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CANDU Fuel Management

In this learning object we take a look at fuel management in CANDU reactors.

1. Changes in the Nuclear Fuel in a Fission Reactor

In a previous learning object we looked at the isotopic changes which occur in nuclear fuel during its irradiation in the reactor.

2. CANDU Reactor-Physics Computational Scheme

The computational scheme for CANDU neutronics consists of three stages. Computer programs have been developed to perform the calculations corresponding to each stage.

The three stages are:

- Lattice-Cell Calculation
- Reactivity-Device (Supercell) Calculation
- Finite-Core Calculation

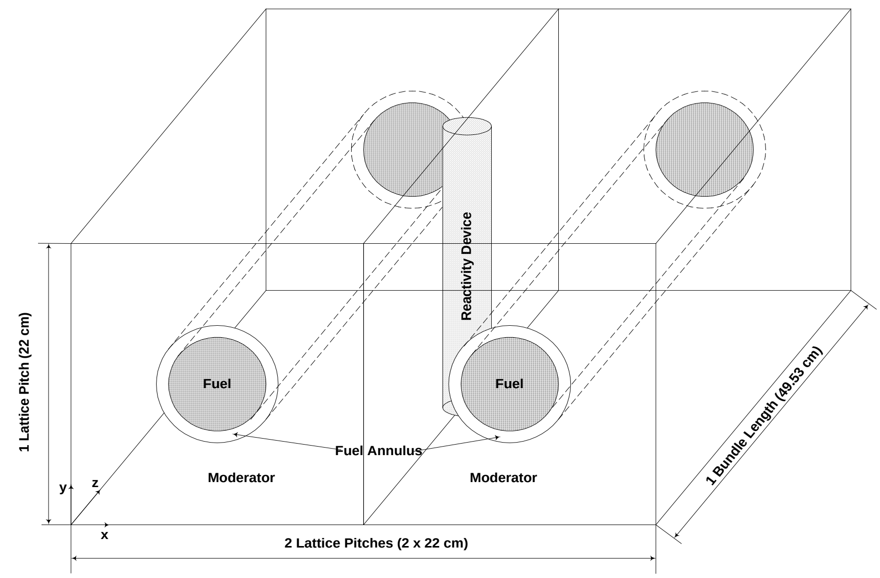
2.1 Lattice Calculation

This first stage involves calculating the average nuclear properties ("homogenized-cell nuclear cross sections" for absorption, moderation, fission, etc.) for the basic lattice cell consisting of the fuel, coolant, pressure and calandria tubes, and moderator, but no reactivity device . We have seen and done lattice calculations in previous sessions.

Quantitatively, the lattice properties govern the neutron-multiplying behaviour of the reactor lattice. The lattice properties must reflect the configuration of the lattice. For example, if the coolant is assumed "voided" in a hypothetical loss of coolant, then the lattice properties must change accordingly.

2.2 Reactivity-Device Calculation

This type of calculation determines the effect of a reactivity device on the nuclear properties in its vicinity. This effect is cast in the form of "incremental" cross sections which are to be added to the homogenized properties of basic cells in a modelled volume around the device, a "supercell" - see Figure 2.1.



Boundary Conditions:

X_- Reflection X_+ Symmetry
Y_- Reflection Y_+ Symmetry
Z_- Reflection Z_+ Symmetry

Figure 2.1 – Supercell for Calculation of Reactivity-Device Incremental Cross Sections

For instance, shutoff-rod incremental cross sections will dictate the local efficacy of the shutoff rods in absorbing neutrons and shutting down the fission chain reaction. The incremental cross sections of a device are obtained by calculating the differences in the supercell homogenized properties between the case when the device is inserted in the supercell, and the case when it is withdrawn from it. These cross sections are then added to the lattice cross sections in the modelled volume around the device in the reactor core.

Because the supercell also contains highly absorbing material, transport theory must again be used. CANDU reactivity devices are perpendicular to the fuel channels. Therefore, realistic supercell models are 3-dimensional, and as a result supercell calculations are best done with the transport code DRAGON, which can do calculations in 3 dimensions.

2.3 Finite-Core Calculation

In this final stage, the entire reactor is modelled. The finite-core computer code must incorporate, for each cell in the reactor, the homogenized basic-lattice cross sections which apply there, together with the incremental cross sections of reactivity devices in their appropriate locations (see Fig. 2.2 below). This model is then used to calculate the 3-dimensional flux and power distribution in the core.

Because homogenized properties are used, diffusion theory can be used to do calculations for the entire core. (Transport-theory calculations for the full core would require a tremendous amount of computer resources.)

The finite-core computer code RFSP-IST is specifically designed for CANDU reactors. It is a very large code (> 1500 subroutines), with many applications and many calculational options. In particular, it can be applied in both steady-state (time-independent) and kinetics (time-dependent) problems, the 3-d flux and power distributions being calculated in two energy groups (1 = fast, 2 = thermal) by solving the corresponding finite-difference diffusion equation.

[Figure credit: CANTEACH Document 20040502, “Introduction to CANDU reactor Physics, by B. Rouben, 2002 September, p.125]

