CHAPTER 18

Fuel Cycles

prepared by

Mukesh Tayal and Milan Gacesa – Independent Consultants

Summary

Most commercial reactors currently use the once-through fuel cycle. However, because there is still a significant amount of useful material (and available energy) in the fuel discharged from these reactors, many fuel cycles are possible in which some fuel components are recycled for further reactor use. In addition, several other fuel cycles are also possible using thorium, which is much more abundant on Earth than the uranium that is the primary source of commercial nuclear power today. The CANDU reactor is designed to use natural uranium fuel, which is less expensive and more efficient in the use of uranium than any known alternative. Some of the features that enable a CANDU reactor to operate on natural uranium also make it eminently capable of using alternative fuels. In spite of increased fuel cost, the use of alternative fuel cycles in a CANDU reactor can lead to benefits such as significant extensions of humanity's energy resources, reduced capital costs of nuclear power plants, reduced volume and duration for long-term storage and disposal of nuclear waste, longer life of some components of nuclear power plants, increased efficiency of the thermal cycle, and reduced severity of some postulated accidents. Several alternative fuel and fuel cycles are described in this chapter, including the enriched uranium fuel cycle, the recovered-uranium cycle, the MOX cycle, the thorium cycle, the DUPIC cycle, the tandem cycle, low void reactivity fuel, and actinide burning fuel. In addition to increasing fuel cost, alternative cycles impose operating requirements that are more challenging than those experienced when using natural-uranium, low-burnup fuel. Incremental technical challenges in the following subject areas are described here: internal gas pressure, power ramps, corrosion, deuterides and hydrides, deposits, end-temperature peaking, bowing, and high burnup structure in the pellet.

Table of Contents

List of Tables

List of Figures

1 Introduction

The term “fuel cycle” commonly refers to all the steps through which the nuclear fuel goes from mining, to use in a reactor, to discharge, possible recycle, and eventual disposal. Therefore, a fuel cycle consists of certain steps in the *front end*, which involves preparation of the fuel; other steps in the *service period*, in which the fuel is used during reactor operation; and still other steps in the *back end* which are necessary to manage, contain, and either reprocess or dispose of spent nuclear fuel safely.

1.1 Overview

This chapter focusses primarily on selected aspects of the “front ends” and the “service periods” of illustrative fuel cycles that use natural uranium, enriched uranium, plutonium, thorium, and their combinations. The discussion covers simplified neutronics, fuel manufacturing, and selected considerations of associated in-reactor fuel performance. The remaining aspects of fuel cycles are covered elsewhere in this book.

The simplest fuel cycle involves a single pass of the fuel through the reactor; this is called the once-through cycle. Current CANDU reactors run on the once-through natural uranium fuel cycle. A significant amount of useful material (and available energy) still exists in the fuel which is discharged from LWR reactors. Therefore, many fuel cycles are possible in which some fuel components are recycled for further reactor use, particularly in a CANDU reactor. In addition, several other fuel cycles are also possible by using thorium, which is much more abundant on Earth than the uranium that is the primary source of commercial nuclear power today.

We can derive a number of benefits by these means, including significant extensions of humanity’s energy resources, reduced capital costs of nuclear power plants, reduced volume and duration for long-term storage and disposal of nuclear waste, longer life of some components of nuclear power plants, increased efficiency of the thermal cycle, and reduced severity of some postulated accidents. However, several other important factors such as cost also need to be considered, as outlined later in Section 6.

1.2 Learning outcomes

The goal of this chapter is for the student to acquire a broad initial overview of some of the more interesting variants of fissile and fertile materials and the associated fuels and fuel cycles. These include fuel cycles based on the use of natural uranium, enriched uranium, plutonium, thorium, and their combinations. The student will also learn how to evaluate some aspects of fuel cycles and the performance of alternative fuels.

2 Neutronics of Fuel Cycles

2.1 Definitions

Fissile Nuclides: Nuclides that have a significant fission cross section at zero energy.

Fissile nuclides can fission by reacting with a neutron of any kinetic energy, including zero or near-zero kinetic energy (i.e., a slow/thermal neutron).

Example of fissile nuclides: ^{235}U, ^{239}Pu, ^{233}U, ^{241}Pu.

^{235}U is the only naturally-occurring fissile nuclide. The interaction of ^{235}U with a thermal neutron gives ^{236}U, which may fission in about 10^{-14} seconds, releasing on average about 2.5 neutrons. Therefore, we can write:

$$n + {}^{235}U \rightarrow \text{Fission Products} + 2.5n + \gamma.$$

Fissionable Nuclei: Nuclides that can undergo fission (either by a slow or by a fast neutron) are called fissionable. Fissile nuclides are therefore a sub-category of fissionable nuclides. Nuclides which can fission only by reacting with a neutron of high kinetic energy (i.e., a fast neutron) are fissionable, but not fissile.

Examples of fissionable nuclides that are not fissile: ^{232}Th, ^{238}U.

In the case of some non-fissile but fissionable nuclides, the energy-dependent fission cross section displays a threshold with incident neutron energy. The cross section is negligible up to the high energy threshold and has a step increase at the threshold.

Fertile Nuclides: Fertile nuclides are nuclides that can capture a neutron and transmute into a fissile nucleus following one or more nuclear decays.

Examples of fertile nuclides: ^{232}Th, ^{238}U (these happen to be fissionable as well).

Conversion Ratio *C*

To yield fissile nuclides, fertile nuclides must be irradiated with neutrons, which are created as a result of fission. Fission, in turn, consumes fissile nuclei, as does capture. We can look at the process of converting fertile nuclides into fissile ones as consuming some fissile nuclides [Lamarsh, 1966].

The *conversion ratio C* can be defined as:

$$C = \frac{\text{rate at which fissile nuclides are produced in the entire reactor by conversion of fertile nuclides}}{\text{rate at which fissile nuclides are consumed (by fission and radiative capture)}};$$

i.e., *C* is the average number of fissile nuclides produced by conversion per fissile nuclide consumed (by fission and radiative capture).

The total number of fissile nuclides produced starting from *N* initial fissile nuclides can be calculated as:

$$NC + NC^2 + NC^3 + \cdots = \frac{NC}{1-C},$$

where $C < 1$.

Including the initial fissile nuclides, the total number of fissile nuclides available for fission is therefore:

$$N + \frac{NC}{1-C} = \frac{N}{1-C}.$$

It can be seen that the total number of nuclides available for fission increases with the conversion ratio. Because *C* is a function of the aggregate nuclear properties of fuel which can change with time, the value of *C* can also change with time.

If $C > 1$, more fissile nuclides are created than destroyed. *C* is then called the *breeding* ratio. *Breeding* can therefore be thought of as conversion with $C > 1$. When breeding occurs, the *breeding gain* can be defined as:

$$G = C - 1.$$

Fuel Burnup

The *specific burnup* (sometimes called just *burnup*, ω) equals the fission energy produced per unit mass of heavy elements in the fuel:

$$\omega = \frac{Fission\ Energy\ Generated}{Total\ Initial\ Mass\ of\ Heavy\ Nuclides\ in\ Fuel}.$$

The usual units for fission energy are either MWh or MWd. Because ^{235}U is by far the most frequently used fissile material, initial mass is usually expressed either as kg or T (tonnes) of U. However, if other fissile materials are used, e.g., plutonium or a mixture of uranium and plutonium, then the initial mass is usually expressed as kg or T of HE (heavy elements).

2.2 Dynamics of Depletion

The reactions by which fertile nuclides produce fissile nuclides are, for example:

$$n + {}^{232}Th \rightarrow \gamma + {}^{233}Th \underset{\beta-decay}{\rightarrow} {}^{233}Pa \underset{\beta-decay}{\rightarrow} {}^{233}U \tag{1}$$

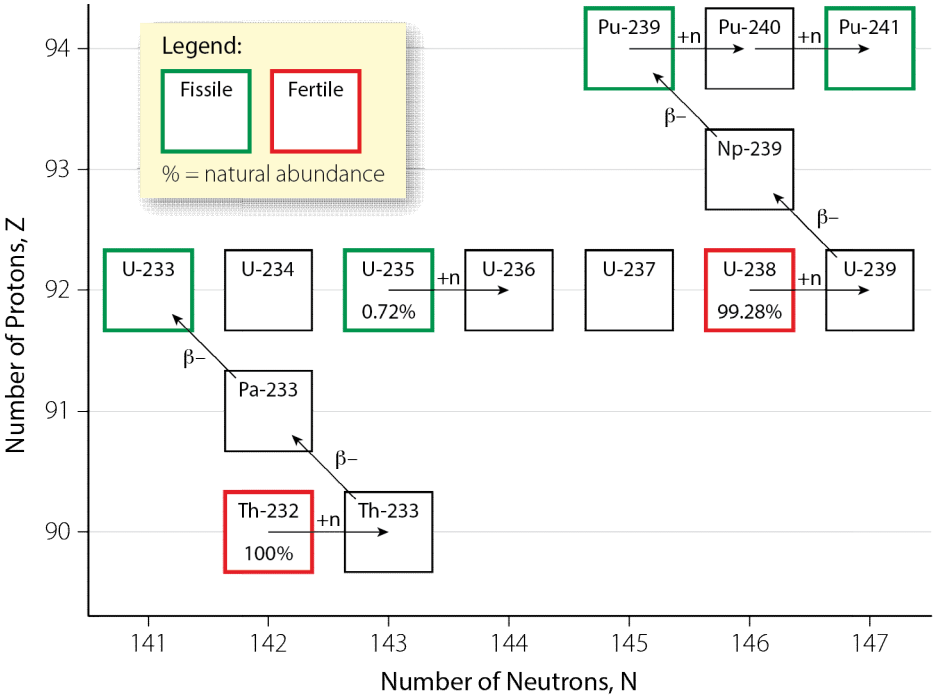
$$n + {}^{238}U \rightarrow \gamma + {}^{239}U \underset{\beta-decay}{\rightarrow} {}^{239}Np \underset{\beta-decay}{\rightarrow} {}^{239}Pu \tag{2}$$

and subsequently:

$$n + {}^{239}Pu \rightarrow \gamma + {}^{240}Pu \tag{3}$$

$$n + {}^{240}Pu \rightarrow \gamma + {}^{241}Pu\ . \tag{4}$$

Figure 1 illustrates a more generalized conversion of fertile nuclides to fissile nuclides.



(Courtesy Bill Garland)

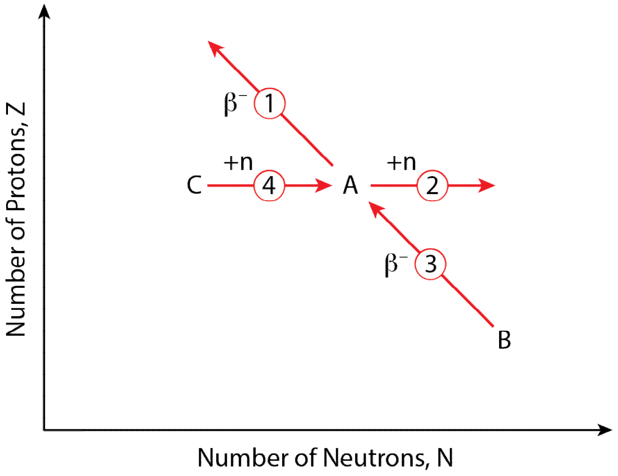
Figure 1 Transformation of specific isotopes

Recall from neutron physics that moving to the right in Figure 1 involves adding a neutron (i.e., +*n*), whereas moving left and up involves beta decay (i.e., β-). For clarity, Figure 1 shows arrows and appropriate symbols (i.e., +*n*, β-, etc.) mainly for the key reactions discussed above.

The general nuclide history can be followed by modelling all processes involved [Lamarsh, 1966; Garland, 2005]:

1. loss by decay of the nuclide;
2. loss by neutron capture by the nuclide, which transmutes it into another isotope of the same element;
3. production by decay from a parent nuclide;
4. production by capture transmutation;
5. imposed fuel loading changes.

The first four processes in this list are illustrated in Figure 2.



(Courtesy Bill Garland)

Figure 2 Generalized isotope transformation processes

The governing rate equation is:

$$\frac{dN_A}{dt} = \underbrace{-\lambda_A N_A}_{(1)\text{decay}} - \underbrace{\left[\sigma_a^A \phi\right] N_A}_{(2)\text{neutron capture}} + \underbrace{\lambda_B N_B}_{(3)\text{parent decay}} + \underbrace{\left[\sigma_\gamma^C \phi\right] N_C}_{(4)\text{capture transmutation}} + \underbrace{F(t)}_{\text{fuel loading}} \quad . \qquad (5)$$

where N_A, N_B, N_C are the number of nuclei (#/cm^3) for isotopes A, B and C respectively, λ is the associated decay constant (sec^{-1}), ϕ is the neutron flux (#/cm^2-sec) and σ_a, σ_γ are the microscopic cross sections (cm^2) for neutron absorption and capture respectively. We can safely assume that the flux is constant over the integration time step and can easily solve this equation numerically. In general, there are many nuclides to track, so they are all solved for simultaneously, thereby tracking the depletion and growth of the various fuel nuclides. Note that the two production processes (3 and 4 above) mean that there is potential for fuel creation, called *conversion* or *breeding* as explained earlier.

The above process yields information on whether or not a given combination and arrangement of materials will achieve criticality and sustain the chain reaction and how much energy can be so produced. The detailed "physics" evaluations of specific fuel cycles use the same general principles that are outlined in earlier chapters on physics and are usually accomplished mainly by changing the inputs of appropriate "generic" physics codes.

3 Overview of Some Possible Fuels and Fuel Cycles

This section outlines some possible fuels and fuel cycles such as the natural uranium cycle, the enriched uranium cycle, the thorium cycle, the breeding cycle, actinide burning fuel, and low void reactivity fuel.

In addition, this section also outlines some interesting and clever fuel cycles that involve extracting significant additional energy from Light Water Reactor (LWR) fuel that would normally be considered as "spent"—that is, fuel that has been discharged from the reactor after having delivered all the energy that is reasonably expected from it. Some cycles that extract additional energy in this manner are: the recovered uranium cycle, the MOX cycle, the

DUPIC cycle, and the Tandem cycle. They are also outlined in this section.

3.1 Reprocessing

Before we delve into various fuel cycles, we first outline reprocessing of LWR fuel as it pertains to this chapter.

Considerable quantities of fission products gradually accumulate inside the fuel element during irradiation. Some fission products are parasitic in that they absorb neutrons without creating proportionate heat. This leaves fewer neutrons to continue the chain reaction, and at some stage, further irradiation of that material risks shutting down the chain reaction. Therefore that fuel is removed from the reactor and is frequently labelled as "spent" or "waste" or "discharged".

Uranium fuel used in LWRs has an initial ^{235}U concentration (enrichment) of between 3% and 5%. After many months (sometimes even a few years) of generating energy in the reactor, most but not all of the original ^{235}U in typical enriched fuels in LWRs is largely used up (e.g., by fission and by radiative capture). Likewise, some fertile nuclides in the fuel, as well as some fissionable nuclides produced by fertile nuclides, are also used up. However, fissionable nuclides are not at zero concentration when the fuel is discharged from the reactor. Spent fuel retains some of the original enriched uranium, and it also contains some fissile plutonium that is produced by the chain reaction (see equations and discussion in Section 2 and in earlier chapters on physics).

The residual fissile content in spent LWR fuel depends on the starting enrichment and the discharge burnup; nevertheless, for illustration purposes, let us say that it is typically in the range of about 1.5% (^{235}U + ^{239}Pu + ^{241}Pu). From the perspective of a CANDU reactor, this is a much higher fissile concentration than even the usual fresh (non-enriched) CANDU fuel (0.7%). Even though LWRs cannot continue to "burn" that fuel, CANDU reactors can, due to the latter's superior neutron efficiency. This creates an opportunity to generate additional energy from spent LWR fuel in a variety of ways, as described later in this section.

However, to make effective use of available thermal neutrons, we first need to remove sufficient parasitic neutron absorbers—such as fission products—from spent LWR fuel. This is done through "reprocessing".

Very simplistically, let us think of fuel as having the following initial and final states before and after its irradiation in an LWR:

$$\underbrace{^{235}\text{U (3-5\%)} + {}^{238}\text{U}}_{\text{(fresh LWR fuel)}} \rightarrow \underbrace{\{^{235}\text{U + Pu}\}\text{ (about 1.5\%)} + {}^{238}\text{U + fission products (FPs)}}_{\text{(spent LWR fuel)}}$$

Let us next focus only on the right-hand side of this expression. Discussions later in this section use the terms explained below:

- For purposes of this chapter, "conventional reprocessing" or "full reprocessing" means extracting, separating, and purifying uranium (mostly ^{235}U and ^{238}U) and plutonium (mostly ^{239}Pu and ^{241}Pu) in spent fuel. This, however, leads to some concerns that collected plutonium can potentially be diverted towards nuclear proliferation. Full reprocessing generally consists of the following main steps: (Step #1) Irradiated fuel is dissolved; (Step #2) Plutonium and uranium are separated from fission product and then

(Step #3) from each other; and (Step #4) purified for enrichment (in the case of REU) or (Step #5) recycling as MOX.

