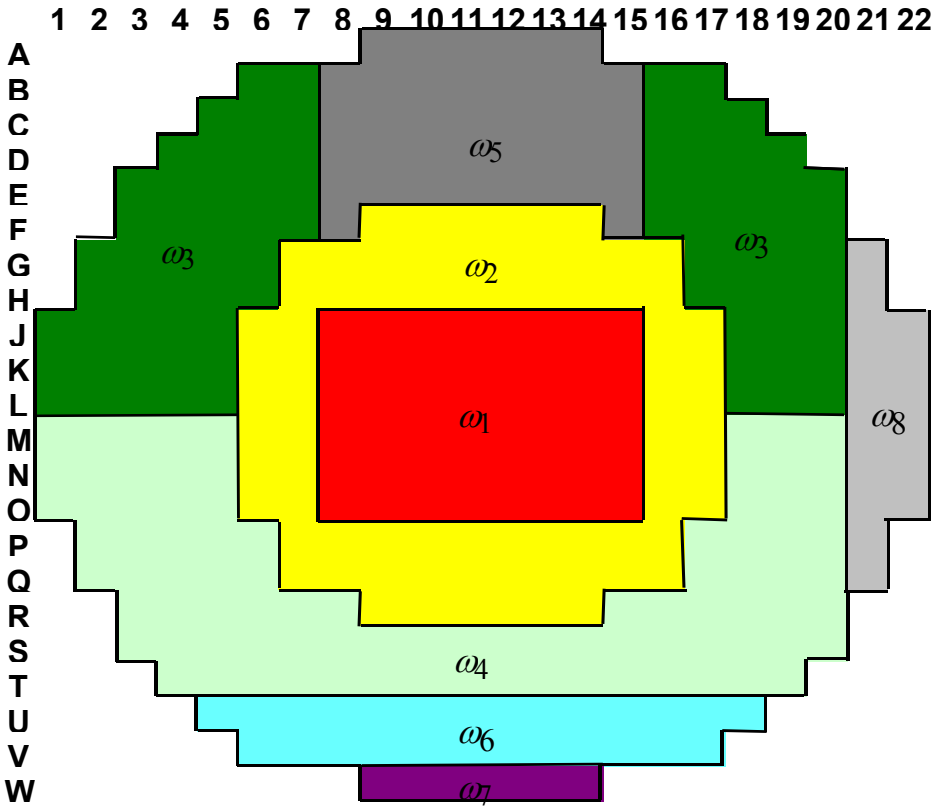
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22
A B C D E F G H J K L M N O P Q R S T U V W
ω5
ω3
ω2
ω3
ω1
ω8
ω4
ω6
ω7

Figure 2

Flow Chart for Time-Average Calculation

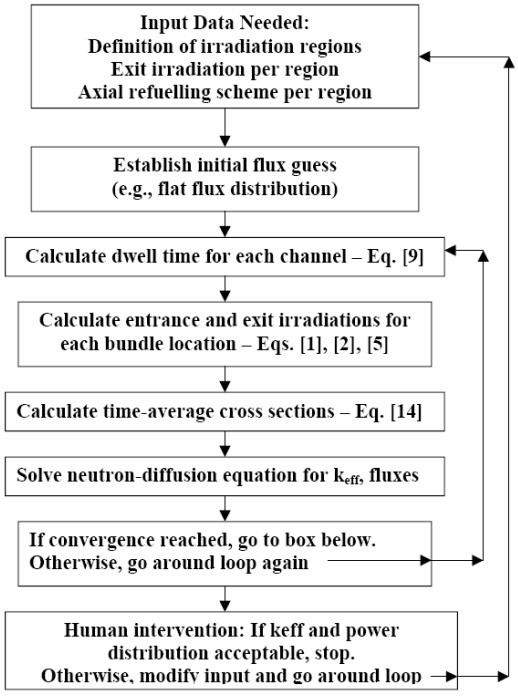

Table 1
Sample Time-Average-Model Results for a CANDU-6 Model

Total Reactor Power (MW)	2061.4
Total Fission Power (MW)	2158.5
Average Channel Power (kW)	5425
Average Bundle Power (kW)	452
Uranium Mass per Bundle (kg)	19.1
Maximum Channel Power (kW)	6604 (N-6)
Maximum Bundle Power (kW)	805 (O-5, Bundle 7)
k_{eff}	1.00250
Average Exit Burnup	3272 MW.h/Bundle 171.3 MW.h/kg(U) 7139 MW.d/Mg(U)
Avg. Channel Dwell Time (FPD)	192
Feed Rate	1.98 channels/FPD 15.83 bundles/FPD
Reactivity Decay Rate (mk/FPD)	-0.385
Average Zone Fill (%)	50.0
Coolant Purity (atom %)	99.0
Moderator Purity (atom %)	99.833