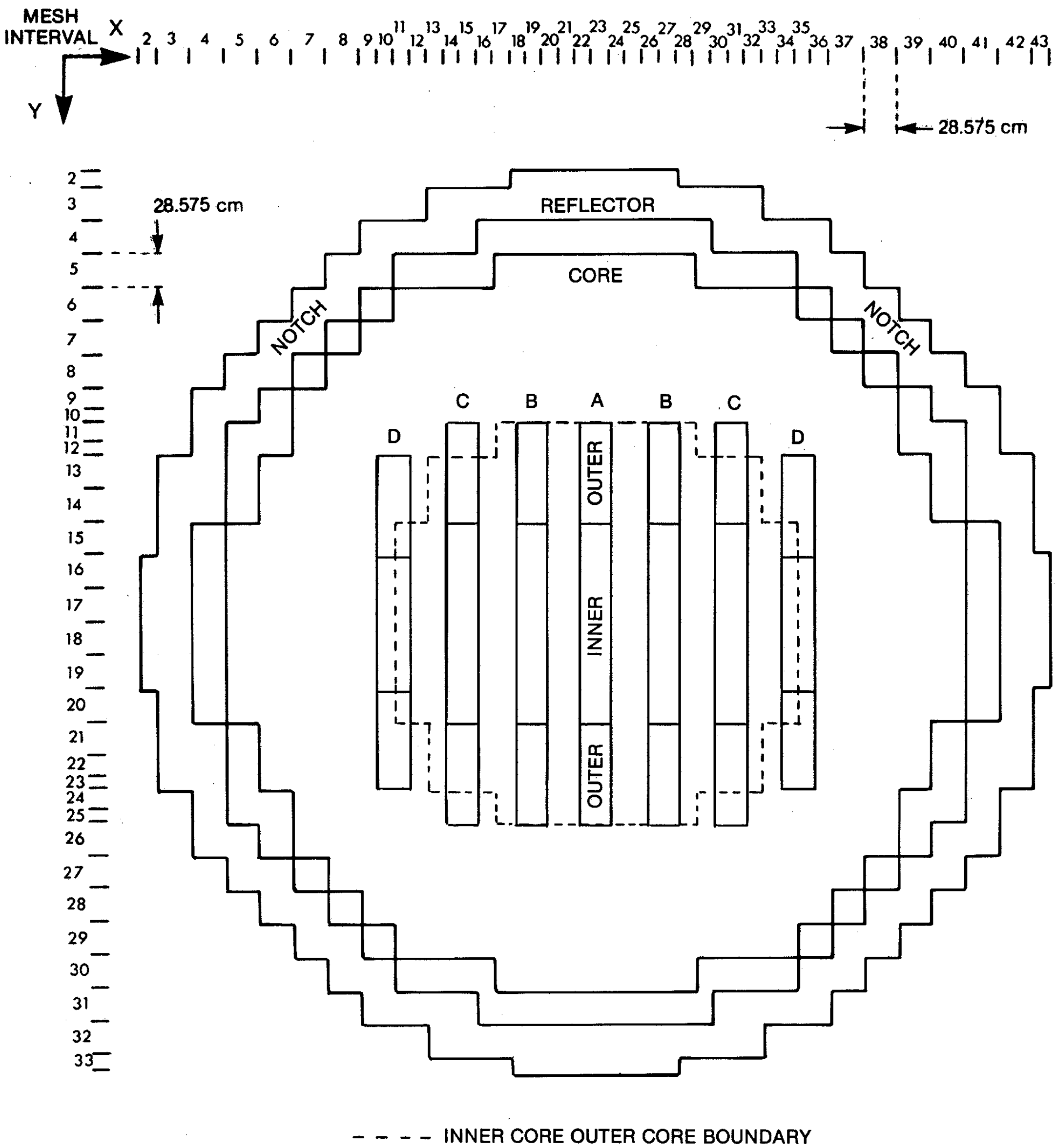


Fig. 2.2 – Face View of Typical RFSP Full-Core Model

3. The Infinite-Lattice Multiplication Constant

3.1 Lattice with Natural-Uranium Fuel

Fig. 3.1 shows k_∞ as a function of irradiation (from PPV) for the classic CANDU-6 lattice fuelled with natural uranium. We have seen this figure before.

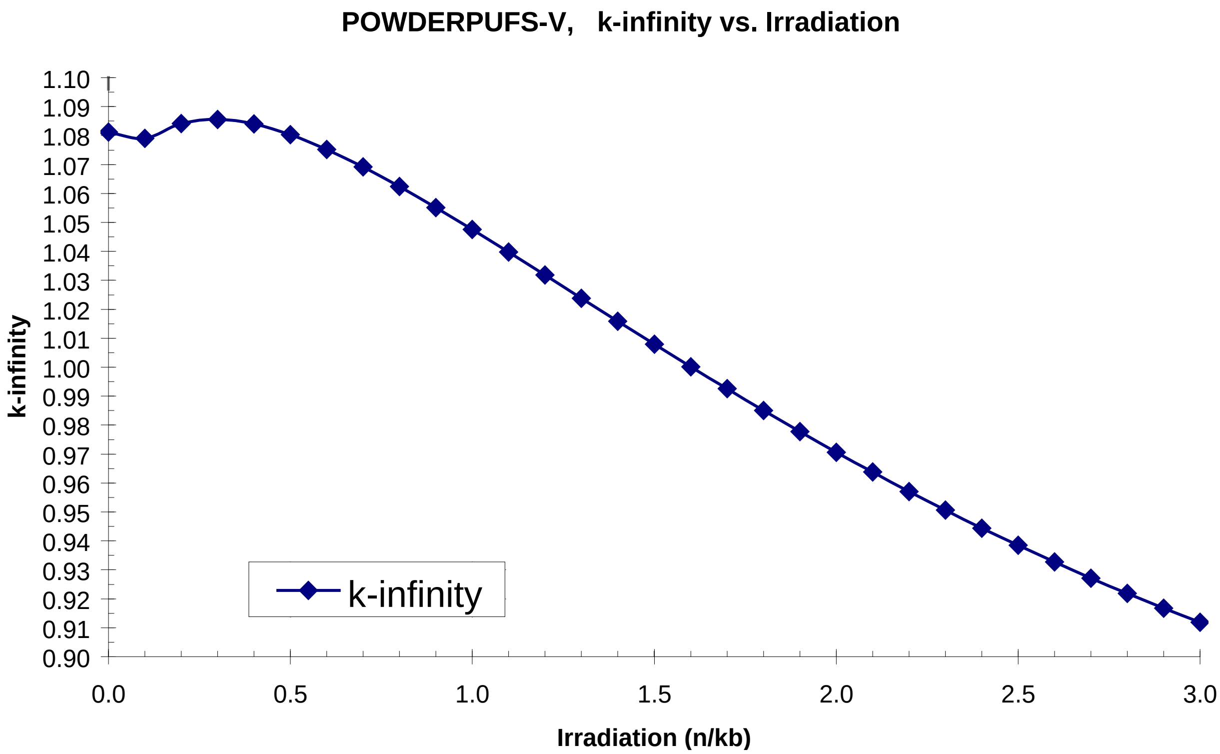


Figure 3.1 k_∞ as a Function of Irradiation

The figure shows that the lattice is ~80 milli-k supercritical for fresh fuel (i.e., at zero irradiation). The reactivity **increases** at first with increasing irradiation, reaching a maximum at approximately 0.4-0.5 n/kb. This phenomenon is due to the production of plutonium, and the fact that the yield cross section from plutonium is slightly higher than from uranium. This reactivity maximum is consequently known as the plutonium peak. (Note that it's the reactivity which is peaking, not the plutonium concentration!)

Beyond the plutonium peak, the reactivity starts to decrease monotonically with increasing irradiation, on account of the continuing depletion of ^{235}U, the fact that the net production rate of plutonium is decreasing, and the mounting fission-product load. , and the lattice reaches zero excess reactivity at an irradiation of about 1.6-1.8 n/kb. This marks a natural point at which the fuel can be targeted for removal from the core, since at higher irradiations the lattice becomes increasingly subcritical, i.e., an increasing net absorber of neutrons. **Thus, channels containing fuel approaching or exceeding these irradiation values become good candidates for refuelling.** This very general statement is made more specific in a later section.

The infinite-lattice multiplication constant k_∞ reaches the "magic" value of 1.048 (corresponding roughly to a critical reactor) at an irradiation of approximately 0.9-1.0 n/kb. Beyond this value of irradiation, the fuel becomes a net absorber of neutrons. It can still remain in the reactor, however, since there is other, younger fuel in the core, on account of on-power refuelling. We will examine this in greater detail in other sessions.

Thus, in fact, fuel need not be discharged from the reactor until it reaches an irradiation such that the k_∞ averaged from 0 to that irradiation is about 1.048. This average "discharge" irradiation is typically 1.7-1.8 n/kb in the CANDU 6. The corresponding average discharge or "exit" burnup is about 170-180 MW.h/kg(U).

3.2 Lattice with Depleted-Uranium Fuel

It is instructive to examine also the infinite-lattice multiplication constant for the depleted-uranium lattice. The infinite-lattice reactivity is shown in Figure 3.2 for depleted uranium with an initial fissile content of 0.52 atom % (as opposed to 0.72 atom % for natural uranium).