- In one variant involving partial reprocessing, *all* fission products are removed from spent fuel, but plutonium is neither separated from uranium nor extracted separately, and therefore concerns about proliferation are reduced compared to "full reprocessing". This product is used in the "Tandem cycle" (discussed later).
- In another variant involving partial reprocessing, no dissolution of spent fuel takes place. The gaseous fission products are removed by crushing and heating the spent fuel; the solid fission products are retained in the fuel. The high residual radio-toxicity of this product significantly reduces its attractiveness for potential diversion, which further diminishes concerns about nuclear proliferation. This product is used in the "DUPIC cycle", as described later in this chapter.
- Plutonium recovered from full reprocessing is used to generate electricity in the "MOX cycle".
- Uranium that is collected from full reprocessing is called "recovered uranium" (RU). Its ^{235}U fraction in spent LWR fuel is higher than in natural (as-mined) uranium. RU is used to generate electricity in the "recovered uranium" cycle.

In summary, a variety of fissile and fertile materials are available from different sources in multiple forms such as natural uranium, enriched uranium, thorium, various products of full or partial reprocessing, and others (e.g., dismantled nuclear weapons). This presents opportunities to design a variety of fuels and fuel cycles in neutron-efficient CANDU reactors, some of which are outlined in the following sections.

3.2 Natural Uranium Fuel Cycle

In this cycle, the initial fissile nuclide is ^{235}U as found in mined (natural) uranium. This yields an inexpensive fuel which is the staple of the current generation of CANDU nuclear power plants.

Fission of initial ^{235}U is subsequently augmented by plutonium, produced by neutrons captured in ^{238}U. By the time the fuel is discharged from the reactor, almost half the fissions come from plutonium.

3.3 Enriched Uranium Fuel Cycle

This cycle uses ^{235}U as the initial fissile nuclide that has been enriched, typically to less than 5%.

Enriched uranium fuels are the staples of a variety of reactor types, such as pressurized water reactors (PWRs), boiling water reactors (BWRs), and Russia's VVER/RBMK reactors. In these reactors, enrichment is used to provide an abundance of fissile atoms, which (1) increases the probability of neutron fission and decreases the importance of moderator efficiency, and (2) prolongs the period before fissile atoms must be replenished in the reactor.

During the late 1900s and early 2000s, CANDU reactor designers seriously considered using enriched fuels in designs such as the Advanced CANDU Reactor (ACR) [Reid *et al*., 2008] and in low void reactivity fuels [Boczar *et al*., 2004]. In CANDU reactors, the additional neutrons generated by enrichment can be used for a variety of purposes such as: (1) using less expensive coolant (i.e., light water) to reduce capital cost; (2) enabling use of thicker pressure tubes to reduce their creep rate and thus to extend their service life; (3) enabling use of burnable neutron absorbers which can reduce the magnitude of a power pulse during a postulated accident; and (4) reducing the volume of spent fuel through extended burnup.

Enriched uranium fuel could also be used in existing CANDU reactors to achieve other benefits such as higher reactor power (through a flatter radial channel power profile).

3.4 Recovered-Uranium Cycle

This is a variant of the enriched uranium fuel cycle. In this cycle, LWR fuel is reprocessed to recover unused uranium. The recovered uranium is reused to make new uranium-based fuel (if necessary, with the addition of fresh uranium).

Thanks to superior neutron efficiency, the CANDU reactor can use RU fuel from LWRs as-is, without additional enrichment, to achieve the benefits mentioned above for enriched uranium fuel.

A number of publications have described the salient attributes of RU fuel, for example Boczar *et al.* [1993 and 2010]. Briefly, the ^{235}U content of RU depends mainly on initial enrichment and discharge burnup of LWR fuel. Conventional reprocessing yields very pure RU with very few contaminants from fission products.

In principle, residual fissile material is also available in used CANDU fuel. However, used natural-uranium (NU) fuel's residual fissile content is much lower than that of used enriched fuels from LWRs. Hence, recovery of residual fissile material for re-use would be significantly more expensive from used NU fuel than from used enriched fuel [Boczar *et al*., 2010].

Significant amounts of high-grade highly enriched uranium (HEU) can also be obtained from dismantled nuclear warheads for use in power reactors.

3.5 MOX Cycle

In this cycle, plutonium recovered through reprocessing LWR fuel is mixed with either fresh or depleted uranium (from enrichment tails) to make mixed uranium- and plutonium-oxide fuel (MOX). The plutonium concentration is adjusted to give the required enrichment and burnup.

Significant amounts of high-grade plutonium can also be obtained from dismantled nuclear warheads for use in power reactors.

3.6 Thorium Cycle

^{232}Th is fertile and can be used to produce ^{233}U, which is fissile and is used *in situ* as it is produced. In the CANDU reactor, a once-through thorium cycle is economically viable, which enables energy to be derived from ^{233}U without reprocessing and recycling. Full benefit from the thorium cycle would require reprocessing of the used fuel to recover and recycle the ^{233}U at an appropriate concentration as new reactor fuel.

Thorium cycles are important because thorium is much more abundant on Earth than uranium. Some countries or regions have thorium resources, but lack plentiful uranium. They can extend the energy from natural uranium (which is the source of the only natural fissile material) by using the thorium cycle. In addition, thorium cycles provide other benefits, such as in fuel performance and fuel safety, as explained in a later section devoted specifically to thorium.

3.7 DUPIC Cycle

This is one of the cycles that can use discharged LWR fuel in CANDU reactors after partial reprocessing. In the DUPIC cycle, the LWR fuel is reprocessed using dry thermal or mechanical methods to remove gaseous fission products [Boczar *et al*., 1996]. This yields a fuel that neutron-efficient CANDU reactors can use.

Uranium and plutonium are not separated during this operation, and therefore this fuel is attractive from the perspective of nuclear non-proliferation. Irradiation tests have confirmed satisfactory performance of DUPIC fuel [Karam *et al*., 2010(a)].

3.8 Tandem Cycle

This is another cycle based on the use of discharged LWR fuel in CANDU reactors after partial reprocessing. In this cycle, the plutonium and uranium in LWR fuel are separated from all the fission products, but are not separated from each other, as is done in conventional LWR fuel reprocessing operations [Hastings *et al*., 1991]. Therefore, the extraction of fissile materials is more attractive from the non-proliferation perspective than the full reprocessing operation, but less attractive than DUPIC, as explained at the beginning of this section.

3.9 Low Void Reactivity Fuel

One important aspect of CANDU reactor safety is its *coolant void reactivity* (CVR). If a significant volume of the coolant were to be lost in an accident, a complex change in the neutron spectrum would initially rapidly increase the reactivity in the affected reactor core. This would automatically trigger corrective action(s) by the reactor's shutdown system(s), which would quickly decrease the reactivity again. Therefore, the affected core's reactivity would peak for a very short duration. The initial increase in reactivity is called "positive coolant void reactivity". In CANDU reactors, the magnitude of the CVR during postulated loss of coolant accidents sets several requirements for the shutdown system, such as speed, redundancy, and depth. A smaller value of CVR (including a negative value) could loosen these performance requirements and/or increase the safety and/or licensability of the reactor.

One option to reduce such reactivity is the appropriate use of a very small amount of burnable neutron absorber (BNA) in the fuel bundle. The low void reactivity fuel (LVRF) bundle uses a burnable neutron absorber (BNA) such as dysprosium and locates it strategically in the centre element. Enriched uranium is used in the outer elements to compensate for neutrons absorbed by the BNA. The amount of neutron absorber and the level of enrichment can be varied to give the desired values of coolant void reactivity and burnup, although the decrease in void reactivity is at the expense of uranium utilization [Boczar *et al*., 2004]. Such a fuel design has been developed and successfully tested in the reactor [Boczar *et al*., 2004].

3.10 Actinide Burning Fuel

The back end of the fuel cycle continues to attract considerable public scrutiny. A potential amelioration involves reducing the duration for which used fuel remains a radioactive hazard. This could potentially be achieved by fissioning some of the long-lived actinides from used LWR fuels in fast reactors. In a variant of this concept, actinides can first be removed from used LWR fuel (through an enhanced reprocessing route) and blended with a fuel matrix such as uranium, thorium, or a neutronically inert material such as zirconia and used as fuel in a CANDU reactor. The resulting waste would have significantly lower decay heat and radiotoxicity. These actinide-burning CANDU reactors could be used on their own to dispose of spent fuel or in conjunction with fast burner reactors. Use of the CANDU burner reactor would significantly reduce the number of fast reactors that would need to be built for ultimate destruction of LWR actinide waste [Hyland *et al*., 2010; Bhatti *et al*., 2013].

As a by-product, this process would also create many rare earths, which are economically very valuable.

3.11 Breeding Cycles

In fast reactors, breeding cycles are possible: more fissile fuel is produced from fertile nuclides than the amount of fissile fuel destroyed. This cycle in fast breeder reactors can then be used multiple times to increase the use of nuclear-fuel resources. For example, ^{238}U is one of the possible fertile fuels for breeder cycles. This nuclide makes up most of mined uranium—over 99%—compared to 0.7% of ^{235}U. Although current thermal reactors do also convert a tiny fraction of the original ^{238}U into energy, designs of thermal reactors are tailored primarily towards extracting energy from the relatively miniscule fraction of ^{235}U that exists in mined uranium. Through breeder cycles, we can produce energy from a far greater fraction of the much more abundant ^{238}U, thus increasing humankind's nuclear-fuel resources by factors of 50–100 compared with their use in thermal reactors.

4 Key Drivers for Advanced Fuel Cycles

This section provides a broad overview of some of the more significant advantages of alternative fuel cycles.

4.1 Increase the Sustainability of the Nuclear Fuel Cycle

Energy resources can be expanded, sometimes dramatically, through the following methods individually or in combination:

- By using thorium, which is much more abundant than uranium. Thorium fuel has other advantages as well in safety, in-reactor performance, and long-term waste reduction, as discussed in Section 9.3;
- By using plutonium, which can be obtained from a variety of sources such as dismantled warheads or by reprocessing used fuel, as discussed in Section 9.4;
- By using enriched uranium, which can be obtained from a variety of sources such as dismantled warheads or by reprocessing used LWR fuel, as discussed in Section 9.2;
- By using used LWR fuel, which has been reconfigured (but not reprocessed), in a CANDU

reactor;

- By using a breeder reactor to extract energy from ^{238}U, which constitutes over 99% of natural uranium, as discussed in Section 9.5.

Section 5 gives a more detailed discussion of resource extension.

4.2 Improve Fuel Cycle Economics

The natural uranium (NU) fuel cycle, which is currently used in CANDU power reactors, provides low-cost fuel and the highest uranium utilization (energy derived from mined uranium) of any current commercial reactor. The fuel bundles for this cycle are also relatively easy to manufacture. Nevertheless, the economics of the CANDU reactor can be further improved by using enriched uranium. Enriched fuel can be used to:

- Reduce the initial capital cost of a CANDU reactor by enabling the use of lower-cost light water coolant.
- Safely uprate reactor power: Enriched uranium fuel can also be used to uprate reactor power (if allowed by the rest of the plant) by flattening the radial channel power distribution across the core. This means that the channels produce close to the same amount of power and that about 15% more power can be extracted from the same number of fuel channels in a given size of reactor.
- Safely increase bundle power: Different amounts of power are produced in different rings of fuel elements. By adding different degrees of enrichment to the fuel elements that produce low power in different rings, their heat output can be raised without exceeding the safe limits of local power (including considerations of heat removal).
- Increase the lifetime of plant components by measures such as thickening the pressure tubes, thus reducing their creep rate and increasing their service life. The enriched fuel compensates for the increase in parasitic material in the core. Longer service life means that mid-term replacement of pressure tubes can be delayed, thus extending the total life of the plant.
- Increase the efficiency of the steam cycle. Thicker pressure tubes also permit operation at higher coolant pressure and hence at higher outlet temperature. At the higher temperature, steam converts a higher fraction of its thermal energy into electricity. This in turn increases the overall efficiency of the steam cycle.
- Reduce the frequency of on-power fuelling. Higher burnup means a longer residence period of the fuel in the channel. This, in some cases, translates into less frequent channel fuelling and hence a decreased number of duty cycles on the fuelling machines. This in turn may potentially also increase the time available for maintenance of these machines.
- Reduce the volume of spent fuel by extending the burnup. The cost of storage, however, may not be reduced proportionally due to other considerations.
- Reduce the cost of the long-term disposal of spent fuel through the transmutation of hazardous fuel components such as long-lived actinides.

4.3 Improve the Health and Environmental Impacts of the Fuel Cycle

The health and environmental impacts of nuclear power can be improved by using enriched uranium in CANDU reactors and by transmuting hazardous components of the fuel, such as long-lived actinides, in a reactor. This reduces the mass of these materials that requires long-term disposal. This in turn reduces the size of the repositories that will be required and reduces potential impacts on human health and the environment.

4.4 Improve Safety Characteristics

Additional neutrons produced by enriched fuel can be used to offset the reactivity reduction when burnable neutron absorbers are used to reduce coolant void reactivity, as discussed in the section on low void reactivity fuel.

4.5 Improve the Proliferation Resistance of Fuel Cycles

The proliferation resistance of used nuclear fuel can be improved by irradiating derivatives in CANDU reactors, as follows:

- Burn both weapons plutonium and plutonium from used reactor fuel, thereby reducing the inventory of material for potential use in a nuclear explosive device.
- Use new fuel types (such as thorium denatured with ^{238}U) which create spent fuel that is unattractive for use in nuclear explosives.

5 Resource Extension: A Major Incentive

This section expands on one benefit of alternative fuel cycles noted in the section above: extension of world-wide resources for fission energy. Although the subject has been considered by many analysts, the discussion below is based largely on Meneley's analysis [2006] of the world energy situation over the next 100 years, focussing largely on oil and nuclear.

As the world's population and living standards continue to rise, demand for energy is also likely to increase, despite conservation. Oil supply analysts agree that world oil production must decline at some time during the 21st Century; some say that we can expect to see the peak of world oil production very soon. Therefore, there could be a substantial long-term gap (demand minus supply) in availability of primary energy. If nuclear energy is used to fill a significant fraction of the expected energy gap, would it also soon run into resource limitations?

Table 1 summarizes the energy requirements of various fuel cycles in thermal reactors. For a given amount of mined uranium, the fuel cycle that is neutronically the most efficient—1.2% enriched uranium in a CANDU reactor—produces about 90% more energy than the cycle that is neutronically the least efficient.

Table 1 Energy output per megagram of mined uranium

	MWy(e)/Mg
Enriched U in PWR, BWR	4.61
Pu Recycle in PWR, BWR	5.41
DUPIC (PWR → CANDU reactor)	6.37
Natural U in CANDU reactor	6.37
1.2% Enriched U in CANDU reactor	8.77

[Source: D. Meneley, 2006]

Based largely on the projections of the International Energy Agency (IEA) supplemented as required by assumptions, Meneley deduced that to fill the projected world-wide gap in primary energy, we would need some 6,000 nuclear reactors, each with a capacity of 1 GWe. This is significantly more than the currently installed nuclear capacity world-wide.

Table 2 summarizes Meneley's results for years of uranium availability for the above scenario, i.e., a nuclear capacity of 6 TWe. It suggests that if enriched ^{235}U were used to fuel pressurized water reactors (PWRs) or boiling water reactors (BWRs), the planet's identified uranium resources would last about 30 years. If the same ^{235}U were instead used in natural-uranium CANDU reactors, the uranium resources could be extended to last about 42 years. If 1.2% enriched uranium were used in CANDU reactors, these uranium reserves would last about 58 years. The extension from about 30 years in LWRs to about 58 years in CANDU reactors is largely due to the superior neutron efficiency of the latter.

If plutonium is recovered from used fuels and reused in LWRs, reactors can be run for 36 years. Alternatively, if used LWR fuel is reused in CANDU reactors in a DUPIC cycle along with top-up enriched uranium, reactors can be run for 42 years—even without the expensive intermediate step of extracting plutonium.

Note, however, that when uranium eventually becomes scarce, prices will likely increase, exploration will be spurred, and additional reserves can be expected to be discovered.

Table 2 Years of uranium availability

	Megawatt-Years per Tonne	Years of Uranium Availability for 6000 1000-MWe Units
Enriched U in PWR, BWR	4.61	30
Pu Recycle in PWR, BWR	5.41	36
DUPIC (PWR → CANDU Reactor)	6.37	42
Natural U in CANDU Reactor	6.37	42
1.2% Enriched U in CANDU Reactor	8.77	58
Fast Reactor	1,000	6700
Fast Reactor using Seawater*	1,000	>100,000

* Seawater contains about 0.003 ppm of uranium, for a total of over 4 billion tons of uranium.

[Courtesy D. Meneley; Extracted from a presentation to Canadian Nuclear Society by D. Meneley, 2006]

A more dramatic way to extend our energy resources is to breed fissile materials from fertile materials. This can usually be done in two ways: one, fertile ^{232}Th can be converted into fissile ^{233}U; and two, fertile ^{238}U can be converted into fissile ^{239}Pu, as summarized below.

"The absorption cross-section for thermal neutrons of ^{232}Th (7.4 barns) is nearly three times that of ^{238}U (2.7 barns). Hence, a higher conversion (to ^{233}U) is possible with ^{232}Th than with ^{238}U (to ^{239}Pu). Therefore, ^{232}Th is a better 'fertile' material than ^{238}U in thermal reactors, but ^{232}Th is inferior to ^{238}U as a 'fertile' material in fast reactors. For the 'fissile' ^{233}U nuclide, the number of neutrons liberated per neutron absorbed is greater than 2.0 over a wide range of the thermal neutron spectrum, unlike ^{235}U and ^{239}Pu. Therefore, contrary to the ^{238}U-to-^{239}Pu cycle in which breeding can be obtained only with fast neutron spectra, the ^{232}Th-to-^{233}U fuel cycle can operate with fast, epithermal, or thermal spectra" [IAEA, 2005].