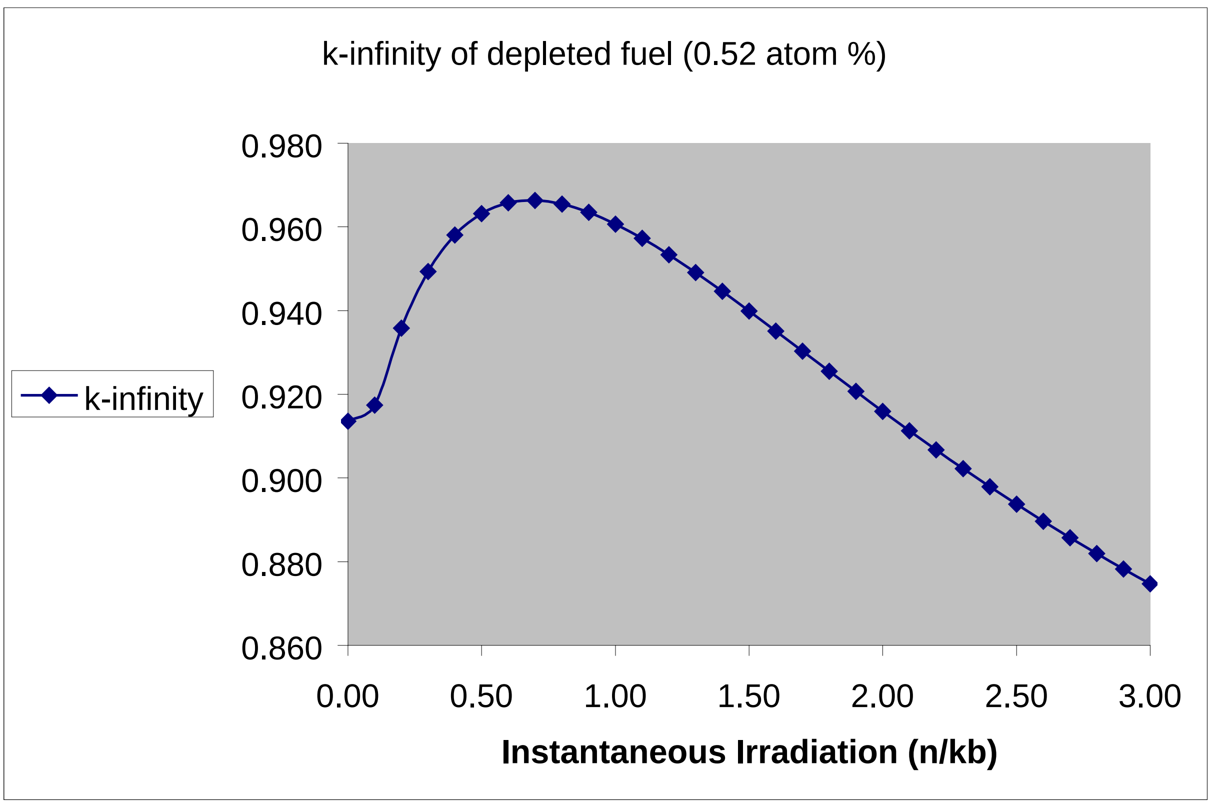


Figure 3.2 - Infinite-Lattice Excess Reactivity of CANDU for Depleted-Uranium Fuel

Note that the plutonium peak is even more pronounced for depleted uranium, a result which is easily explained by the fact that the role of ^{238}U conversion to plutonium is relatively greater when the smaller ^{235}U content. Note also, however, that the depleted-uranium lattice is subcritical at all irradiations, i.e. it is always a neutron absorber. This explains the use of depleted fuel to reduce excess reactivity, and also flatten the flux distribution, in the initial core. Depleted fuel is also occasionally used to reduce the power ripple on refuelling.

4. Equilibrium-Core Design

The equilibrium core contains fuel with a distribution of burnups, and therefore also of irradiations, ranging from zero to discharge values. Ideally, while the nominal ("average-picture") equilibrium core could be analyzed by doing time averages of quantities over simulations of a long period of reactor operation, this is not very practical.

Instead, a "time-average" model is developed to analyze the nominal equilibrium core to a good approximation. This model will be described in detail following the discussion of power flattening.

4.1 Radial Flattening of the Power Distribution

The flux distribution in the reactor depends on the reactor size and geometry and on the distribution of irradiation. Fuel with a high irradiation has low reactivity, and depresses flux in its vicinity. Similarly, the neutron flux tends (everything else being equal) to be high in regions where the fuel has low irradiation. This fact can be used to "shape" the flux (and power) distributions in the equilibrium core.

Radial flux (and power) flattening can be achieved by **differential fuelling**, i.e. taking the fuel to a higher burnup in inner core regions than in outer core regions. This can be done by judicious adjustment of the relative refuelling rates in the different core regions. In this way the flux and power in the outer region can be increased, resulting in a greater number of channels having power close to the maximum value. Thus, a higher total reactor power can be obtained (for a given number of fuel channels) without exceeding the limit on individual channel power. This reduces the capital cost of the reactor per installed kW.

The radial flattening is quantitatively measured by the radial form factor:

$$\text{Radial form factor} = \frac{\text{Average channel power}}{\text{Maximum channel power}}$$

Radial flattening is further assisted by the use of adjuster rods, whose main purpose is in xenon override. Adjuster rods also provide axial power flattening.

Note that while flattening of the power distribution reduces the reactor capital cost, by reducing the number of channels required to produce a given total power, it does tend to increase the neutron leakage, which is proportional to the flux gradient at the edge of the core. This loss of neutrons does have a consequent increase in fuelling cost.

4.2 Time-Average Model

This is a model in which the lattice properties at each bundle location are not determined from an instantaneous or snapshot burnup. Instead, the properties at each location are averaged over the range of irradiations experienced by the fuel during the interval of time it resides at that location between refuellings (called the "dwell" time for that location).

This allows the effect of the actual refuelling scheme used (e.g. 8-bundle shift, 4-bundle shift, etc.) to be captured. Calculations are performed in the *TIME-AVER module of RFSP. The mathematical framework of this module is described in this section, for the specific case of an 8-bundle-shift refuelling scheme as an example – see Figure 4.1. [Figure credit: CANTEACH Document 20054305, "Time-Average Model", by B. Aesenault, 2000, p.19]

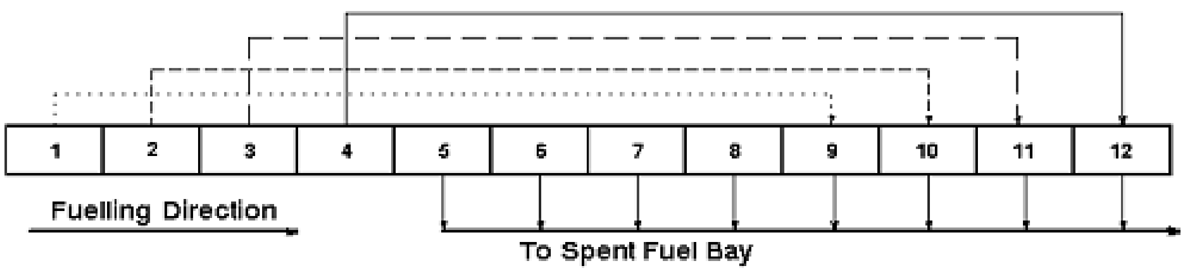


Figure 4.2 – Eight-Bundle-Shift Refuelling Scheme

Time-average nuclear cross sections are defined at each bundle position in core by averaging the lattice cross sections over the irradiation range [ω_{in}, ω_{out}] "experienced" over time by fuel at that position, where ω_{in} is the value of fuel irradiation when the fuel enters that position in core and ω_{out} is the fuel irradiation when the fuel leaves that position. The irradiation range the fuel experiences is then ω_{in} to ω_{out}.

The time-average thermal neutron absorption cross section (for example) at some core position r, $\Sigma_{a2}^{t.a.}(r)$, is then

$$\Sigma_{a2}^{t.a.}(r) = \frac{1}{(\omega_{out} - \omega_{in})} \int_{\omega_{in}}^{\omega_{out}} \Sigma_{a2}(\omega)\, d\omega \qquad (4.1)$$

where the basic lattice cross sections inside the integral sign are determined as functions of irradiation using the lattice code (e.g., POWDERPUFS-V).

Now, let ϕ_{jk} be the time-average fuel flux at axial position k in channel j. Here k ranges from 1 to 12 since there are 12 bundles per channel (in most CANDU reactors), and j ranges over the channels, e.g. from 1 to 380 in the CANDU 6. The bundle position is then labelled jk for short.

Let also T_j be the average time interval between refuellings of channel j (also known as the *dwell time* of channel j).

Then the irradiation increment which the fuel at position jk will experience over its residence time at that position will be

$$\Delta\omega_{jk} = \phi_{jk}.T_j \tag{4.2}$$

If the fuel entered position jk with an irradiation $\omega_{in,jk}$, then its exit irradiation from that position, $\omega_{out,jk}$, is given by

$$\begin{aligned} \omega_{out,jk} &= \omega_{in,jk} + \Delta\omega_{jk} \\ &= \omega_{in,jk} + \phi_{jk}.T_j \end{aligned} \tag{4.3}$$

When a channel is refuelled with an 8-bundle shift, the first 8 positions in the channel receive fresh fuel and the entrance irradiations for positions 9-12 are simply the exit irradiations from positions 1-4 respectively. Thus we can write in this case (see Figure 6.3):

$$\omega_{in,jk} = 0 \qquad k = 1,...,8 \tag{4.4a}$$

$$\omega_{in,jk} = \omega_{out,j(k-8)} \qquad k = 9,...,12 \tag{4.4b}$$

More generally, for an N-bundle-shift refuelling scheme, these equations would become

$$\omega_{in,jk} = 0 \qquad k = 1,...,N \tag{4.5a}$$

$$\omega_{in,jk} = \omega_{out,j(k-N)} \qquad k = (N+1),...,12 \tag{4.5b}$$

In addition to the refuelling scheme, we have other degrees of freedom in the time-average model. These are the values of exit irradiation $\omega_{exit,j}$ for the various channels j, which we can select to our liking. In principle there are as many degrees of freedom as there are channels. (Of course the values of exit irradiation are not totally free, but are collectively constrained by the requirement to obtain a critical reactor.) The *relative* values of $\omega_{exit,j}$ can be used to "shape" the flux to a desired reference distribution.

The relationship between exit irradiations and fluxes can be derived as follows (done here explicitly for the 8-bundle-shift case). Since, in the eight-bundle-shift refuelling scheme, bundles 5 to 12 leave the core at each refuelling, then by definition of exit irradiation

$$\omega_{exit,j} = \frac{1}{8}\sum_{k=5}^{12} \omega_{\mathrm{out,jk}} \tag{4.6}$$

In view of Equation (4.3) this can be written

$$\omega_{exit,j} = \frac{1}{8}\sum_{k=5}^{12}\left(\omega_{\mathrm{in,jk}} + \phi_{\mathrm{jk}}\mathrm{T_j}\right) = \frac{1}{8}\left[\sum_{k=5}^{8}\left(\omega_{\mathrm{in,jk}} + \phi_{\mathrm{jk}}\mathrm{T_j}\right) + \sum_{k=9}^{12}\left(\omega_{\mathrm{in,jk}} + \phi_{\mathrm{jk}}\mathrm{T_j}\right)\right] \tag{4.7}$$

and in view of Equation (4.5) we can write

$$\omega_{exit,j} = \frac{1}{8}\left[\sum_{k=5}^{8}\phi_{\mathrm{jk}}\mathrm{T_j} + \sum_{k=1}^{4}\phi_{\mathrm{jk}}\mathrm{T_j} + \sum_{k=9}^{12}\phi_{\mathrm{jk}}\mathrm{T_j}\right] = \frac{\mathrm{T_j}}{8}\sum_{k=1}^{12}\phi_{\mathrm{jk}} \tag{4.8}$$

It is easy to derive the generalization of this result to an N-bundle-shift refuelling scheme:

$$\omega_{exit,j} = \frac{\mathrm{T_j}}{\mathrm{N}}\sum_{k=1}^{12}\phi_{\mathrm{jk}} \tag{4.9}$$

The dwell time Tj therefore satisfies

$$Tj = \frac{\mathrm{N}\,\omega_{\mathrm{exit,j}}}{\sum_{k=1}^{12}\phi_{\mathrm{jk}}} \tag{4.10}$$

We now have all the equations required for the time-average flux distribution to be calculated. These equations are:

- the finite-difference form of the time-independent neutron diffusion equation to solve for the flux distribution,
- Equation (4.10) to compute the dwell time for each channel,
- Equations (4.3) and (4.4) to calculate $\omega_{in,jk}$ and $\omega_{out,jk}$ for each bundle in core,
- Equation (4.1) (and similar equations for the other cross sections) to calculate the time-average lattice properties.

The degrees of freedom in this time-average problem, which can be chosen by the reactor designer, are the channel-specific axial refuelling scheme - e.g. 8-bundle-shift, 4-bundle-shift, etc…(need not be the same for all channels) - and the channel-specific exit irradiations. The latter, however, must be in a reasonable range of values if a k_{eff} of unity is to be possible. And the variation of the exit irradiations over the core (e.g., inner core vs. peripheral region) will determine the degree of radial flattening of the power distribution. A possible subdivision of the core in irradiation regions for purposes of a time-average calculation is illustrated in Figure 4.2. [Figure credit: CANTEACH Document 20040502, “Introduction to CANDU Reactor Physics”, by B. Rouben, 2002 September, p.138]