Thorium resources have not yet been explored as extensively as uranium; nevertheless, experts estimate that thorium in the Earth's crust is some three to four times as abundant as uranium [IAEA, 2005]. Boczar *et al*. [2010] have estimated that "if all the plutonium from used natural uranium fuel from existing CANDU reactors operating over their lifetime were to be used in a mixed thorium/plutonium cycle, two to three times the existing nuclear capacity in Canada (currently about 15 GWe) could be sustained indefinitely" in a self-sufficient thorium cycle in a CANDU reactor, e.g., a near-breeder. With a breeder, an expanding nuclear system could be fueled essentially indefinitely (depending on the breeding ratio).

Other experts have estimated that ^{238}U in "waste" CANDU fuel alone that has already been accumulated in Canada can, after reprocessing and recycling in fast breeder reactors, produce some 4,000 years of electricity at today's consumption rates [Ottensmeyer, 2012].

Moreover, additional ^{238}U is available underground. Therefore, in principle, fast breeder reactors can provide energy for millennia to come, as shown in Table 2.

At this time, these are only theoretical possibilities. Many technical, economic, and practical concerns remain about fast breeder reactors, such as the very large fissile inventory needed to start a fast reactor and the very daunting concerns about proliferation.

6 Other Considerations

In addition to the drivers described above, various other technical and non-technical factors should be considered in deciding on the most appropriate fuel cycle for a specific power plant, utility, or country. Although a detailed discussion of this subject is beyond the scope of this book, five major aspects are summarized below: energy independence, resource utilization, resistance to nuclear proliferation, fuel performance, and cost.

Energy Independence

Plants that enrich large quantities of uranium are very expensive and are operational in only a few nations. Fuel cycles based on natural uranium do not require a nation to build a plant to enrich uranium, nor do they require a nation to rely on a few other nations for continual supply of enriched uranium over a very long period. Therefore, natural uranium cycles promote national energy independence.

Resource Utilization

To produce a given amount of energy, some fuel cycles require less mined uranium than others. For example, the combination of heavy water and natural uranium used in current CANDU reactors requires some 40% less mined uranium per unit energy than fuel cycles that use light water and enriched uranium. This can be a consideration in countries that do not have much uranium and would like to stretch their domestic uranium to the extent practical.

Resistance to Nuclear Proliferation

There is concern in some quarters that during reprocessing, potent fissile material can potentially become accessible and hence available for diversion to uses other than electricity generation. Some fuel cycles such as DUPIC are comparatively more resistant to nuclear proliferation.

Fuel Performance

Overall, current fuels perform very well in current fuel cycles, as noted in Chapter 17. In fuel cycles that differ significantly from current cycles, satisfactory fuel performance must be designed and demonstrated. Depending on the specific cycle, new R&D may potentially also be required.

Cost

Generally, enriching and reprocessing fuel is expensive, and the radioactivity and radiotoxicity of spent enriched fuel tends to be higher than that of natural uranium. At the same time, enriched fuel generally yields higher burnup, which tends to reduce fuelling cost (within a range) and also reduces the volume of spent fuel to be stored. Therefore, this is a complex topic, and its impact also changes with local conditions and with time.

In summary, the choice of a specific fuel cycle must consider not only the drivers discussed earlier, but also other techno-commercial and geo-political aspects, some of which are noted above.

7 Pertinent Features of the CANDU Reactor

A number of features of a CANDU reactor enable it to use alternative fuel cycles more effectively than other reactor types, mainly through higher neutron efficiency and also through easier processing and fabrication of recycled fuel. These features are explained below.

Higher Neutron Efficiency

Natural uranium has low fissile content. To enable fission in a sustained chain reaction, CANDU reactors are, of necessity, designed with high neutron efficiency. This in turn makes CANDU reactors attractive for using other low-fissile materials in alternative fuel cycles.

The following main features contribute to the high neutron efficiency of CANDU reactors [Boczar, 2012]:

- CANDU reactors use heavy water (D_2O) as both moderator and coolant. This reduces parasitic absorption of neutrons compared to light water.
- The moderator, which is separate from the coolant, is at low temperature. This promotes neutron efficiency.
- In CANDU reactors, a small cluster of fuel elements (i.e., a fuel bundle) is surrounded by a large amount of low-temperature moderator. This thermalizes a higher fraction of neutrons.
- The principal structural components inside the core, e.g., the calandria tubes, pressure tubes, and fuel bundle, are made from zirconium alloys. This reduces parasitic absorption of neutrons, for example in comparison to components made of stainless steel.
- CANDU reactors use relatively thinner structural materials. For example, as a proportion of fuel element diameter, sheaths are significantly thinner in CANDU reactors than in LWRs, reducing parasitic absorption of neutrons in the former.
- CANDU reactors are refueled on-power, as necessitated by the use of NU. This has a consequential benefit in that the reactor can be operated effectively with low excess reactivity in the core, obviating the need for burnable neutron absorbers and minimizing neutron absorption in control materials.
- CANDU reactors contain equipment for on-power refuelling, which provides flexibility in accommodating alternative fuels.

The resulting high neutron economy of a CANDU reactor enables high fuel utilization regardless of the fissile material (^{233}U, ^{235}U, or plutonium). About 90% more energy can be extracted from a particular fissile material than in an LWR reactor [Boczar *et al.*, 1996], as shown in Table 1.

Easier Fabrication and Reprocessing of Fuel

The simple, small fuel bundle design of a CANDU reactor (see Chapters 8 and 17) facilitates remote processing and fabrication for highly radioactive recycled fuels.

A key "enabler" of these advantages is the separation of moderator and coolant in a CANDU reactor through pressure/calandria tubes. This separation permits the reactor designer to adjust the moderator and the coolant independently to exploit neutronic characteristics more effectively. For example, the moderator can be kept at low temperature to promote neutron efficiency, while the coolant can simultaneously be kept at high temperature to promote the thermal efficiency of the steam cycle. In another example, the choice of coolant fluid and the moderator-to-coolant-volume ratio can potentially be fine-tuned to reduce capital cost without adversely affecting fuel cooling, as is done with the Advanced CANDU Reactor. CANDU reactors are unique among successful contemporary reactor systems in offering this important flexibility to designers.

8 Natural Uranium Fuel Cycle

The front end of the NU cycle starts with mining, refining, and conversion of uranium into ceramic UO_2 powder. The next segment of the front end is fuel manufacturing, including the procurement and processing of Zircaloy components. These front-end segments are described below.

8.1 Production of Ceramic UO_2 Powder

The steps involved in production of reactor-grade ceramic UO_2 powder include mining, refining, and conversion, as summarized below.

There are four methods of obtaining uranium from ores:

- Open-pit mining;
- Underground mining (the main method until the recent past);
- *In-situ* leaching (currently the leading method);
- Recovery from mining of other minerals such as copper, phosphate, or gold.

Regardless of how the uranium is obtained, it cannot be fed directly into a nuclear power station because it contains mineral and chemical impurities, including radioactive daughter products of uranium. Processing is required before uranium can be used efficiently to generate electricity.

If uranium is obtained by open-pit or underground mining, the first process after mining is crushing (milling) of the ore, followed by dissolution in an acid or alkaline medium to separate the uranium metal from unwanted rock and purification with chemicals which selectively leach out the uranium. The uranium-rich solution is then chemically separated from the remaining solids and precipitated out of solution. Finally, the uranium is dried. The resulting powder is uranium oxide concentrate, U_3O_8, commonly referred to as “yellowcake” due to the bright yellow colour that it frequently, but not always, exhibits. The unwanted solids, called “tailings”, are stored at the mine site for safe disposal.

If the *in-situ* leaching method is used, an acidic or alkaline mining solution is passed directly through the underground ore body by means of a series of bores or wells, and uranium is brought to the surface in a dissolved state for purification. The uranium-rich solution is treated as described above, the principal difference being that no tailings are produced by this method.

Uranium meeting nuclear-grade specifications is usually obtained from yellowcake through a tributyl phosphate solvent-extraction process. First, the yellowcake is dissolved in nitric acid to prepare a feed solution. Uranium is then selectively extracted from this acid feed by tributyl phosphate diluted with kerosene or some other suitable hydrocarbon mixture. Finally, uranium is stripped from the tributyl phosphate extract into acidified water to yield highly purified uranyl nitrate, $UO_2(NO_3)_2$. Uranyl nitrate is the starting material for conversion to uranium dioxide (UO_2) powder or to uranium hexafluoride (UF_6). Both these conversion routes conventionally begin with calcining the nitrate to UO_3.

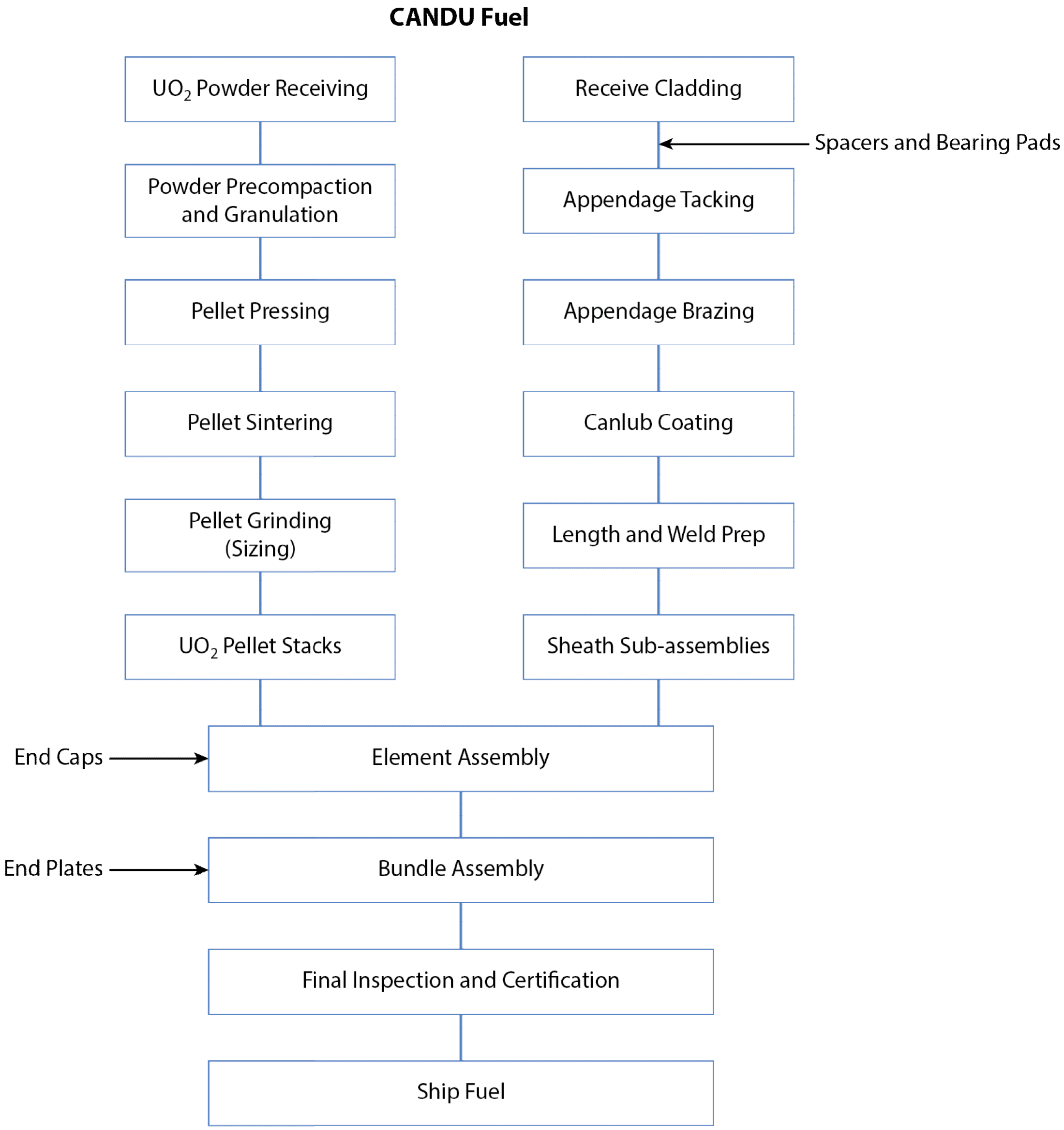
A large percentage of the UO_3 produced in Canada is converted to uranium hexafluoride (UF_6) destined for enrichment plants for use in LWRs, and the remainder is converted to natural uranium dioxide (UO_2) ceramic-grade powder for use in CANDU reactors. The processes involved in producing natural uranium (NU) UO_2 powder for CANDU fuel pellets are described in more detail below. The UO_3 is dissolved in nitric acid, and the resulting solution is diluted and reacted with ammonium to precipitate ammonium diuranate (ADU). The liquid is decanted, and the ammonium diuranate slurry is dried to remove the water. The resulting ammonium diuranate powder is dried in a rotating kiln in hydrogen gas to produce UO_2 powder. The powder may then be subjected to further physical “conditioning” to improve its mechanical “reactivity” during pellet sintering. Small batches of each powder lot may be subjected to sintering trials as part of the final QA acceptance test before it is sent to manufacturers of CANDU fuel bundles.

At each production stage mentioned above, the uranium compound is subjected to stringent quality control surveillance to ensure that the final product meets industry technical specifications with respect to the content of specified impurities as well as certain physical characteristics related to subsequent manufacturability. In-reactor behaviour of CANDU fuel bundles has demonstrated that some aspects of fuel performance (e.g., fission product behaviour) are related not only to powder characteristics, but also to the process parameters used in producing the powder.

8.2 Fuel Bundle Manufacturing

A simplified CANDU fuel bundle manufacturing flow sheet is shown in Figure 3.

Manufacturing of bundles starts in two separate production streams: production of UO_2 pellets (left-hand side (LHS) of Figure 3), and production of sheath sub-assemblies and other Zircaloy components (right-hand side (RHS) of Figure 3). These two production streams require such diverse processes and expertise that they are not only carried out in separate production lines, but also may be carried out at different geographical locations.



(Courtesy Candu Owner's Group)

Figure 3 CANDU fuel manufacture: assembly flow chart

The UO_2 pellet production stream starts with receipt of UO_2 powder (Step 1, LHS of Figure 3). As received, the powder consists of very fine particles that tend to agglomerate and impede rapid and consistent filling of the pellet pressing dies. To improve the "flow" of the powder, it is subjected to two physical processes: compaction and granulation (Step 2). During this step, a lubricant is also added to aid pellet pressing (Step 3), which is followed by pellet sintering (Step 4). Sintering is one of the most critical steps in pellet production and requires precise control of temperature, time, and the atmosphere used during the sintering cycle. Pellet density and density distribution as well as physical integrity (no cracks or "chips") are critically dependent on proper control of this process step.

As sintered, the pellets are hourglass-shaped (see Chapter 17 for additional discussion of this phenomenon). The sintered pellets are ground (Step 5) in a "centreless" grinder to (1) remove the hourglass shape and produce a straight cylindrical pellet shape, (2) size the pellet to the desired diameter, and (3) impart a fine finish to the cylindrical pellet surface.

The ground pellets are sorted by diameter and placed into stacks (Step 6) ready for insertion into sheath sub-assemblies, which come from the second stream. Pellet stack length is controlled to tight tolerances because it is the principal means for ensuring adherence to the required axial clearance within the fuel element. Pellet stacks and sheath sub-assemblies are sorted by diameter and matched to satisfy the diametric clearance requirements. In addition, to prevent interference with the sheath, the pellets at both ends of the stack may be tapered, or their diameter may be reduced. Additional discussion on this topic is provided below.

Some of the greatest challenges in CANDU fuel production occur in the UO_2 pellet production stream. These challenges derive from the pellet technical specifications, some of which are unique to CANDU fuel. A few of the major challenges are high pellet density; precise dimensional control, especially of pellet diameter; small size of permissible visible surface flaws, referred to as “chips”; and high quality of surface finish, especially the pellet cylindrical surface (see Chapter 17 for additional discussion on pellet/sheath interaction). These challenges have been successfully met by manufacturers of CANDU fuel. This is substantiated by the almost complete absence of defects ascribed to pellet manufacturing deficiencies, the only notable exception being failures observed in “overstuffed” fuel elements, which were intentionally manufactured to challenge the extreme limits of the relevant parameters: maximum pellet density, minimum diameter clearance, and minimum axial clearance. Once defects were witnessed in “overstuffed” elements, the practice was subsequently discontinued.

The second manufacturing stream, the production of sheath sub-assemblies, starts with receipt of tubes finished to the required diameter and cut to length (Step 1, RHS of Figure 3). Some manufacturers (including both Canadian manufacturers) have “vertically integrated” Zircaloy tubing production into this stream, so that the stream actually starts with receipt of zirconium tube hollow (TREX), which undergoes several stages of reduction and associated processing (not shown in Figure 3), culminating in cut-to-length tubes.

In parallel with receipt of tubes, Zircaloy sheet/strip, bar/rod, wire, graphite slurry, and beryllium are also received. The sheet (or wire) is used to manufacture spacers and bearing pads (appendages). When manufactured from sheet, the appendages are punched and “coined” into their final shape from the sheet after it is coated with beryllium. When manufactured from wire, the appendages are cut to length, coined, and subsequently coated with beryllium. The appendages are “tacked” to the tubes with spot welds (Step 2) and heated until the beryllium coating and the Zircaloy metal in the appendage and tube form an alloy, i.e., “braze” (Step 3). Appendage dimensions, the shape of the coined surfaces, the thickness of the brazing metal, and the location and tacking of the appendages on the tube must be controlled to tight tolerances to enable the manufacturer to achieve the required dimensional tolerances for the assembled bundle. The manufacturers have successfully met the challenges of this part of the manufacturing process, as evidenced by the absence of any fuel failures caused by detached appendages or appendages causing bundle incompatibility within the fuel channel.

Graphite slurry is diluted with water or an industrial alcohol, applied to the inside of the tube sub-assemblies, and dried to form a coating of precisely controlled thickness and hydrogen content referred to as "CANLUB" (Step 4). The effect of CANLUB on in-reactor performance of fuel elements is discussed in Chapter 17. The ends of the sheath sub-assemblies are machined (Step 5) to (1) achieve the required final length, and (2) produce the profile required for end-cap welding. In addition, CANLUB and any debris are removed from the end of the sub-assembly lest they interfere with the subsequent welding process.

The two process streams (LHS and RHS in Figure 3) come together at the element assembly step, i.e., the insertion of pellet stacks into the sheath sub-assemblies and the welding of element end closures (endcaps). Endcaps are machined from bar or rod to match the sheath sub-assembly weld preparation and to include other features (internal and external) which are crucial to the final configuration of the element assembly. They are then welded to the sheath using resistance welding, which is a thermo-electric process.

During resistance welding, the sheath is held by a collet. Heating during resistance welding softens the sheath locally. The combination of the heating and cooling cycle that occurs during the welding cycle and the restraint provided by the collet causes the sheath diameter to shrink immediately adjacent to the weld. To maintain the minimum required diametric clearance in this region of the element, the pellets at both ends of the stack have a reduced diameter (usually tapered to mimic the sheath diameter profile).

After insertion of the pellet stack into the sheath sub-assembly, and before welding of the end caps, the air trapped inside the sheath sub-assembly is purged with helium (or a mixture of helium and argon). The end cap is then welded while the weld region is blanketed by inert gas.

Similarly to the pellet production stream, the Zircaloy component stream has its own unique CANDU-related challenges, the most unique and critical being the dimensional and process parameters related to welding of endcaps to sheath sub-assemblies. To appreciate the importance of this operation more fully, it is useful to remember that the largest CANDU reactors have close to one-half million of these welds in the active part of the core and that one defective weld is cause for concern, requiring urgent action to remove it from the core. In comparison, LWR reactors have no discontinuities or joints of any type in fuel cladding within the active part of the core. To make sure that the contribution of endcap welding to defect statistics in CANDU reactors remains insignificant, defects due to endcap welding must be maintained at or below the part-per-million (ppm) level, which is extremely challenging. Careful examination of the fuel manufacturing flow chart, including the relevant component design and process parameters, reveals more than 15 design and process parameters which must be maintained within control limits and must come together at the precise moment the weld is made. It is not surprising that of the very few defects in operating CANDU fuel which have been ascribed to manufacturing causes, more than 90% are related to endcap welding. Fuel integrity statistics for CANDU reactors and LWRs are discussed in Chapter 17; notwithstanding the contribution from endcap welding to defects in CANDU fuel, defects in CANDU fuel from all causes are significantly fewer than defects in LWR fuel (see Chapter 17).

Following endcap welding, the fuel elements are inserted into an assembly jig and the element ends resistance-welded to endplates to form the final fuel assemblies, i.e., the fuel bundles. End plates are stamped out of an incoming strip. Although this final process in itself is not as challenging as some of the processes described above, it is important to note that some of the dimensional requirements on the bundle assembly are equivalent in precision to those which are normally associated with machined components. To achieve this, manufacturers must have stringent control, not only of the final assembly process, but also of the processes used to produce the component parts, particularly element sub-assemblies and endplates.

Manufacturing is the final step in the realization of a product, converting the intent specified in the design drawings and technical specifications into functioning hardware. For one unit of energy produced, current CANDU reactors use five to six times the mass of fuel used in LWRs. Despite this relative disadvantage, fuelling cost per unit energy has always been much lower for CANDU than for LWR. This can be attributed mainly to three factors: (a) CANDU reactors avoid the high cost of enrichment, (b) CANDU reactors use 40% less mined uranium per unit of energy produced than LWRs, and (c) CANDU fuel manufacturers have managed to keep their fabrication costs very low while keeping fuel reliability very high. In addition, Canadian fuel manufacturers have also been very successful in transferring access to their technology to all non-Canadian manufacturers of CANDU fuel.

9 Details of Selected Alternative Fuels and Fuel Cycles

From the ten possible alternative fuel cycles summarized in Section 3, selected five alternative cycles are described in greater detail in the sections below.

9.1 Natural Uranium Equivalent (NUE) Fuel

In all respects other than the sourcing of the fissile material (^{235}U) and the starting amount of fissile material (%^{235}U in total U), this “cycle” is essentially the same as the natural uranium (NU) cycle. The starting feedstock (UO_2 ceramic-grade powder) for this cycle is a blend of recycled enriched uranium (REU) and depleted uranium (DU) from enrichment plant tails. The proportion of each component is determined by the respective content of ^{235}U in each. For instance, if the ^{235}U content of REU is 0.9% ^{235}U in total U and that of DU is 0.2% ^{235}U in total U, then the proportion of the two components per unit weight of the blend, with an enrichment of 0.71% ^{235}U in total U, can be calculated as follows:

If x is the fraction of REU in the blend, then the fraction of DU is $(1 - x)$. If the enrichment of the final mixture is 0.71%, then:

$$x*0.9 + (1 - x)*0.2 = 0.71. \tag{7}$$

By solving Equation (7) for x, one can conclude that the blend needs to consist of 73% REU and 27% DU. Similar calculations can also be performed for other combinations of starting ^{235}U contents of the constituent components. It is likely that the blended ^{235}U fissile content for this cycle will have to be very slightly higher than for NU to compensate for impurities in the REU and DU, which are expected to be higher than the respective impurities in NU. The exact ^{235}U content in the blend would be determined for each batch of fuel procured, based on the measured amount of impurities in each component.

Once the NUE powder has been obtained, the fuel bundle is manufactured using the same processes that are used to produce NU fuel bundles, as described earlier.

Satisfactory performance of NUE fuel has been confirmed in “demonstration” irradiations in a commercial power plant in China [Jioa *et al*., 2009].

9.2 Extended Burnup Fuel

As noted in Section 3, enriched uranium can be used for several purposes in a CANDU reactor. One of them is to increase significantly the discharge burnup of CANDU fuel. This means that a given amount of energy can be produced by fewer fuel bundles, reducing the amount of spent fuel (i.e., high-level waste) that needs to be stored and disposed of.

To balance the above, one must consider (a) the increased cost of fuel, because enrichment is expensive, (b) technical considerations in several key areas of fuel performance, such as internal gas pressure, power ramps, corrosion, hydriding, deposits, end-temperature peaking, bowing, high burnup structure, and (c) other effects related to plant operation, such as the effect of enriched uranium on coolant void reactivity (CVR) and on refuelling operations.

Explanations of how fuel integrity is affected by some of these mechanisms have already been discussed in Chapter 17 from the perspective of normal CANDU burnup. High burnup exacerbates some of these, especially those discussed below. The sub-sections that follow provide additional science that can aid in developing successful fuel designs for extended burnup, which is a common feature of many of the alternative fuel cycles described in Section 3.

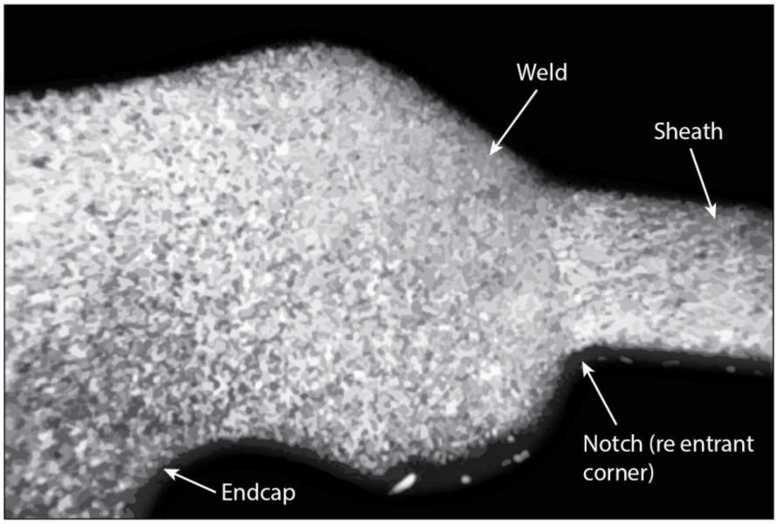
From the perspective of maximum utilization of mined uranium, the optimal enrichment (and hence burnup) for a CANDU reactor is around 1.2%, which is much lower than the typical enrichments in LWRs. Therefore, many of the performance challenges discussed below for extended burnup fuel are significantly lower in CANDU fuel than in high-burnup LWR fuels.

However, in CANDU fuel, the high burnup occurs in conjunction with high element power, which frequently exacerbates the challenge to fuel integrity. On the other hand, in CANDU reactors, on-line fuelling provides a measure of flexibility in shaping the axial and radial power profiles throughout the core. This can be used to help keep peak element ratings within limits and reduce the size of power ramps resulting from refuelling.

9.2.1 Internal Gas Pressure

Extended burnup has the potential to increase, among other things, the internal gas pressure within a fuel element (see Chapter 17). If excessive, the higher gas pressure can potentially overstress and crack the fuel sheath. Some CANDU fuel elements indeed failed in the Bruce reactor when they stayed in the reactor for unusually long periods and their internal pressure inadvertently exceeded failure limits. Likewise, excessive internal pressure has also failed a few fuel elements in experimental fuels as well [Floyd, 2001].

Three techniques are available to avoid such failures. First, fission gas release can be reduced, e.g., by reducing element ratings. Second, the void space in the element can be increased, e.g., by using bigger chamfers, gaps, or plena [Floyd, 2001]. Third, stress concentrations at critical locations can be reduced by using a larger notch radius at the sheath/endcap junction [Tayal *et al*., 1993], see Figure 4.



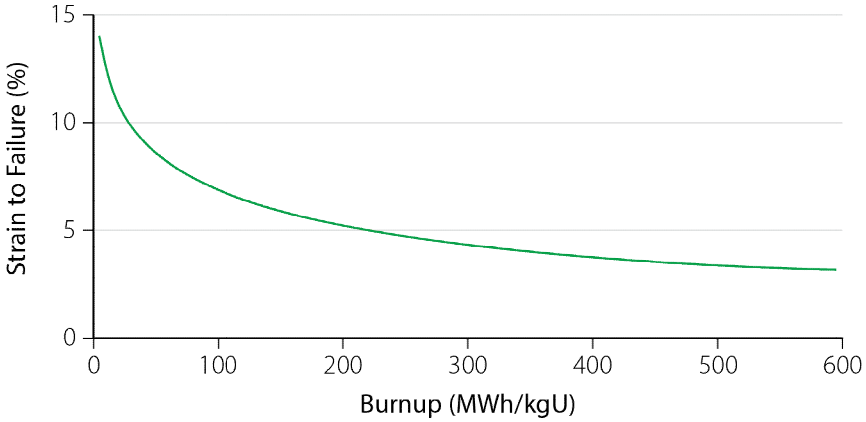
[Source: Page, 1976]

Figure 4 Illustrative notch at the sheath/endcap junction

9.2.2 Zircaloy ductility and power ramp

As noted in Chapter 17, a power ramp causes thermal expansion of a pellet, which in turn pushes the sheath. The sheath can crack if its strain exceeds its ductility in the corrosive environment of fission products.

Irradiation reduces the ductility of Zircaloy significantly, as illustrated in Figure 5 [Tayal *et al.*, 1995]. For example, Figure 5 suggests that irradiation to 200 MWh/kg reduces the ductility of Zircaloy by a factor of three to four compared to its initial as-fabricated value.



[Source: Tayal *et al.*, 1995]

Figure 5 Illustrative effect of irradiation on ductility of Zircaloy

Decreased ductility leads to lower tolerance to power ramps at extended burnups. In Chapter 17 (Section 8.2), the power ramp defect threshold of the sheath was considered in a narrow burnup range of 140 ± 20 MWh/kgU. Fuel can be operated to much higher burnups, especially LWR fuel and extended burnup CANDU fuel. To design safe operating conditions for such fuel, we need to know the power ramp defect thresholds at the higher burnups. To obtain an illustrative idea, let us use Figure 5 of this chapter to extend the defect threshold curve of Figure 24 in Chapter 17 to other burnups. The result, Figure 6, illustrates how higher burnups can decrease resistance to power ramps in non-CANLUB fuel. The decrease is initially rapid, but slows down after burnup of about 100 MWh/kgU. Similar trends are expected in CANLUB fuel as well.

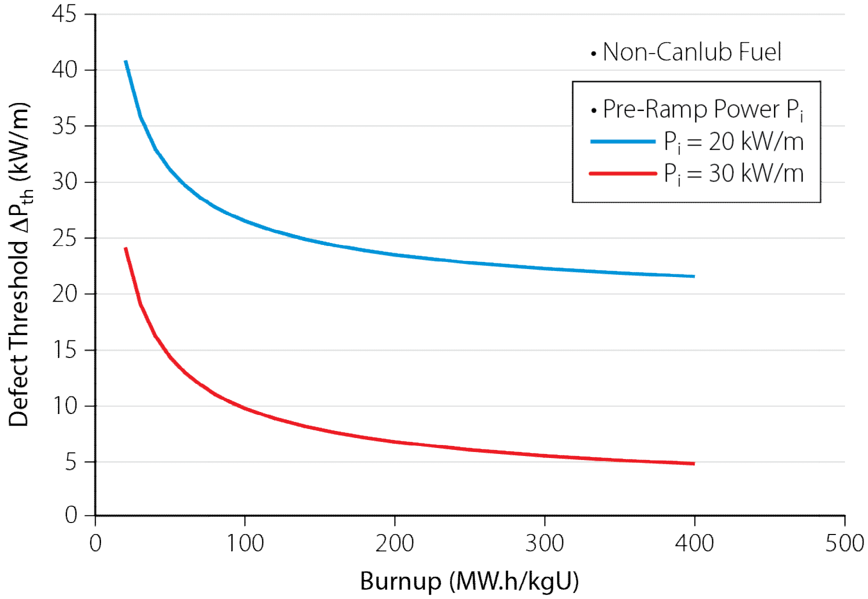


Figure 6 Illustrative effect of burnup on power ramp defect threshold

Exercise: As an illustrative example, assume that a power ramp of 10 kW/m is being considered in non-CANLUB fuel at a burnup of 300 MWh/kgU. Before the ramp, the fuel is operating at 20 kW/m. (a) Will the fuel survive this ramp? (b) Will the fuel survive the same ramp if its pre-ramp power is 30 kW/m?

Figure 6 shows that the first ramp is below the defect threshold, and therefore the fuel will survive it. The second ramp, however, is above the defect threshold; therefore, the fuel will have a non-zero probability of developing defects.

If the fuel's expected ramps are significantly higher than the defect threshold, appropriate steps must be taken to increase the sheath's tolerance to power ramps. These could include, for example, optimizing the shape of the pellet to decrease its thermal expansion; a thicker coating of CANLUB to reduce the concentration of aggressive species of fission products at the sheath surface; or even an entirely new CANLUB formulation.

The decreased ductility of Zircaloy at extended burnup also affects several other areas of fuel performance, for example the mechanical strength of the endplate during discharge. These factors must all be considered carefully during the design process.

9.2.3 Corrosion

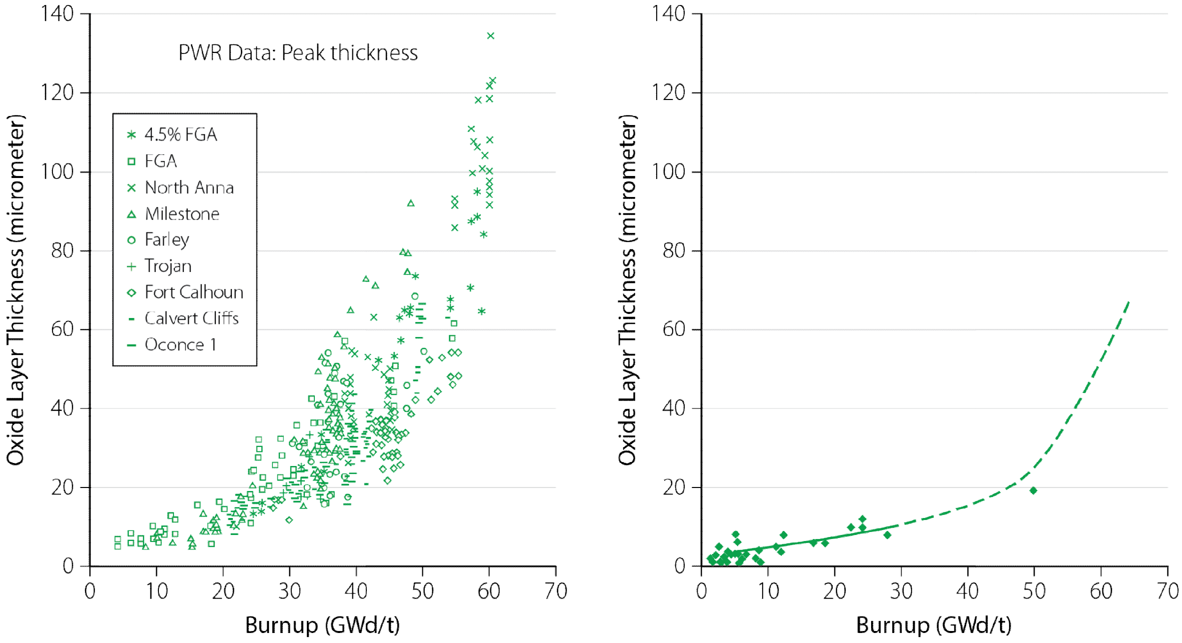
Higher burnup is usually associated with longer residence periods. The latter can potentially increase sheath corrosion. This tends to be much more pronounced at the outer surface of the sheath, which is in contact with water.