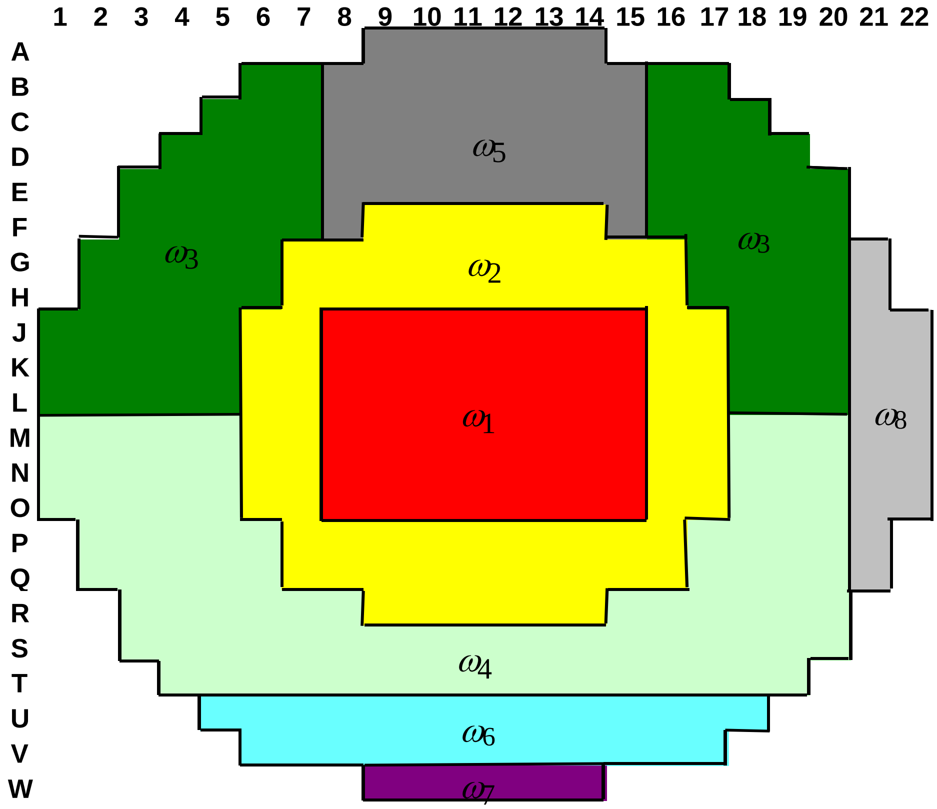


Figure 4.2 A Possible Time-Average Model, With 8 Irradiation Zones

Since consistency must be achieved between the flux, the channel dwell times, the individual-bundle irradiation ranges $[\omega_{in}, \omega_{out}]$, and the lattice properties, an iterative scheme between the solution of the diffusion equation and the other equations is employed until all quantities converge. Figure 4.3 shows the iterative scheme of calculations.

[Figure credit: CANTEACH Document 20040502, “Introduction to CANDU Reactor Physics”, by B. Rouben, 2002 September, p.139]

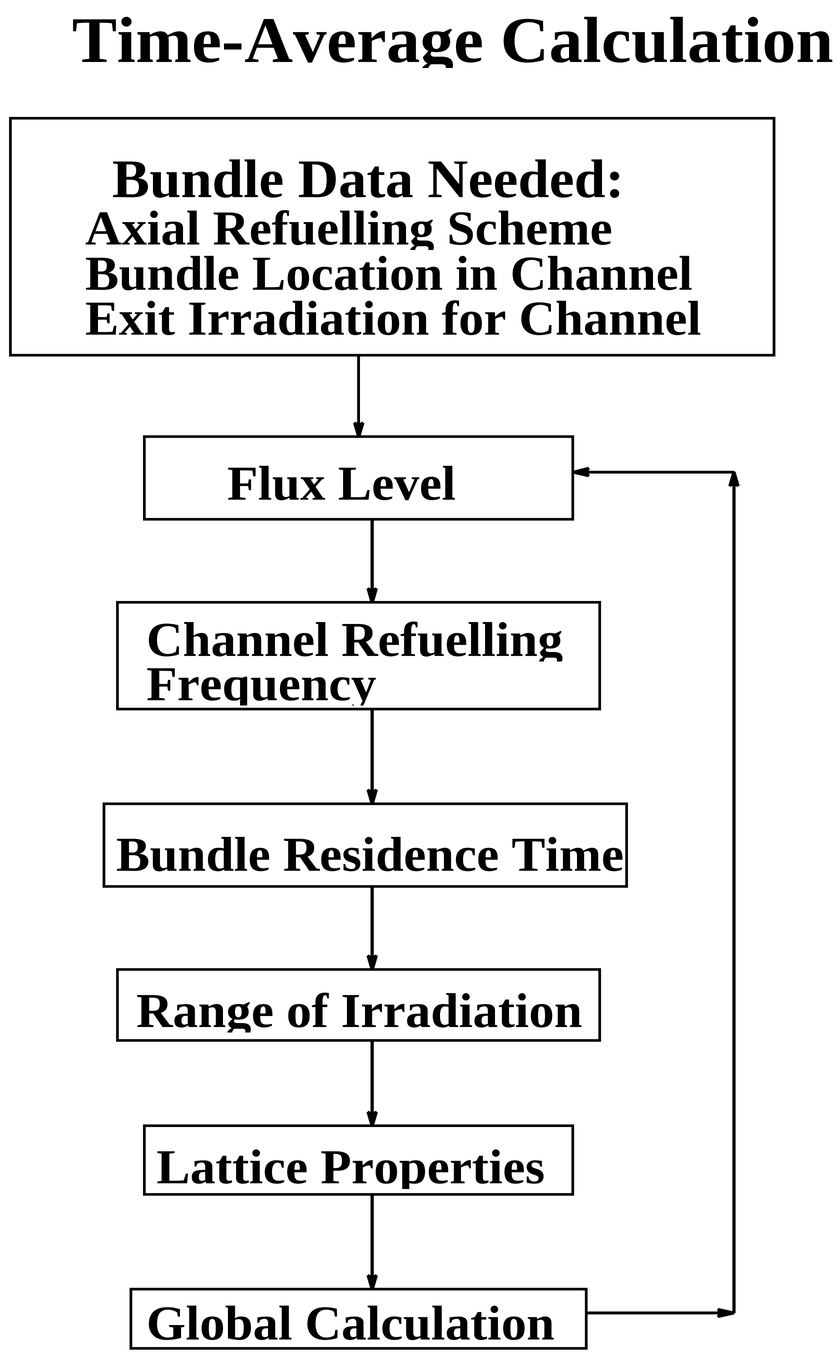


Figure 4.3 Time-Average Self-Consistency Problem

Typically, several “outer” iterations by the human user in selecting values of exit irradiation $\omega_{exit,j}$ in the various regions are required to attain a final solution giving a critical reactor and a desired flux shape (the flux shape is often measured by the degree of flattening, or radial form factor).

The time-average model is useful at the design stage, to determine the reference three-dimensional power distribution, the expected refuelling frequency of each channel (or its inverse, the channel dwell time), and the expected value of discharge burnup for the various channels.

Once the time-average calculation is completed (with a reasonable result), the time-average power distribution becomes the target power distribution for fuel management. Also, the channel dwell times from the time-average calculation (see Figure 4.4) provide a very useful guideline for the time intervals at which specific channels should be refuelled.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									310	300	294	294	300	310								
B						356	300	253	237	227	226	226	227	237	253	300	356					
C					317	270	233	204	193	188	190	190	188	193	204	233	270	317				
D				304	258	222	198	177	171	169	173	173	169	171	177	198	222	258	304			
E			316	257	220	195	179	163	160	160	164	164	160	160	163	178	195	220	257	316		
F			265	222	197	180	169	158	173	174	177	177	174	173	158	169	180	197	222	265		
G		293	236	201	186	174	174	170	172	173	173	173	173	172	170	174	174	186	201	235	286	
H		258	213	187	178	170	172	170	171	172	172	172	172	171	170	172	169	178	187	212	252	
J	319	239	198	178	173	177	173	184	185	187	187	187	186	185	184	173	177	172	177	197	233	311
K	298	224	188	171	169	175	172	185	187	189	192	192	189	187	184	172	175	169	170	187	218	290
L	286	217	183	166	163	170	170	184	187	191	196	196	191	187	184	170	170	162	165	182	210	278
M	286	216	181	164	160	167	168	183	186	191	196	196	190	186	182	168	167	159	163	180	210	278
N	299	223	185	166	160	166	167	181	185	188	192	192	188	185	181	167	165	159	165	184	216	291
O	315	237	195	172	163	167	167	180	183	185	186	186	185	183	179	166	167	163	172	194	230	306
P		259	211	184	173	165	168	167	170	170	170	170	170	169	167	168	165	173	184	211	252	
Q		294	237	203	188	175	174	169	170	170	170	170	170	170	169	173	174	187	203	237	287	
R			274	232	208	188	173	165	173	172	173	173	172	173	165	172	188	208	231	273		
S			339	275	237	207	184	172	167	166	169	169	166	167	172	184	207	237	274	338		
T				336	281	238	208	191	182	179	183	183	179	182	191	208	237	281	336			
U					339	282	239	213	199	193	195	195	193	199	213	239	281	339				
V						350	283	241	221	209	206	206	209	221	241	283	350					
W									290	271	260	260	271	290								

____ Boundaries Of Radial Zones

_ _ _ Boundary Of Inner Burnup Region

Figure 4.4 Channel Dwell Times (in Full-Power Days) from Time-Average Calculation

It can be seen that the dwell times in the inner core range typically between 150 and 160 full-power days (FPD). In the outer core, the dwell times present a large variation, from about 135 FPD for channels just outside the inner core (where the flux is still high but the exit irradiation is, by design, lower than in the inner core) to almost 300 FPD for some channels at the outermost periphery of the core.

5. Ongoing Reactor Operation with Channel Refuellings

After the initial period following first reactor startup, on-power refuelling is the primary means of maintaining a CANDU reactor critical. Thus, a number of channels are refuelled every day, **on the average**. Note that refuelling is not necessarily done **every** calendar day; some stations prefer to concentrate all refuelling operations to 2 or 3 days within each week.

Replacing irradiated fuel with fresh fuel has immediate consequences on the local power distribution and on the subsequent period of operation of the reactor. These must be well understood and are discussed in the following subsections.

5.1 The Channel-Power Cycle

The "refuelling ripple" is the consequence of the daily refuelling of channels and the "irradiation cycle" through which each channel travels. See Figure 5.1.

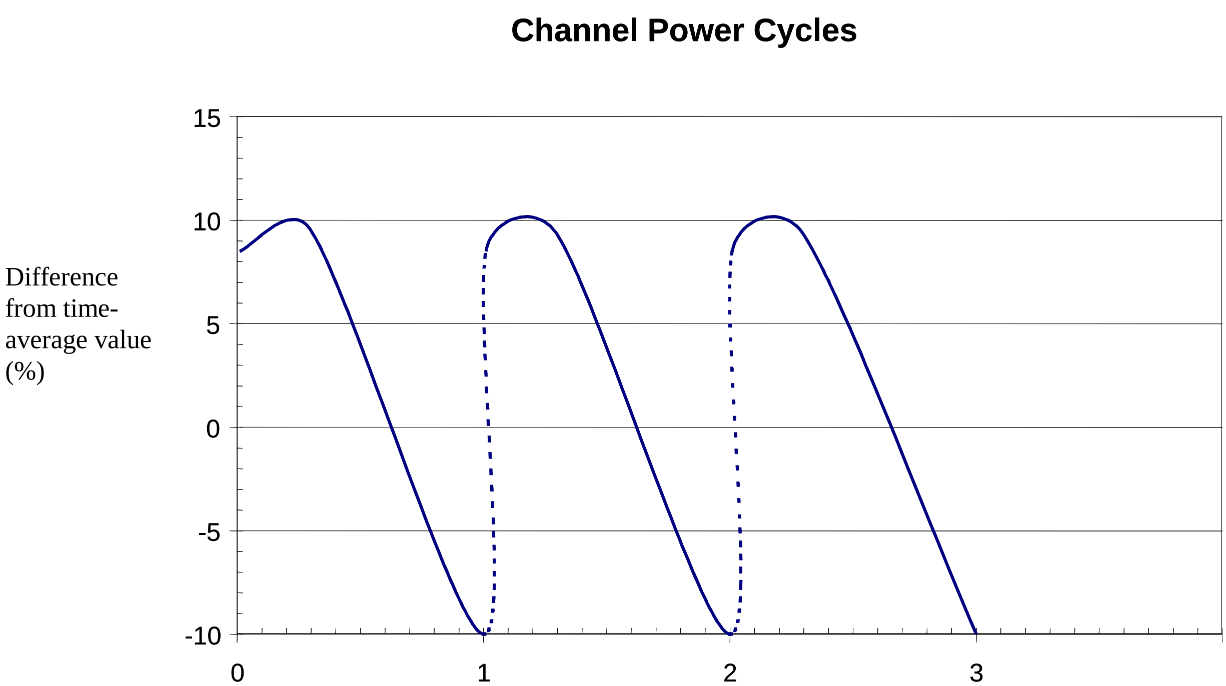


Figure 5.1
Sketch of Power Cycles Experienced by a Fuel Channel

This cycle may be described as follows.

- When a channel is refuelled, its local reactivity is high, and its power will be several percent higher than its time-average power.

- The fresh fuel in the channel then initially goes through its plutonium peak as it picks up irradiation. This means that in fact the local reactivity **increases** for about 40 to 50 FPD, and the power of the channel tends to increase further. The higher local reactivity tends to promote a power increase in the neighbouring channels also.

- Following the plutonium peak, the reactivity of the refuelled channel starts to decrease, and its power drops slowly. Approximately half-way through its dwell time, the power of the channel may be close to the power suggested by the time-average model.

- The reactivity of the channel and its power continue to drop. Eventually, the channel becomes a net “sink” or absorber of neutrons, and nears the time when the channel must be refuelled again. At this time the power of the channel may be 10% or more below its time-average power. When the channel is refuelled, its power may jump by 15 to 20% or even more.

The time-average model does not exhibit refuelling ripple, because there is no “new” or “old” fuel in this model: the properties at all points are averages over the range of irradiation experienced.

In an operating reactor, with channel refuellings, the power of each channel goes through an “oscillation” about the time-average power during every cycle. This cycle repeats every time the channel is refuelled, that is, with a period approximately equal to the dwell time suggested by the time-average model. The cycle length is not **exactly** equal to the dwell time, because channels are not refuelled in a rigorously defined sequence. Instead, as described in the previous section, channels are selected for refuelling based on instantaneous, daily information about the core power and irradiation distributions. In addition, the CANDU fuelling engineer has much flexibility in deciding how the core should be managed, and in fact can decide to modify the global power distribution by changing the refuelling frequency (dwell time) of various channels.

As individual channels are refuelled and go through their channel-power cycle, the specific sequence of these discrete refuellings results in variability in the instantaneous peak channel and bundle powers in the core.

5.2 Channel-Power Peaking Factor

At any given time, there are several channels in the core which are at or near the maximum power in their cycle. Therefore, the maximum <u>instantaneous</u> channel power is always higher than the maximum <u>time-average</u> channel power, since the time-average model does not include the refuelling ripple.

Because many safety analyses are normally carried out in a time-average model, it is very important to quantify how much higher the instantaneous power distribution peaks above the time-average distribution. The Channel-Power Peaking Factor (CPPF) is defined to capture this concept:

$$CPPF = \underset{m}{Max}\left[\frac{CP_{instantaneous}(m)}{CP_{time-average}(m)}\right] \tag{5.1}$$

where m runs over all channels in the core, or at least over all channels except perhaps the channels with the very lowest power, i.e., except the last two outermost rings of channels.

The CPPF value varies from day to day, as the various channels which have fairly recently been refuelled go through their cycle. However, the average CPPF value must obviously depend on the axial refuelling scheme used. The greater the number of bundles replaced at each operation, the greater the reactivity increment, and therefore the greater the refuelling ripple (and therefore the CPPF). When the 8-bundle-shift refuelling is used, a typical value for the CPPF is in the range 1.08-1.10. With a 4-bundle-shift scheme, the typical CPPF is likely to be 1.04-1.05.