When part of the sheath corrodes, its ability to carry mechanical load decreases. Strength assessments must account for this loss, and therefore we need to quantify sheath corrosion. Furthermore, very thick layers of oxide tend to flake, causing a potential for debris and deposits in the heat transport system. In addition, flaking often leaves behind non-uniform patches of oxides. These cause significant axial and circumferential variations in sheath temperature, which in turn cause hydrogen and deuterium to diffuse and concentrate at local peaks. The shape of the resulting hydrogen- (and deuterium)-rich areas often resembles a lens, and therefore the product is often called a "hydrogen lens". If local peaks of hydrogen, hydrides, deuterium, and deuterides become pronounced, they can embrittle the sheath, which in turn can significantly challenge its mechanical integrity. For all the above reasons, the extent of corrosion must be kept within acceptable limits.

In CANDU reactors, lithium hydroxide (LiOH) is added to maintain coolant pH in the 10.5 range, which is intended to minimize deposition of iron-based system corrosion products ("crud") on both fuel surfaces and steam generator tubes [Barber *et al*., 1982]. From out-of-reactor autoclave tests, several investigators have determined that oxidation of zirconium alloys is not affected by the LiOH concentrations that are required to maintain coolant pH near 10.5. Similarly, in-reactor, no acceleration of zirconium-alloy oxidation has been observed in the pre-transition oxidation period, with additions of lithium hydroxide to control pH in the 10.5 range. However, under very long exposures and high burnups which develop relatively thick post-transition oxide films, some acceleration of oxidation has been observed on fuel elements in light water reactors [Garzarolli *et al*., 1982; Cox, 1985]. The latter has been attributed to possible concentration of the lithium hydroxide within the thick oxide films due to temperature rise during heat transfer [Courtesy COG].

Waterside corrosion has indeed been identified in LWRs as a process that warrants the use of more corrosion-resistant materials in the sheaths of very high burnup fuels. In CANDU reactors, however, the burnup, and therefore the residence period, is far less, and therefore current CANDU reactors are still quite far from requiring such an expensive change. Even in enriched CANDU fuel with burnup of about 30 MWd/kg, corrosion is not expected to be a problem.

In summary, there are many detailed nuances in sheath corrosion. Nevertheless, for purposes of this initial introduction, Figure 7 shows an illustrative progression of sheath corrosion with burnup. Note the large scatter in the data, which is typical. Also note that the overall trends are similar in CANDU and PWR sheaths.



(a) PWRs **(b) CANDUs**

[Source: Tayal *et al*., 1995]

Figure 7 Waterside corrosion in the sheath

<u>Example</u>: How much corrosion can be typically expected in a CANDU sheath (a) at 240 MWh/kgU; (b) at 480 MWh/kgU?

From Figure 7(b), we would expect an oxide layer of about 5 µm at 240 MWh/kgU and one of about 15 µm at 400 MWh/kgU. These are both very thin layers and pose no risk of flaking. Note the disproportionate increase in oxidation with burnup.

9.2.4 Deuterides and hydrides

Waterside corrosion liberates deuterium or hydrogen:

In heavy water: $Zr + 2D_2O = ZrO_2 + 2D_2$ (8)

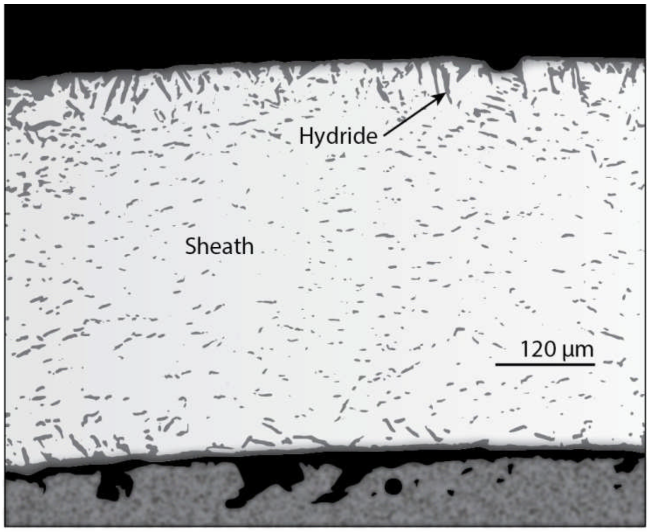
In light water: $Zr + 2H_2O = ZrO_2 + 2H_2$. (9)

In this section, for simplicity, we will use the terms "deuterium/deuteride" to include generically "hydrogen/hydride" depending on context.

Some fraction of the liberated deuterium is dissolved in the sheath, a process sometimes also called "uptake". Over time, the deuterium diffuses due to gradients of concentration, temperature, and stress and can accumulate preferentially at a few spots. When the local concentration exceeds the solubility limit, deuterium precipitates in the form of deuteride platelets, referred to as "deuterides". The solubility limit decreases with lower temperature, and therefore deuterides tend to precipitate preferentially in relatively colder regions.

Depending on their orientation, deuterides can embrittle Zircaloy, thereby reducing its ability to carry tensile loads. For this reason, deuteride levels must be maintained within acceptable limits.

If the coolant is light water rather than heavy water, the same qualitative assessment applies, except that instead of deuteride platelets, hydride platelets (referred to as “hydrides”) are formed. Figure 8 shows an illustrative example of hydrides in a sheath that was irradiated to a burnup of about 600 MWh/kg HE in light water coolant. Note the relatively higher concentrations of hydrides near the outer surface because that surface is cooler than the inner one.



[Source: Karam *et al.*, 2010(b)]

Figure 8 Illustrative distribution of hydrides in the sheath at about 25 MWd/kg HE

There is no evidence that excess hydrogen in the coolant causes primary defects in fuel-bundle metallic components, or that it participates in the formation of secondary damage in fuel metallic components. This may be because there is no thermodynamic mechanism for excess H^+ (or H_2) to enter the matrix of metallic components. The mechanism for introducing deuterium (not hydrogen) into the matrix of CANDU-bundle metallic components is dependent on D_2O coming into contact with an unoxidized metallic component, at which point oxygen, having a greater affinity for the metal than deuterium, combines with the metal, causing the D_2O molecule to dissociate and leaving the liberated D^+ radical (or D_2) inside the matrix of the metallic component. D^+ radical (or D_2) then dissolves within the metal matrix. Excess D^+ (or D_2) within the metal then precipitates to form deuterides in the coldest part of the metallic component, as explained earlier.

Typical values of hydrogen/deuterium (H/D) pickup in extended-burnup CANDU sheaths are available from a compilation presented by Floyd [2001]. For comparison purposes, sheath D concentrations are expressed in terms of equivalent H (=D/2). With the exception of siloxane-coated fuels, equivalent H concentrations up to 200 μg/g are widely observed, with most of the data bounded by 150 μg/g. The data contain large variability with respect to burnup. This suggests that besides burnup, other parameters such as coolant chemistry, neutron fluence, coolant/sheath temperatures, and sheath properties exert significant influence on H/D pickup.

9.2.5 Deposits

[This section was contributed primarily by John G. Roberts, JGRChem Inc. and Dr. Craig R. Stuart, Canadian Nuclear Laboratories].

Deposits are sometimes observed on the external surfaces of fuel bundles and are sometimes also called “crud”. Most deposits on CANDU fuel tend to be magnetite-based compounds. In BWRs, “scale-type” crud contains high concentrations of copper, up to and sometimes even in excess of 50%. In PWRs, deposits tend to be compounds based mainly on nickel ferrite. Deposits on LWR fuels can also incorporate lithium and boron.

Deposits on fuel are undesirable because they can potentially impede heat transfer. In BWRs, they have sometimes even caused through-wall holes in the sheath [IAEA, 2010].

Deposits on fuel are very rare in CANDU reactors, but relatively more frequent and also thicker in BWRs [IAEA, 2010]. The causes of these deposits are rooted mainly in materials and coolant chemistry, which differ in different types of reactors. This is a very complex subject, and its detailed treatment is beyond the scope of this chapter. However, pertinent experts have identified the following four significant differences between the various reactor types that affect in-core deposits:

a) different materials among reactor types, e.g., carbon steel in the CANDU heat transport system and stainless steels in LWRs;
b) presence of boron in PWR coolant and its absence in CANDU coolant;
c) higher degree of success in CANDUs in maintaining consistent chemistry over time in the primary heat transport system (PHTS); and
d) much longer fuel residence period in an LWR (usually a few years) than in a CANDU reactor (usually less than two years).

As for boron in the coolant, recall that CANDU reactors separate the coolant from the moderator using pressure tubes and calandria tubes. This enables CANDU designers to maintain separate chemistries in the coolant and in the moderator. Each system can then be focussed on optimally addressing its own respective needs. For instance, in a CANDU reactor, the additives required for neutron management are injected into the moderator, not the coolant, and therefore the chemistry of the PHTS is not affected. In contrast, when boron is added in an LWR’s moderator-cum-coolant to manage the neutronics, it also has a chemical impact on the entire primary heat transport system (PHTS), including the steam generators and all the pipes. If this promotes crud in the PHTS of an LWR, it is then carried by the coolant into the core, where it can deposit on fuel surfaces. In this way, separation of moderator and coolant in separate systems is one aspect that contributes to fewer crud deposits on CANDU fuel.

In summary, the lesser quantities of deposits on CANDU fuel than on PWR fuel are due principally to different materials in the coolant circuit, lack of boron in the coolant, higher consistency over time in coolant chemistry, and shorter fuel residence period.

9.2.6 End-temperature peaking

Consider two neighbouring fuel bundles in a CANDU channel. Their fuel stacks do not abut each other; they are separated by the two endplates, endcaps, and axial gaps, as shown in Figure 9. Because of the absence of UO_2 in this region of bundle-to-bundle contact, neutron absorption cross sections are smaller here than in the bulk of the fuel bundle. Some fraction of the unabsorbed neutrons from the bundle-to-bundle junction—the "excess" neutrons—becomes available for capture by neighbouring UO_2. This results in an increase in neutron flux in this region; the flux is higher a few centimetres into the fuel pellets on either side of this junction than in the axial centre of the bundle. This is called *end-flux peaking* (EFP) and is illustrated schematically in Figure 9.

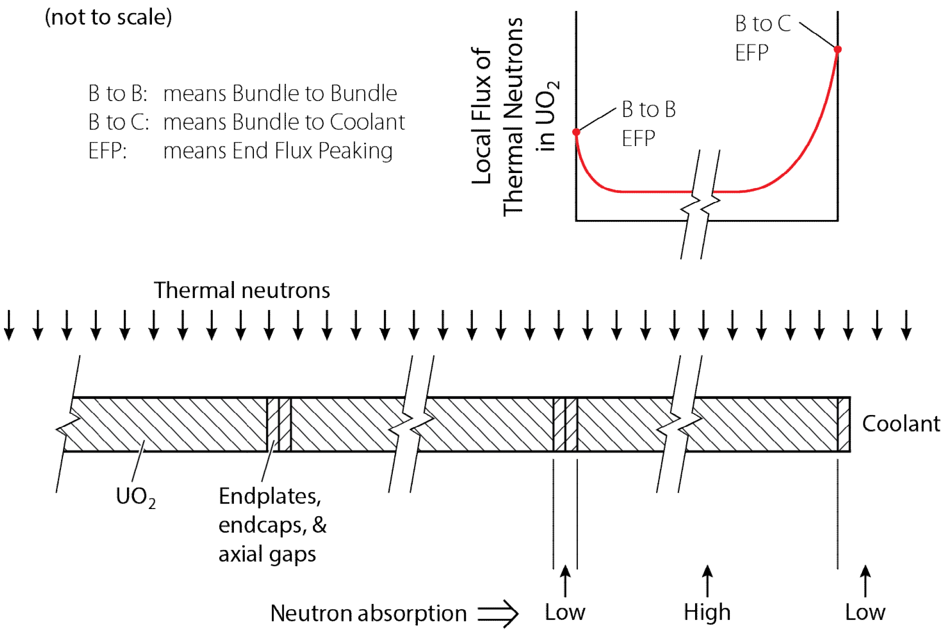
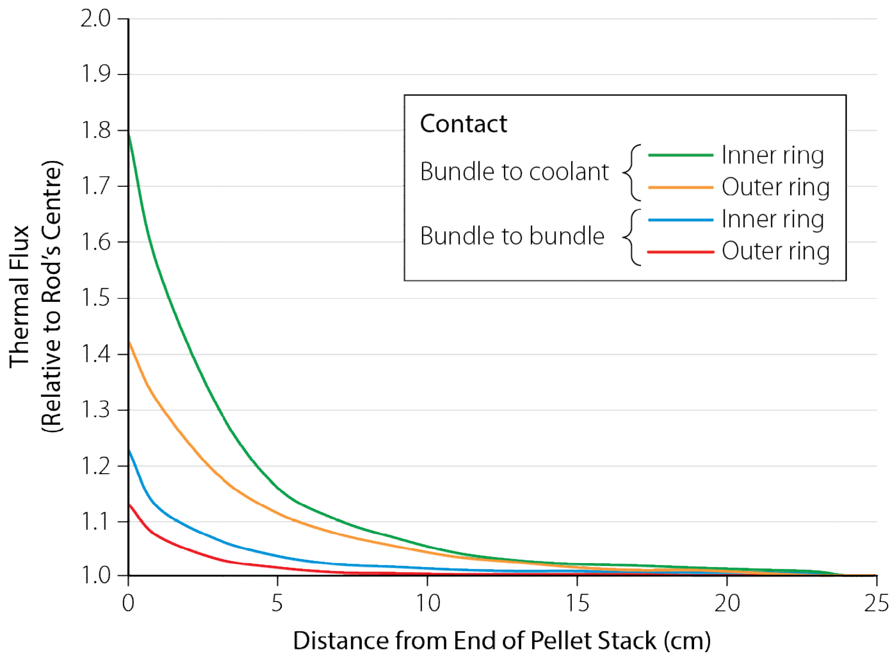


Figure 9 Concepts for end-flux peaking

End-flux peaking is even more severe at the very end of the fuel string because of the larger step change in absorption cross section at that location, the "bundle-to-coolant contact".

The *end-flux peaking factor* is defined as the ratio of the local thermal flux at the end of the fuel rod to the thermal flux at the centre of the rod. Figure 10 illustrates typical magnitudes of end-flux peaking. For bundle-to-bundle contact, thermal flux at the end of the pellet stack can be about 20% higher than that at the centre of the rod. For bundle-to-coolant contact, the local flux at the end can be considerably higher—some 80% higher than the flux at the centre [Roshd *et al*., 1978].

Flux peaking in UO_2 fuel leads directly to higher local heat generation and potentially also to higher local temperature, depending on heat transfer in the end region of the fuel element. This is called *end-temperature peaking*. The higher temperature in turn is associated with other detrimental effects such as higher thermal expansion and increased surface heat flux.



[After Roshd *et al*., 1978]

Figure 10 Illustrative peaking of flux at the end of a 28-element fuel bundle

From the perspective of local temperature, increased local heat generation can be partially, or even wholly, offset by better heat transfer in the end region. The area near the end of the pellet stack is not only cooled by radial transfer of heat, as in other locations, but also by axial transfer of heat through the endcap, as shown in Figure 11.

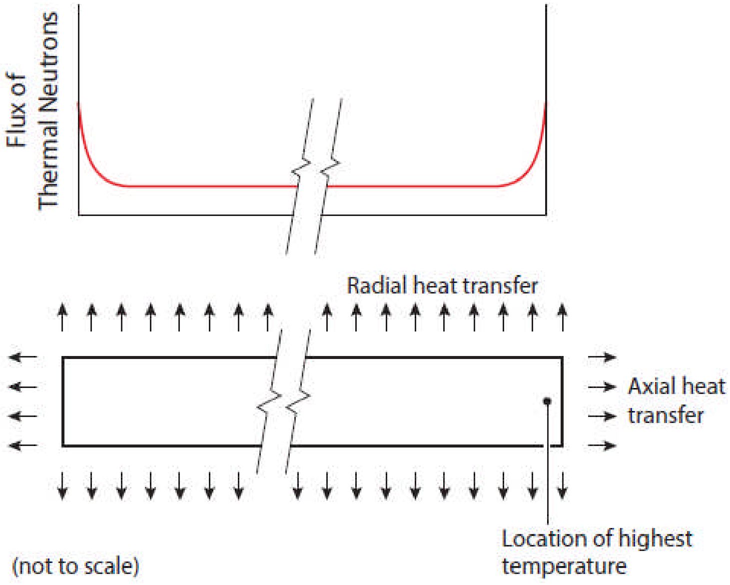
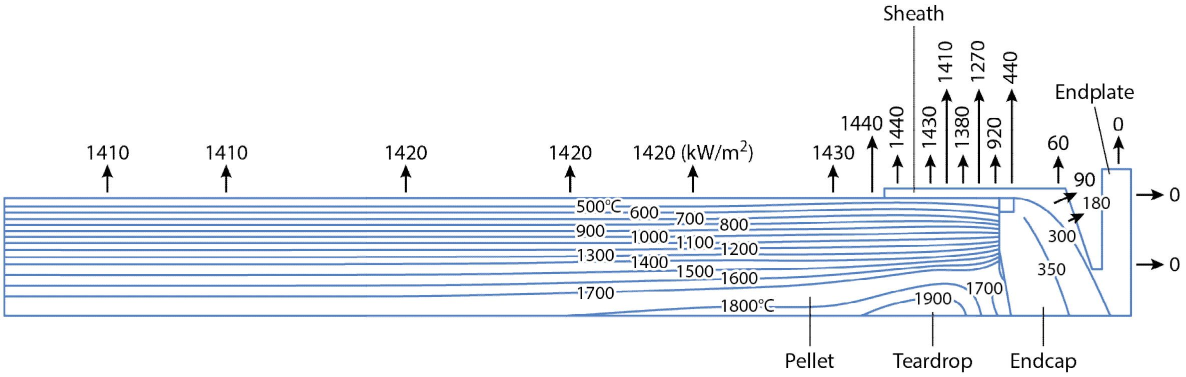


Figure 11 EFP: neutron flux and heat transfer

During bundle-to-bundle contact, the strong axial heat transfer usually depresses the local temperature at the very end of the pellet stack, despite EFP. For this reason, the peak local temperature occurs not at the very end of the fuel stack, but a short axial distance inward of it, as shown in Figure 11. Therefore, although at high power levels, the local temperature in the end region does tend to increase due to end-flux peaking, it increases less than would be expected from the amount of end-flux peaking alone.

Figure 12 illustrates a typical temperature distribution at high power in the end region during bundle-to-bundle contact [Tayal, 1989(a)]. Isotherms in the end region are shaped like teardrops, a shape determined by the net balance between local generation and local transfer of heat. Teardrop-shaped voids are indeed observed in UO_2 pellets that are irradiated at very high power levels. In Figure 12, the local thermal flux in the end region is about 10% higher than at the element's mid-point. Due to axial heat transfer, the increase in peak local temperature is limited to about 150°C in this case.



[Source: Tayal, 1989(a)]

Figure 12 End-temperature peaking: illustrative isotherm

In contrast, at low power levels and for small values of end-flux peaking, e.g., during bundle-to-bundle contact, axial transfer of heat is stronger than the local increase in heat flux. Under such conditions, the local pellet temperature does not peak at all in the end region; instead, it decreases only near the very end of the pellet stack [Tayal, 1989(a)].

Enrichment increases the degree of end-flux peaking and hence also the amount of end-temperature peaking. This tends to decrease the margin to melting in enriched fuel [Girgis *et al.*, 1990].

9.2.7 Bowing

Bowing means lateral deformation. It is of two kinds: element bowing and bundle bowing. Element bowing refers to lateral bending of individual fuel elements. Bundle bowing refers to lateral bending of the fuel bundle as a whole or of a number of elements acting in concert.

In an experiment in Ontario, intentional local dryout in a fuel element led to excessive local bowing, which led to contact between the fuel sheath and a neighbouring component. The latter in turn led to local overheating and therefore significant local oxidation in the region of contact. Failures of fuel sheaths due to bowing have also been reported in the WR1 experimental reactor in Manitoba [Veeder *et al.*, 1974].

Excessive outward bowing in the outer ring of elements has also led to "sticking" of the fuel bundle in the channel of Manitoba's WR1 reactor, creating difficulties in removing the fuel bundle [Veeder *et al.*, 1974]. The WR1 reactor was cooled by organic oil which dissociated into components, some of which were solids at operating temperature. Some of the solids deposited on the bundle surfaces, including the surfaces of matching element spacers. This increased the overall diameter of the fuel bundle and caused it to stick in the WR1 reactor channel.

In another experiment in a reactor in Ontario, excessive element bowing caused many central bearing pads to press tightly against the pressure tube for a significant duration. This trapped some coolant water between the bearing pad and the pressure tube. With time, the trapped, stagnant water boiled off. This increased the local concentration of LiOH in the coolant (LiOH is used to control coolant pH). At high concentration, an LiOH-water solution is corrosive. This led to local corrosion in the bearing pads and also likely in the pressure tube. This process is called *crevice corrosion*. There is always a potential for crevice corrosion at the bottom bearing pads; excessive bowing can lead to crevice corrosion at other locations around the bundle circumference if the bearing pads come into contact with the pressure tube. Although a small amount of crevice corrosion is acceptable, its extent needs to be limited to reasonable levels lest it threaten the integrity of the pressure tube.

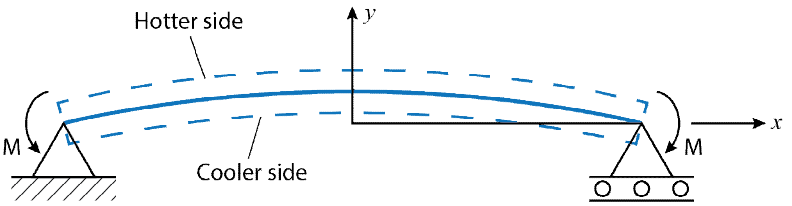
Excessive bowing can also alter the thermal hydraulics in the channel, which in turn can complicate the evaluations and confirmations of heat transfer coefficients between the sheath and the coolant. In light water reactors, excessive bowing of a fuel assembly has led to concerns about control rod movements and dryout. Therefore, bowing and dryout also have a mutual feedback cycle. For these reasons, it is important to limit bowing to acceptable levels. Bowing can increase with time, and therefore it needs to be addressed in high-burnup fuels.

When CANDU sheaths do bow, spacer pads prevent them from touching each other at their mid-planes. Likewise, bearing pads at three axial locations reduce the likelihood of a sheath touching the pressure tube in the absence of a severe local dry patch.

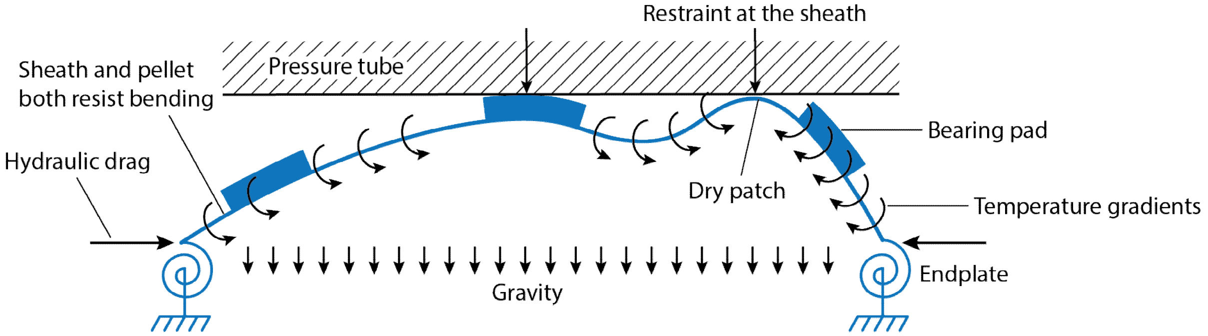
Major Driving Forces

Fabricated fuel elements are not necessarily perfectly straight, i.e., they may be bowed. In addition, during normal operations, fuel elements initially bow due to non-uniform temperature around the circumference of a fuel element and also because of mechanical loads (a list of causes of bowing is given later in this section).

Circumferential variations in temperature lead to differences in thermal expansion around the sheath circumference. The hotter side of the fuel element becomes relatively longer than the colder side. The fuel element then deforms laterally (bows) towards the longer side to accommodate the extra length. Bowing due to thermal effects is illustrated schematically in Figure 13(a).



(a) Idealization for Free Bowing



(b) Major Forces

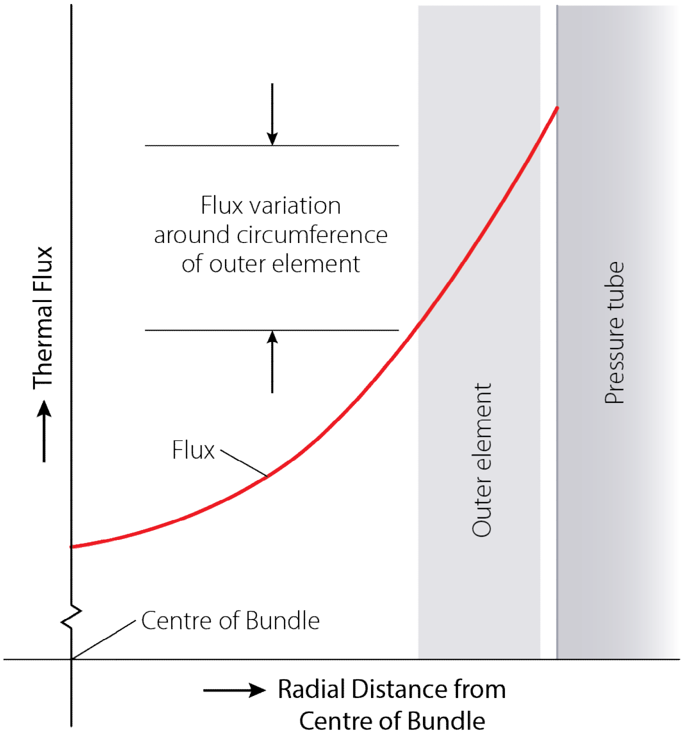
[Source: Tayal, 1989(b)]

Figure 13 Model for element bowing

Gravitational pull also acts to bend a horizontal fuel element. Once a fuel element has bowed, a moment-arm is available to the axial forces generated by hydraulic drag; this magnifies the amount of bow. Therefore, the initial bow is the net result of the following major influences:

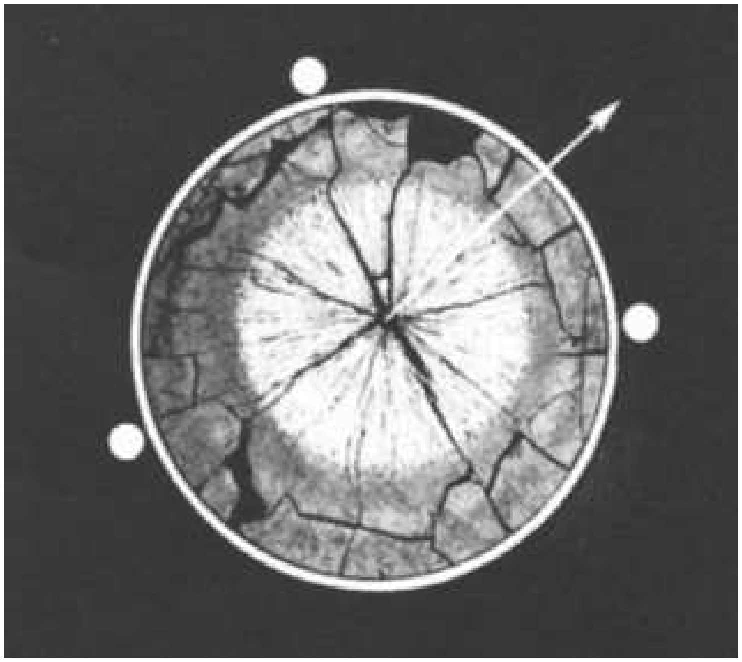
- initial as-fabricated bow;
- thermal effects, e.g.,
    - non-uniform heat transfer between the sheath and the coolant due to variations in sub-channel geometry and local flow conditions;
    - non-uniform coolant temperature due to imperfect mixing among sub-channels; local hotspots, if any (such as a dry patch);
    - asymmetric heat production due to neutron flux gradients.
- mechanical effects:
    - gravitational pull;
    - axial load, e.g., hydraulic drag.

Across the diameter of a CANDU fuel bundle, the flux profile is approximately parabolic, as shown in Figure 14. That means that in any given fuel element, heat production may not always be centred at the geometric centre of the fuel element. Figure 15 provides an illustrative example of this: the centre of grain growth has shifted in the direction of maximum neutron flux. Asymmetric heat production will necessarily set up circumferential temperature variations in the pellet and in the sheath, contributing to their bowing.



[After Veeder *et al*., 1974]

Figure 14 Illustrative circumferential variation in flux of thermal neutrons

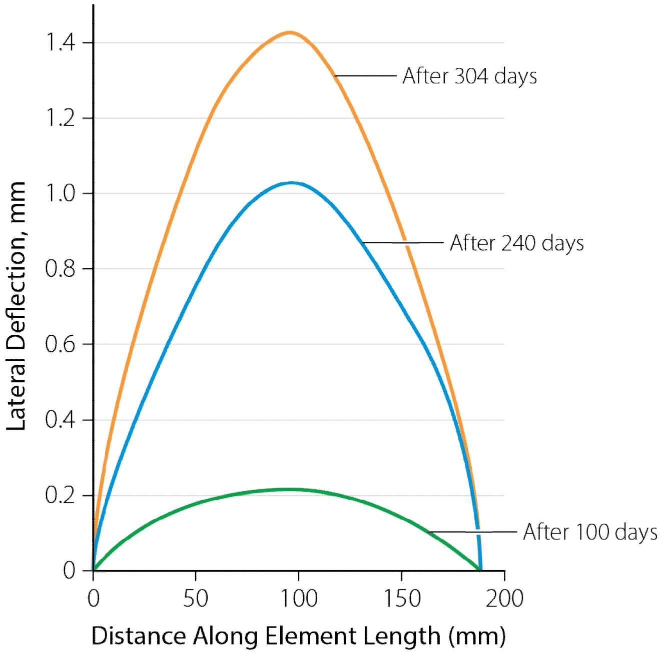


[Source: Veeder *et al*., 1974]

Figure 15 Off-centre grain growth

The loads mentioned above are resisted mainly by the flexural rigidities of the sheath and of the pellets, by endplates, and by lateral restraints (if any) from the pressure tube and from neighbouring fuel elements. Most of these loads as well as the major restraints are shown schematically in Figure 13(b).

After initial bowing has occurred, creep, additional deposits on spacer pads, or both can increase the bow further over time. For example, Figure 16 shows post-irradiation bow in a fuel element measured after 100, 240, and 304 days in the reactor under similar irradiation conditions. Note the significant increases in bow with continued irradiation. Therefore, the design process needs to consider the potential for additional bowing in extended-burnup fuels.



[Source: Veeder *et al*., 1974]

Figure 16 Increase of bowing during continued irradiation

Simplified Model for Initial Thermal Bowing

Veeder *et al*. [1974] and Tayal [1989(b)] have outlined models for bowing calculations. As a simplified illustrative example, let us consider the empty fuel sheath shown in Figure 13(a) that experiences a linear variation in temperature around its circumference and uniformly along its length. In the absence of hydraulic drag load and gravity, the equation for thermal bending of this beam can be written as [Gere *et al*., 1997]:

$$d^2y/dx^2 = \alpha\, \Delta T/d, \tag{10}$$

where y is the transverse deflection, x is the axial distance, α is the coefficient of linear thermal expansion of the sheath in the axial direction, ΔT is the difference in temperature across the diameter, and d is the diameter.

Furthermore, let us assume for simplicity that neighbouring fuel elements or pressure tubes do not impede lateral deflection of the fuel elements and that the endplates provide simple supports to the fuel element. Under these assumptions, the boundary and symmetry conditions simplify to:

$$y = 0 \text{ at } x = l/2, \text{ and } dy/dx = 0 \text{ at } x=0. \tag{11}$$

Equation (10) can be integrated under boundary and symmetry conditions (11) to yield the following equation [Gere *et al*., 1997] for unconstrained thermal bowing at the mid-plane, δ:

$$\delta = \alpha \, \Delta T \, l^2/(8d). \tag{12}$$

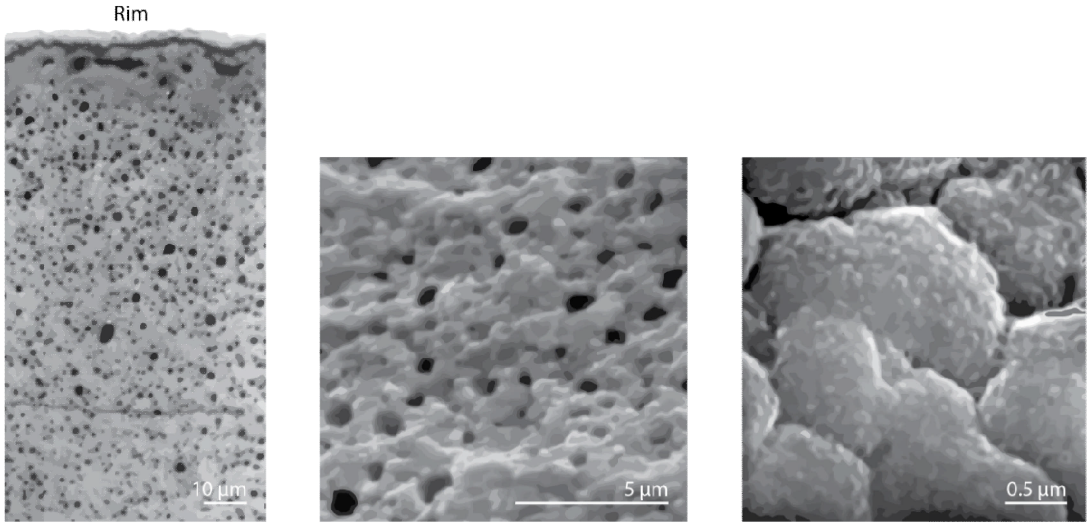
As a specific example, let us consider an empty sheath with a diameter of 13 mm, a length of 500 mm, and a coefficient of linear thermal expansion of 6.72 µm/(mm•°C). It experiences a thermal variation of 5°C around its circumference. Equation 12 predicts that in the absence of mechanical loads, its initial mid-span thermal bow would be about 0.08 mm.