The exact value of the CPPF is extremely important because it is used to calibrate the in-core ROP detectors. The hundreds of flux shapes that are used in the ROP safety analysis (to determine the detector positions and setpoints) are all calculated in the time-average model, assuming many different core configurations. But because the real instantaneous channel powers are higher than the time-average powers used in the ROP analysis, channels would reach their “critical channel power” (power at which there is fuel dryout) earlier than in the time-average model. To take this into consideration and ensure proper safety coverage in the instantaneous power shape, the in-core ROP detectors are calibrated each day to the instantaneous value of CPPF.

In order to maximize the margin to trip, it is obviously important that the CPPF be kept as low as possible. This is why a careful selection of channels to be refuelled needs to be made always. A way in which CPPF can be kept low by design is by using, say, 4-bundle-shift refuelling instead of 8-bundle-shift refuelling, or using a mixed 4- and 8-bundle-shift scheme, where the 4-bundle shifting is done in the inner core (high-power region).

Another way in which poor refuelling strategy could impact on reactor operation is as follows. Concentrated refuelling in the vicinity of an ROP detector will increase its reading, even though this may not increase the CPPF in the core. The high detector reading may lead either to spurious trips or to power deratings (to restore operating margin), both of which lead to loss of power production.

Determining the daily CPPF value, and ensuring detectors are calibrated to the correct value, are on-going duties of the fuelling engineer or reactor physicist at a CANDU nuclear generating station.

5.3 Criteria for Selecting Channels for Refuelling

One of the main functions of the fuel engineer (or site reactor physicist) is to establish a list of channels to be refuelled during the following period (few days) of operation. To achieve this, the current status of the reactor core is determined from computer simulations of reactor operation, the on-line flux mapping system, the ROP and RRS in-core detectors, and zone-control-compartment water fills. The computer simulations of reactor operation provide the instantaneous 3-dimensional flux, power and burnup distributions.

Exercise

Imagine that it is your responsibility to select a set of channels for refuelling in the next few days. List as many principles or rules which you would use to set up a list of channels for refuelling.

5.4 Initial Fuel Load and Transient to Onset of Refuelling

Let us now return to the period which marks the beginning of reactor operation.

In the initial core, fresh fuel is present throughout the core. There is no differential burnup which can assist in flattening the power distribution. Consequently, the power of the central core region would be unacceptably high if no alternate means of flattening the radial power distribution were provided. However, an alternate means **is** readily available: depleted fuel. As we have seen earlier, this depleted fuel is a net absorber of neutrons.

In the initial core of the CANDU 6 (i.e., the initial fuel load), two depleted-fuel bundles (of 0.52 atom percent ^{235}U content) are placed in each of the central 80 fuel channels. This is shown in Figure 5.3. The bundles are located in positions 8 and 9, where the numbering is from the channel refuelling end. In these axial positions, the depleted-fuel bundles are removed from the core in the first refuelling visit of each of these channels.

cont'd

[Figure credit: CANTEACH Document 20040502, "Introduction to CANDU Reactor Physics", by B. Rouben, 2002 September, p.161]

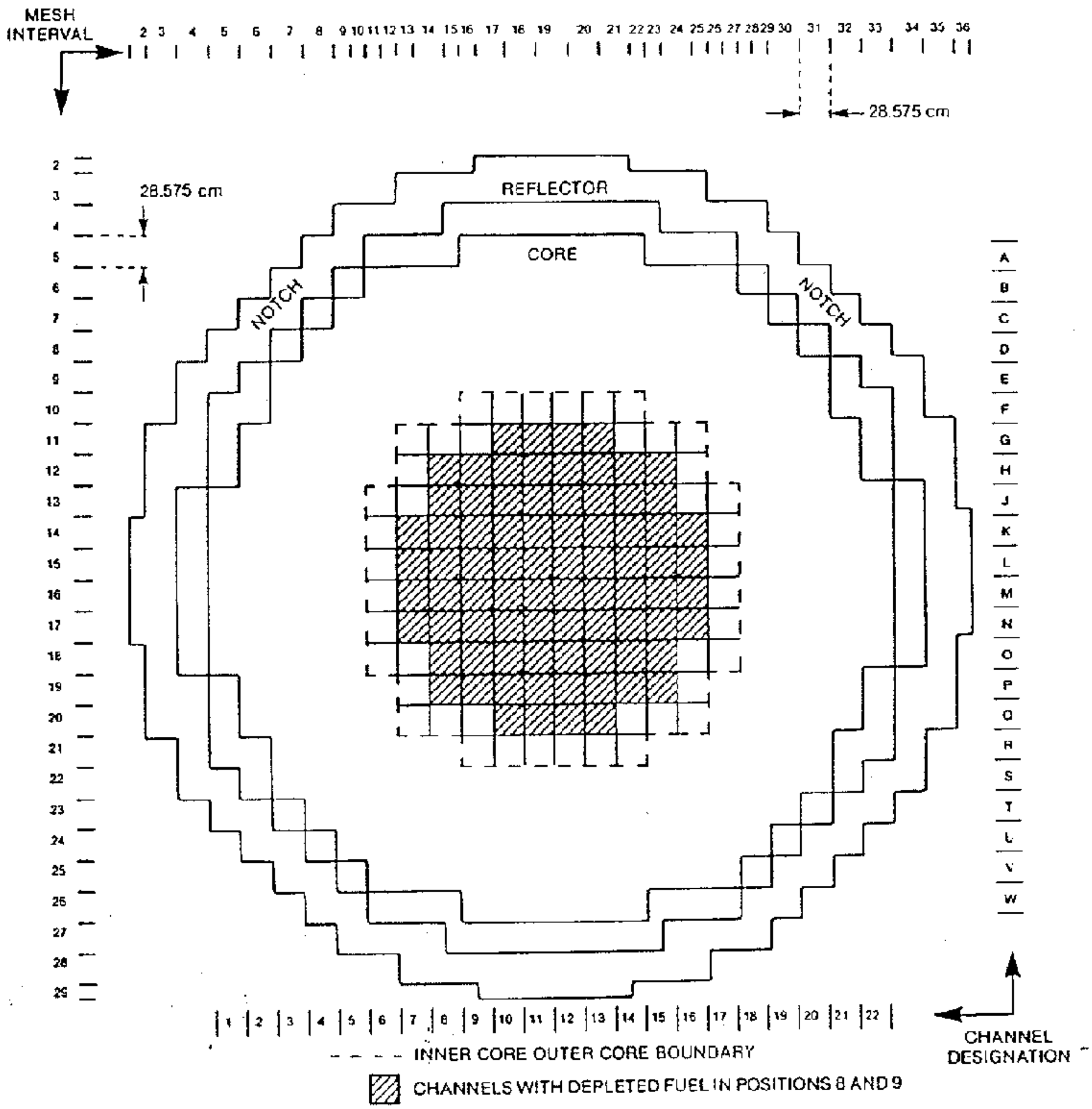


Figure 5.3 Channels with Depleted Fuel in Initial Core of CANDU 6

Even with some depleted fuel in the core, the fact that all fuel is fresh results in a net excess reactivity in the core – see Figure 5.4. [Figure credit: CANTEACH Document 20040502, "Introduction to CANDU Reactor Physics", by B. Rouben, 2002 September, p.163]