The pellet operates at a much hotter temperature than the sheath, and its coefficient of thermal expansion is about twice that of Zircaloy. Hence, a pellet can have a significant impact on the thermal bow of a fuel element, especially if it is in tight diametric contact with the sheath. Mechanical loads can also have a significant impact on element bow. Their quantitative treatments are complex and beyond the scope of this chapter; interested readers can explore several detailed papers that have been published on this subject, e.g., Veeder *et al*. [1974] and Tayal [1989(b)].

9.2.8 High burnup structure

Very high local burnup can cause the formation of "high burnup structure" (HBS) at the periphery of the pellets [Baron *et al*., 2009, 2012], also called the "rim effect" and "cauliflower structure". To date, this has been observed mainly in LWR fuels because their burnups tend to be significantly higher than those of current CANDU fuels. The discussion below is based largely on Baron *et al*. [2012].

Figure 17 shows a typical HBS. It is believed to be formed by subdivision of pellet grains when the local burnup typically exceeds 55–60 MW•d/kg (or 1300–1400 MW•h/kg).

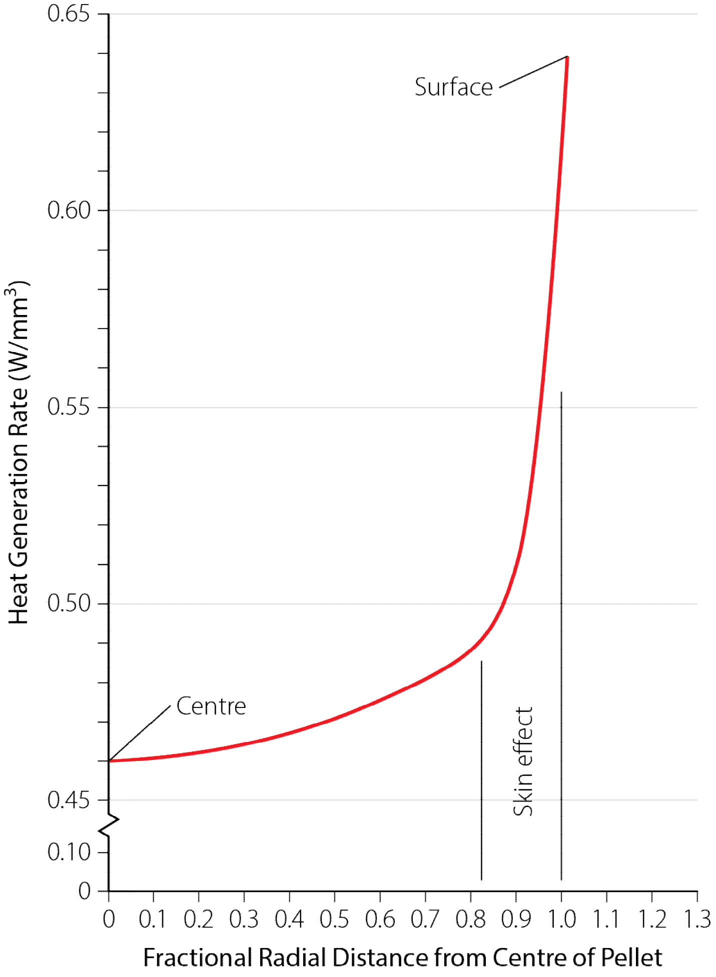


(a) Overview near the Pellet's Rim **(b) Vicinity of the Rim** **(c) Close-up of Cauliflower Structure**

[Source: Baron *et al*., 2009, 2012]

Figure 17 High burnup structure near the rim of a pellet

In UO_2 fuels, the local burnup has a sharp peak at the radial periphery of the pellet due to "neutronic self-shielding" (also called the "skin effect"), as shown in Figure 18. Therefore, in UO_2 fuels, HBS tends to be confined to a very narrow shell at the pellet's radial periphery.



[Source: Tayal, 1987]

Figure 18 Illustrative heat generation rate across the radius of the pellet

Generation of HBS is a complex process and is described in detail in many publications such as Baron *et al*. [2012]. At a very simplified level, irradiation produces, among other things, point defects and dislocation loops in the pellet. They do recover thermally, but the recovery is relatively slow in the comparatively colder regions near the periphery of the pellet. Therefore, the dislocations gradually become denser (more numerous) and more tangled. This eventually causes the crystalline structure of the pellet to become unstable, initiating a restructuring driven by the energy stored in the material. This in turn results is subdivision of grains, often into range of 0.2 to 0.5 µm diameter.

In comparison, typical grain diameter is about 10 µm in as-fabricated CANDU pellets. By considering the ratio of volumes of as-fabricated vs. subdivided grains, i.e., (about 10 µm / (0.2 to 0.5 µm))^3, one can deduce that HBS subdivides a typical grain into a few thousand smaller grains.

Compared to the parent material, UO_2 restructured by HBF is typically less dense and more viscous.

9.3 Thorium Fuel

[Main source: World Nuclear Organization, WNA, 2014]

Thorium does not have a fissile isotope, which means that it does not undergo fission by thermal neutrons. It is, nevertheless, a fertile material. This means that thorium absorbs thermal neutrons to create, or "breed", ^{233}U, which in turn undergoes fission by thermal neutrons.

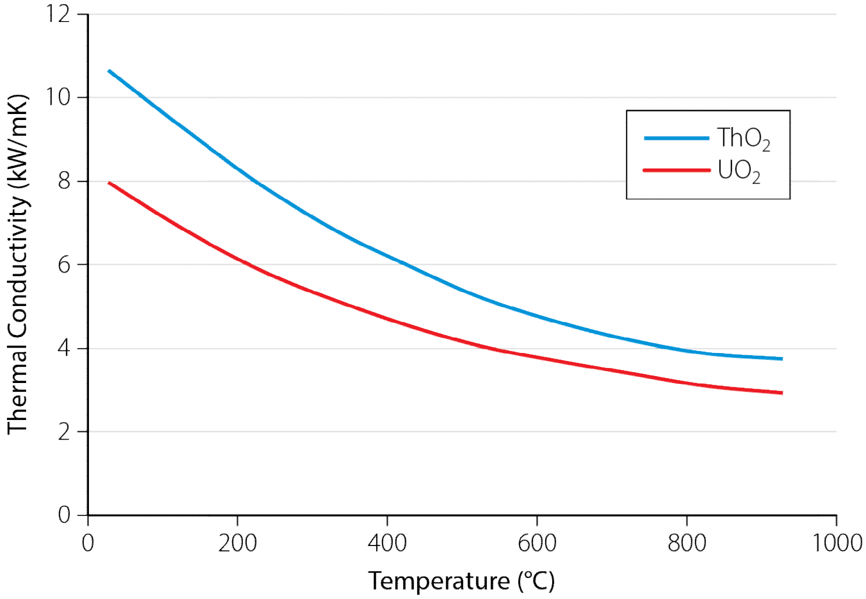
Section 2.2 summarizes the neutronics of how fertile ^{232}Th converts to fissile ^{233}U. Fission of a ^{233}U nucleus releases about the same amount of energy (200 MeV) as that of ^{235}U.

During the early years of nuclear power development, interest in thorium was prompted by a concern that economically available uranium resources would not be sufficient to support the ambitious nuclear program then envisioned. Subsequently, new uranium resources have been discovered that are thought to be sufficient to support the currently envisioned nuclear program for at least a few decades. Development aimed at exploiting the natural advantages of thorium and demonstrating the feasibility of its introduction into all known types of thermal reactors (and breeder reactors) has continued, with considerable success. Some countries are planning to introduce commercial thorium-fueled reactors, mainly for strategic reasons of energy independence. A comprehensive review of the work done internationally is provided in [IAEA, 2005], which is recommended for additional reading.

Experts estimate that thorium is about three to four times as abundant as uranium. Therefore, thorium can significantly moderate the cost of nuclear fuel in the future if it can be used to generate electricity. In addition, thorium and uranium are distributed differently among the various geographic locations on Earth. For example, countries such as India, China, and Turkey have comparatively little uranium, but substantial amounts of thorium. Such countries may potentially find thorium especially attractive for power generation.

The preferred method of using thorium is in the form of thorium dioxide, also called thoria, which is a ceramic. The reasons are analogous to those for using uranium in the form of UO_2 (see Chapter 17). The following properties of thorium dioxide pellets contribute to improving various aspects of their in-reactor performance (compared to UO_2 pellets) [Boczar, 2012]:

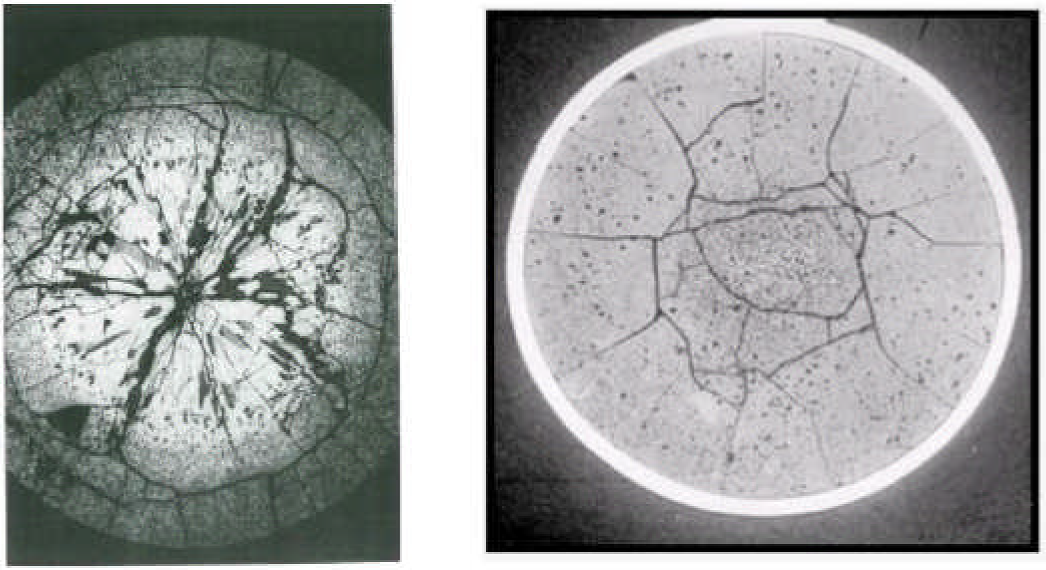
- Figure 19 shows that for a wide range of temperatures, the thermal conductivity of unirradiated thoria is about 25–35% higher than that of unirradiated UO_2 [Pillai *et al.*, 2000; MacDonald *et al.*, 1976]. Therefore, for a given power level, thorium pellets operate at lower temperatures;
- The lower temperature of ThO_2 reduces fission gas release, which reduces internal gas pressure, which in turn helps achieve higher burnup;
- The melting temperature of unirradiated ThO_2 is about 3350°C [IAEA, 2005], which is among the highest of any oxide. This compares with a melting point of about 2840°C in non-irradiated UO_2. This increases the margin to melting in ThO_2. It also introduces challenges in pellet production, requiring a higher sintering temperature;
- Thorium has only one oxidation state; therefore, it has a higher resistance to oxidation;
- Thorium is chemically inert. Therefore, it can be expected to yield a more stable form of waste that is expected to be more resistant to leaching.



Sources: [UO_2: MacDonald *et al*., 1976] [ThO_2: Pillai *et al*., 2000]

Figure 19 Thermal conductivities of unirradiated pellets

Test irradiations to date have confirmed the beneficial impacts of these properties. Compared to UO_2 fuel irradiated under similar conditions, (Th, Pu)O_2 fuel has exhibited minimal grain growth (as shown in Figure 20), benign fission gas release, and low sheath strain [Karam *et al*., 2010(b); Livingstone *et al*., 2013; Kupferschmidt *et al*., 2013]. All these observations suggest that thorium pellets indeed operate at considerably lower temperature.



Extensive Columnar Grains Equiaxed Grains

(a) UO_2 (65 kW/m) (b) $(Th,Pu)O_2$ (67 kW/m)

[Source: Kupferschmidt, 2013]

Figure 20 Illustrative structures in irradiated pellets

9.3.1 Energy from thorium

The paragraphs below are based largely on and/or reproduced from Boczar et al. [2012], IAEA [2005], and WNA [2014].

As illustrated in Section 2.2 , ^{232}Th must first be converted into ^{233}U for use in power generation. “Therefore, thorium fuel concepts require that ^{232}Th be first irradiated in a reactor before fissile material can be made available. The ^{233}U that is produced can either be chemically separated from the parent thorium fuel and recycled into new fuel, or the ^{233}U may be usable *in-situ* in the same fuel form”.

“Thorium fuels therefore need a fissile material as a ‘driver’ so that a chain reaction can be initiated and maintained. The only fissile driver options are ^{233}U, ^{235}U, or ^{239}Pu, none of which is easy to supply”.

A number of configurations are possible for arranging thorium and a fissile driver in a reactor; some have been used successfully to demonstrate the feasibility of thorium implementation. In a CANDU reactor, three illustrative configurations are the following [Boczar, 2012]:

- Homogeneous fuel. In this option, each pellet contains a mixture of thorium and the fissile material.
- Heterogeneous “mixed bundles”. Such bundles contain thorium and the fissile material in separate fuel elements.
- Heterogeneous “mixed channels”. In this option, each bundle contains either only thorium or only the driver fuel. Each channel contains only one type of bundle.

“In the heterogeneous fuel arrangements, the high fissile (and therefore higher-power) fuel zone is called the *seed* region, and it is physically separated from the fertile (low-power) thorium part of the fuel, called the *blanket*".

"It is possible—but quite difficult—to design thorium fuels that produce more ^{233}U in thermal reactors than the fissile material they consume. Thermal breeding with thorium is only really possible if neutron economy in the reactor is very good (i.e., low neutron loss through escape or parasitic absorption). The possibility to breed significant amounts of fissile material in slow (thermal) neutron systems is unique to thorium-based fuels and is not possible with uranium fuels. Irrespective of whether breeding is achieved, more practical thorium fuel cycles in a CANDU reactor offer a significant reduction in uranium requirements".

"Another distinct option for using thorium is as a 'fertile matrix' for fuels containing plutonium (and even other transuranic elements like americium). No new plutonium is produced from the thorium component, unlike for uranium fuels, and so the level of net consumption of this metal is rather high".

"In fresh thorium fuel, all the fissions (thus power and neutrons) derive from the driver component. As the fuel operates, the ^{233}U content gradually increases, and it contributes more and more to the power output of the fuel. The ultimate energy output from ^{233}U (and hence indirectly from thorium) depends on numerous fuel design parameters, including fuel burnup attained, fuel arrangement, neutron energy spectrum, and neutron flux (affecting the intermediate product protactinium-233, which is a neutron absorber)".

Specific thorium cycles can be designed to suit a country's specific circumstances.

9.3.2 Margin to melting

As an illustrative example, consider two neighbouring fuel elements, one fueled with UO_2, the other with ThO_2. Both are operating at 60 kW/m; the coolant temperature is 300°C. If end-temperature peaking is ignored, what are the margins to melting in the two fuel elements?

To work this out, we can use the theoretical background given in Chapters 6 and 7 about the temperature distribution within a fuel element, along with some assumptions to simplify this illustrative calculation. Recall that if other parameters are the same, to a first approximation, the temperature difference between the centre and the surface of the pellet is inversely proportional to the thermal conductivity of the pellet.

Guided by the previous illustrative examples in this book, assume that the UO_2 fuel element has a central temperature of about 1900°C and that the melting point of unirradiated UO_2 is about 2840°C. Therefore, in the UO_2 fuel element, the thermal margin is about 2840 – 1900 = 940°C.

The peak operating temperature in the ThO_2 pellet can be approximated as follows:

- Guided by the previous illustrative examples, assume that the temperature increase between the coolant and the pellet surface is about 100°C.
- Therefore, the temperature at the surface of the pellet is: 300 + 100 = 400°C.
- In the UO_2 pellet, the temperature increase from the surface to the centre of the pellet is 1900 – 400 = 1500°C.
- Guided by Figure 19, assume that to a first approximation, the thermal conductivity of ThO_2 is about 30% higher than that of UO_2. Therefore in a thoria pellet operating at the

same power, the temperature increase from the surface to the centre of the pellet is about 1154°C (=1500°C / (1 + 0.3)).

- Therefore, the temperature at the centre of the thoria pellet is 400 + 1154 = 1554°C.
- The melting point of ThO_2 is about 3350°C [IAEA, 2005], and therefore the margin to melting in the ThO_2 pellet is about 3350 – 1554 = 1796°C.

Therefore, for the same operating power of about 60 kW/m, the margin to melting in ThO_2 fuel is 1796°C, which is significantly higher than in UO_2 fuel.

9.3.3 Internal gas pressure

Pellet temperature is a main driver of fission gas release. Above a threshold temperature, fission gas release increases rapidly with local temperature.

In the preceding illustrative example, the UO_2 fuel operates a few hundred degrees above the threshold temperature for fission gas release. This results in significant fission gas release in UO_2 fuel.

Assuming that the threshold temperature is similar in ThO_2 and UO_2, the ThO_2 fuel described above operates much closer to the threshold temperature for fission gas release. This means that the ThO_2 pellet will release much less fission gas.