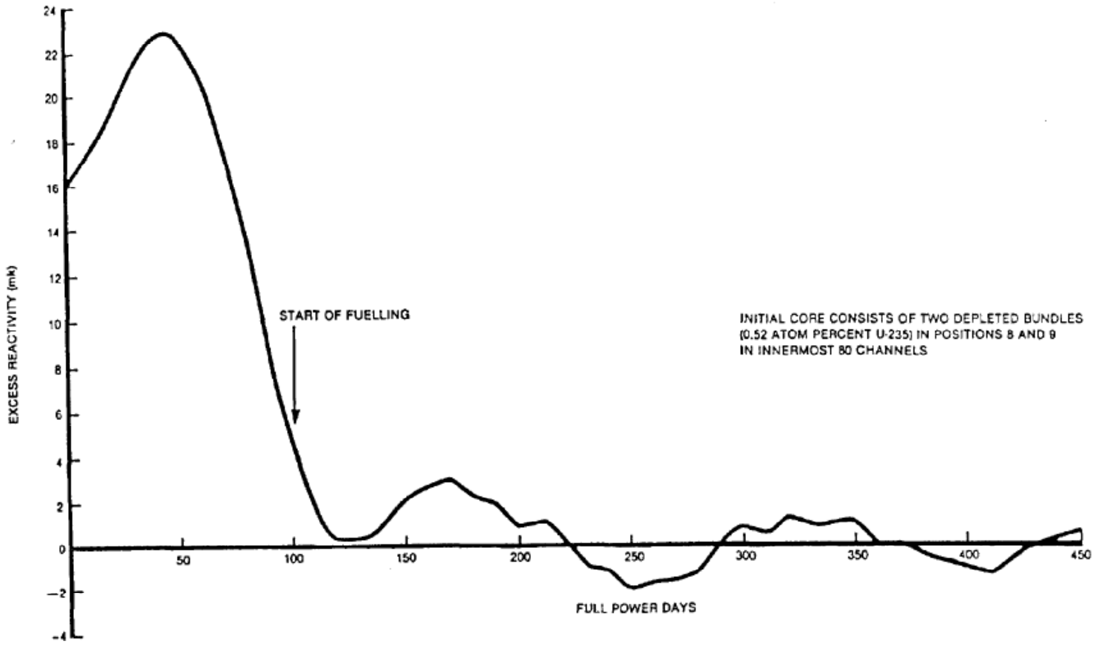


Figure 5.4 Core Excess Reactivity as a Function of Time from First Full Power

The core reactivity starts at approximately 16 milli-k at full power on FPD 0. Because all the fuel goes through its plutonium peak at about the same time, the excess reactivity initially increases, from about 16 mk to a maximum of about 23 mk between FPD 40 and FPD 50 This excess reactivity is compensated by soluble boron in the moderator. The boron coefficient of reactivity is about -8 milli-k per ppm of boron. Thus the boron concentration (at full power) is initially approximately 2 ppm, rising to about 3 ppm at the plutonium peak. Following the plutonium peak, boron must be removed (by ion exchange) as the excess reactivity drops gradually to zero at about FPD 120.

During this entire first period in the reactor life, refuelling is not necessary since there is already excess reactivity. Actually, refuelling is started about 10 or 20 FPD before the excess reactivity reaches 0, i.e. around FPD 100, because the refuelling rate would be too great if one waited until the last possible moment to start.

Beyond the plutonium peak, the overall power distribution flattens out, and the maximum bundle power drops.

5.5 Period from Onset of Refuelling up to Equilibrium

When refuelling begins, the inner core region has the highest burnup and the lowest power relative to the equilibrium power distribution. Refuelling begins in this region, causing power to rise both because of the addition of fresh fuel and the simultaneous discharge of irradiated and depleted fuel. Only some of the channels in this region can be refuelled, however, otherwise the power would rise excessively.

Refuelling of outer-region channels follows. In this region, channel burnup decreases with increasing distance from the core centre (i.e., decreasing power). Therefore, the refuelling tends to proceed generally from the central core region towards the periphery. However, not all channels at a given radius can be refuelled at the same time. Some channels in each ring are initially bypassed for two reasons: first, it is desirable not to refuel adjacent channels simultaneously, because this would cause a local power peak; and second, it is desirable to have a distribution of burnup in each ring when equilibrium is reached. Channels missed on the first refuelling of a ring will be refuelled later, until, when the last channels are visited, the burnup in each ring is uniformly distributed between zero and discharge value. Note that this means that the channel with the highest burnup is not always the one which is refuelled.

After refuelling begins, the rate of refuelling rapidly approaches its equilibrium value (approximately 16 bundles per FPD for the CANDU 6). Over short periods, there can be a considerable variation from this average rate.

5.6 Consequences of Fuelling-Machine Unavailability

If refuelling were to stop, core reactivity would continuously decrease. The rate of reactivity decay is about 0.4 mk/FPD in the CANDU-6 core. The reactor regulating system (RRS) would of course attempt to maintain criticality.

The first action that the reactor regulating system (RRS) would take to maintain criticality is to lower the level of water in the liquid zone-control compartments. Eventually, the water will be drained to the lower limit of the control range.

Since the desirable operating range of the zone controllers is between 20% and 70% (i.e., a range of 50%), and since the full reactivity range of the zone controllers (from 100% down to 0%) provides about 7 milli-k of reactivity, the number of days which can be "survived" without refuelling is typically about 3.5 mk/(0.4 mk/FPD), i.e., about 8 FPD.

The operator would also ensure that any poison which might exist in the moderator at the time would be removed. Every ppm of boron is worth about 8 mk, however the operating license usually limits the amount of boron in the core in full-power operation to about 0.625 ppm (5 milli-k), so this represents at most about 12 FPD without refuelling.

Continued lack of refuelling would lead to withdrawal of the adjuster rods in their normal sequence. This would permit operation to continue for several weeks. However, as the adjuster rods are withdrawn, the reactor power must be gradually reduced because of changes in the power distribution associated with spatial changes in the distribution of absorption cross section. In effect, withdrawal of the adjusters results in a radially "peaked" power distribution, i.e., higher channel and bundle powers at the center of the core, which forces a power derating in order to remain in compliance with the licensed channel and bundle powers (7.3 MW and 935 kW respectively). The amount of derating necessary increases with the number of adjusters withdrawn.

6. Summary

There is a very wide variety of reactor-physics calculations. They touch all aspects of reactor design, analysis, and operation. The physics calculations require an entire suite of specialized computer codes, ending with the full-core code. The physics codes exchange data among themselves and with codes of other disciplines, especially thermalhydraulics. Many analyses in fact require self-consistency between all components of the calculation.

One specific application of reactor-physics codes is in fuel management. Fuel management in CANDU has both design and operations aspects.

The design component consists of establishing:

- the desired time-average power distribution for the equilibrium core, which will be used as the target power shape by the site fuelling engineer, and
- the configuration of depleted fuel in the initial core.

The design of the time-average distribution is facilitated by the flexibility in selecting region-specific (or, in the limit, channel-specific) target exit-irradiation values and axial refuelling schemes, allowed by the CANDU on-power-refuelling feature.

The operations component is the responsibility of the site fuelling engineer or reactor physicist. It involves:

- core-follow calculations, typically performed 2 or 3 times per week to keep close track of the in-core flux, power, and burnup distributions and of the discharge burnup of individual bundles,
- the selection of channels for refuelling, based on the current core state, power and burnup distributions and zone-control-compartment water fills, and
- the determination of the CPPF (channel-power-peaking factor) value, used as a calibration factor for the ROP detectors.

The job of the site reactor physicist never gets boring. These tasks keep the job interesting and stimulating.