The above illustrative examples point to some of the large benefits in operating performance that are possible with thorium. Along with thorium's impressive benefits, we do also need to consider, as noted earlier, the higher costs of some thorium cycles.

9.4 Mixed Oxide (MOX) Fuel

MOX stands for Mixed Oxide, which means that the fuel contains a mixture of uranium and plutonium (Pu).

Section 2.2 summarizes the neutronics of Pu generation and fission. A slow neutron can fission ^{239}Pu into mainly barium and strontium. This releases 207 MeV of energy, which is very similar to the energy released by fission of ^{235}U (202 MeV).

In a real-world application of "swords-to-plowshares", plutonium extracted from dismantled nuclear warheads can be used to generate electricity as MOX fuel. Plutonium can also be obtained by reprocessing used LWR fuel. Although feasible, reprocessing of used CANDU fuel is seen as impractical because of the relatively small quantity of plutonium (and fissile uranium) contained in used CANDU fuel. LWRs are already being used to dispose of military plutonium, and MOX is a commercial reality in European LWRs. More detailed overviews of the MOX cycle are described in many publications such as [Boczar, 2012; Dyck *et al*., 2005].

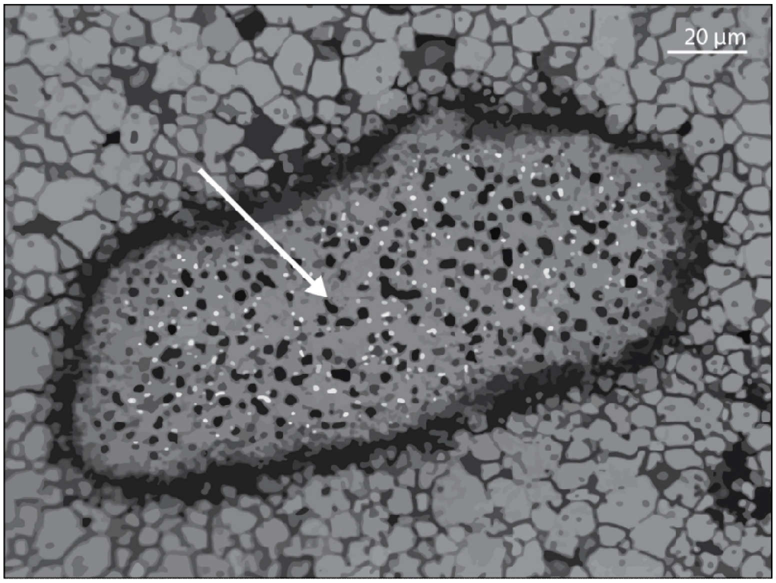
One interesting aspect of the common variant of MOX fuel is its tendency to form "agglomerates" arising out of non-homogeneities in the mixture of UO_2 and PuO_2. The amount of agglomerates depends on the specific manufacturing process, as explained below.

Reasonably homogeneous mixtures of UO_2 and PuO_2 can be produced by vibratory milling of UO_2 and PuO_2 powders in the proportions desired in the final fuel. The resulting homogeneity is expected to yield the best performance in MOX fuel. However, this process is quite time- and resource- consuming. Alternative manufacturing processes which produce less homogenous mixtures require less effort, but they also contain agglomerates, meaning that trade-offs must be considered [Harrison *et al*., 2010].

For purposes of illustration, two such alternative processes are: [Harrison *et al*., 2010]:

> Alternative Method # 1: A master mix with high Pu concentration is prepared by vibratory milling of the UO_2 and PuO_2 powders together. The required amount of this "master mix" is added to additional UO_2 powder to achieve the desired final Pu concentration. This approach results in discrete "islands"—also called agglomerates—of master mix contained in the UO_2 matrix.
>
> Alternative Method # 2: Same as Alternative Method # 1 above except that only PuO_2 is milled (rather than a mixture of PuO_2 plus UO_2). The resulting powder is then added to UO_2, resulting in nearly pure PuO_2 "islands" distributed within the UO_2 matrix.

The two alternative fabrication processes above create islands of relatively higher plutonium concentration. The islands undergo relatively higher amounts of fission and produce correspondingly higher quantities of fission gases. Figure 21 shows a typical island in a fuel element that was fabricated using Alternative Method # 1. The arrow points to the area of high fission. Similar areas were also observed in pellets produced by the second alternative method.



Arrow points to the high fission area.

[Source: Harrison *et al*., 2010]

Figure 21 High fission area (HFA) in an irradiated MOX fuel pellet

Within an island, the pellet behaves as if it were operating at significantly higher local power. This results in larger local fission gas release. This is partly offset in the remaining parts of the pellet, which behave as if they are operating at a lower local power. Because fission gas release increases non-linearly with local temperature, the overall fission gas release from non-homogeneous pellets is greater than from homogeneous pellets. However, there is significant scatter in both the UO_2 and the MOX data, and the limited MOX data to date in Canada are in the general range of scatter of the UO_2 data.

Non-homogeneous MOX fuel also shows larger post-irradiation sheath strain than homogeneous fuel.

Note that despite the higher gas release and strain noted above, the overall performance of MOX fuel is quite acceptable in many LWRs worldwide [IAEA, 2003].

9.5 Breeder Fuel

[Main source: Wikipedia, 2015]

In a typical thermal reactor fuel cycle, energy is extracted from ^{235}U as well as from ^{239}Pu, the latter being bred from ^{238}U. Because some fraction of the starting ^{235}U remains unused when the fuel is discharged, less than 1% of the available energy is currently extracted from mined uranium.

The dominant isotope—over 99%—in natural uranium is ^{238}U, which is not fissile. However, it is fertile, which means that it can be converted into a fissile isotope (^{239}Pu) through fast neutrons, and energy can then be extracted from it as well.

^{238}U can be sourced from virgin uranium, from the recycled uranium from reprocessing irradiated fuel, or from depleted uranium from enrichment plant tails.

As noted earlier, some amount of fertile ^{238}U is routinely converted to fissile ^{239}Pu in CANDU reactors. One measure of a reactor's performance is the *conversion ratio*—the ratio of the rate at which fissile nuclides are produced in the entire reactor by conversion, to the rate at which fissile nuclides are consumed through radiative capture and fission. All practical nuclear reactor fuels experience some degree of conversion.

As long as any amount of a fertile material exists within the neutron flux of the reactor, some new fissile material is always being created. For example, commonly used light water reactors have a conversion ratio of approximately 0.6. Pressurized heavy water reactors (PHWR) running on natural uranium have a conversion ratio of 0.8. In a breeder reactor, the conversion ratio is greater than one. "Breakeven" occurs when the conversion ratio is exactly one, and the reactor produces exactly as much fissile material as it uses. (This is the basis of the so-called self-sufficient equilibrium thorium CANDU reactor, which at equilibrium contains as much ^{233}U in the used fuel as is in the fresh fuel, accounting for reprocessing losses.)

Historically, breeder reactor development has focussed on reactors with low breeding ratios, from 1.01 for the Shippingport reactor running on thorium fuel and cooled by conventional light water to over 1.2 for the Russian BN-350 liquid-metal-cooled reactor. Theoretical models of breeders with liquid sodium coolant flowing through tubes inside fuel elements ("tube-in-shell" construction) show that breeding ratios of at least 1.8 are possible.

Some experts have estimated that thousands of years' worth of ^{238}U is available world-wide for use in fast breeder reactors.

10 Synergy between CANDU Reactors and LWRs

As mentioned in previous sections, the superior neutronic characteristics of CANDU reactors can be used for a variety of synergies with LWRs, e.g.:

- extracting additional energy from used LWR fuel, e.g., through RU/NUE, through ex-

tended burnup, through DUPIC, or through plutonium recycling (e.g., MOX), and/or
- destroying some of the long-lived actinides removed from spent LWR fuels through reprocessing.

Based on the fissile content of used LWR fuel, one can calculate the number of LWRs from which the used fuel can be used as fuel by a single CANDU reactor [Boczar *et al*., 1996].

The quantities used in the example below are not necessarily actual values for a specific CANDU or LWR reactor, although they are reasonably representative of the two reactor types and are used to illustrate how the actual quantities would be used to determine the synergy between an actual LWR and an actual CANDU reactor by substituting real, specific values.

LWR reactor:

- capacity: 1000 MW(e)
- total uranium in core: 80,000 kgU
- amount of U refueled annually: 20,000 kg (one-quarter of the core replaced annually)
- ^{235}U contained in spent fuel (annually): 180 kg (0.9% ^{235}U in total U).

CANDU reactor:

- capacity: 1000 MW(e)
- total uranium in core: 120,000 kg (6000 bundles containing 20 kg U each)
- annual requirements:
    - annual refuelling: 140,000 kg of U (7000 bundles);
    - annual ^{235}U requirements: approximately 1,000 kg ^{235}U (0.71% ^{235}U in total of 140,000 kg U).

Synergy:

- annual ^{235}U arising from spent LWR fuel: 180 kg
- annual ^{235}U requirements for CANDU reactor of equal electrical capacity: 1,000 kg
- proportion of ^{235}U used in CANDU fuel that is derived from LWR arising: 0.73% (see Section 9.1 above); the remainder of ^{235}U is derived from depleted U from tails of enrichment plants.
- CANDU reactor's annual ^{235}U requirements supplied by LWR: 730 kg (73% of 1,000).
- Number of LWRs required to supply ^{235}U for CANDU reactors of equal electrical capacity (rounded to the nearest whole number): 4 (= 730/180).

The example above assumes that only ^{235}U arisings would be used. If plutonium were also used, the proportion of LWR per CANDU reactor would be accordingly reduced.

11 Closure

In summary, some illustrative fuels and fuel cycles are the natural uranium cycle, natural-uranium equivalent fuel, extended burnup fuel, thorium, mixed oxide fuel, DUPIC fuel, low void reactivity fuel, actinide burning fuel, and breeder fuel.

The natural-uranium fuel cycle that is currently used in CANDU power reactors is simple, reliable, and low-cost. It is also very amenable to localization, which promotes national energy independence.

Alternative fuel cycles offer several potential strategic advantages, such as significantly extending humanity's energy resources, reducing the initial capital cost of a power plant, increasing the lifespan of a CANDU power plant, increasing the thermal efficiency of the steam cycle, reducing the coolant void reactivity, reducing the volume of spent fuel, and increasing the power output from a fuel bundle and the reactor.

Many alternative fuel cycles result in more expensive fuel, additional challenges in the front end, enhanced challenges during the service period, and reduced challenges in the back end.

Therefore, a careful assessment of net benefit is required before implementing any particular fuel cycle.

12 Problems

Section 1 – Introduction

Q1.1 – Why would one consider fuel cycles other than the once-through cycle?

Section 2 – Neutronics of Fuel Cycles

Q2.1 – Define fissile nuclides and give examples. Which is the only naturally occurring fissile nuclide?

Q2.2 – Define fissionable nuclide and provide examples.

Q2.3 – Define fertile nuclides and provide examples.

Q2.4 – What is the conversion ratio as related to nuclides?

Q2.5 – In the context of conversion ratio, what is breeding and breeding gain?

Q2.6 – What is burnup as applied to nuclear fuel?

Section 3 – Overview of Some Possible Fuels and Fuel Cycles

Q3.1 – List the fuel cycles which are used, or can be considered for use, in CANDU rectors and provide a brief rationale for using each.

Section 4 – Key Drivers for Advanced Fuel Cycles

Q4.1 – List the principal advantages of the once-through natural uranium (NU) fuel cycle used in CANDU compared to fuel cycles using enriched uranium.

Q4.2 – With reference to the three parts of the fuel cycle, provide a general assessment of the advantages or disadvantages of alternative cycles compared to the once-through CANDU-NU fuel cycle.

Q4.3 – Compared to the NU fuel cycle used in CANDU, many alternative fuel cycles tend to increase the cost of fuel and also pose various degrees of additional technical challenge. Nevertheless, they are considered for use in CANDU because they offer other advantages. List these advantages and provide a brief explanation of each.

Section 5 – Resource Extension: A Major Incentive

Q5.1 – Compared to LWRs, CANDU reactors can better use alternative fuel cycles, principally because of their superior neutron efficiency. How much more efficient is CANDU than an LWR in the use of uranium?

Q5.2 – In current CANDU reactors, peak neutron efficiency occurs at about 1.2% enrichment. If 1.2% enriched uranium is used in CANDU reactors, how long will the planet's identified uranium resources (mentioned above) last?

Section 6 – Other Considerations

Q6.1 – List the considerations that have to be taken into account when deciding whether or not to "recycle" spent fuel.

Section 7 – Pertinent Features of the CANDU Reactor

Q7.1 – CANDU reactors can use alternative fuel cycles principally because of their superior neutron efficiency. List the seven features of the CANDU reactor that enable it to have superior neutron efficiency.

Q7.2 – Explain how the design of CANDU fuel facilitates the adoption of alternative fuel cycles.

Section 8 - Natural Uranium Fuel Cycle

Q8.1 – List the four methods by which natural uranium can be obtained.

Q8.2 – Unlike other types of fuels (e.g., coal, gas, or oil), as-mined uranium ore cannot be fed directly into a nuclear power station. Explain why.

Q8.3 – Produce a simplified flow sheet of the processes involved in the manufacture of CANDU fuel bundles.

Q8.4 – Why is as-received uranium powder subjected to two physical processes, compaction and granulation?

Q8.5 – What are the three process parameters that the operator controls during pellet sintering?

Q8.6 – Why are pellets ground after sintering?

Q8.7 – Pellet stack length is the principal means by which axial clearance is maintained inside the finished fuel element. What are the other means?

Q8.8 – Why do the pellets at the ends of the pellet stack have a reduced diameter?

Q8.9 – List the four features of uranium dioxide pellets that render their production challenging.

Q8.10 – Similarly to the pellet production stream, the Zircaloy component stream has its own unique CANDU-related challenges, the most unique and critical being welding of endcaps to sheath sub-assemblies. Explain why endcap welding is a critical and challenging process.

Q8.11 (a) – Identify five design and/or process parameters that must be controlled to achieve a good endcap weld; **Q8.11 (b)** for bonus marks, identify an additional five parameters; **Q8.11 (c)** for additional bonus marks, identify five more parameters.

Q8.12 – For one unit of energy produced, current CANDU reactors use five to six times the mass of fuel as is used in LWRs. Despite that relative disadvantage, fuelling cost per unit energy has always been much lower for CANDUs than for LWRs. List the three reasons for this.

Section 9 – Details of Selected Alternative Fuels and Fuel Cycles

Q9.1 – A fuel manufacturer has purchased REU fuel containing 1% ^{235}U and DU fuel containing 0.4% ^{235}U. He wishes to blend them to obtain NUE fuel. What should be the fraction of REU fuel in the blend?

Q9.2 – List the benefits which can be derived from using enriched fuel in a CANDU.

Q9.3 – List the commercial and technical aspects that must be considered in increasing element rating, burnup, and bundle power by using enrichment.

Q9.4 – Why is it important to control fuel-element internal gas pressure, and what can fuel designers do to achieve this?

Q9.5 – A power ramp of 15 kW/m is contemplated in CANLUB fuel at a burnup of 400 MWh/kgU. The pre-ramp power is 20 kW/m. Is this fuel element expected to survive the power ramp?

Q9.6 – List at least two reasons for keeping sheath corrosion below the allowable limit.

Q9.7 – List the reasons for the reduced amount of corrosion on CANDU fuel compared to PWR fuel

Q9.8 – In the example of Section 9.2.3, what is the expected thickness of waterside corrosion in a CANDU sheath at 800 MWh/kgU?

Q9.9 – What causes end-flux (and end-temperature) peaking?

Q9.10 – Pellet/sheath interaction causes an unrestrained thermal bow of 0.6 mm in a fuel element that has a diameter of 13 mm and a length of 500 mm. What is the equivalent temperature variation around its circumference?

Q9.11 – What are the two main reasons for considering the use of thorium in place of uranium?

Q9.12 – Which is the fissile isotope in thorium?

Q9.13 – The preferred method of using thorium is in the form of thorium dioxide, also called thoria, which is a ceramic. The reasons are analogous to those for using uranium in the form of UO_2 (see Chapter 17). List the properties of thorium dioxide pellets that contribute to improving their in-reactor performance compared to UO_2 pellets.

Q9.14 – Describe three ways that thorium and the fissile driver can be arranged to power a CANDU reactor.

Q9.15 – (a) What are the two possible sources of plutonium that can be used in CANDU? (b) Is reprocessing of spent CANDU fuel a likely source of plutonium?

Q9.16– What is a common challenge in the production of MOX?

Q9.17 – Why are "agglomerates" undesirable in MOX pellets?

Section 10 – Synergy between CANDU Reactors and LWRs

Q10.1 - Assume that the waste of LWRs contains 0.9% ^{235}U and 0.6% Pu. Further assume that DU can be procured with ^{235}U content at any value in the range from 0.2%^{235}U to 0.4%^{235}U. We wish to use the ^{235}U and Pu from LWR to feed a CANDU of equal capacity, after blending with DU. If each CANDU and each LWR is 1000 MW(e), determine the required content of ^{235}U in DU, such that a single LWR will feed one CANDU of equal capacity. Solving this problem highlights and reinforces six (6) aspects covered in this chapter. What are they?

13 References

D. Barber, D. H. Lister, 1982. "Chemistry of Water Circuits in CANDU Reactors", Proceedings, Symposium on Water Chemistry, Report IAEA-SM-264/15 and AECL-8217, 1982.

D. Baron, M. Kinoshita, P. Thevenin, R. Largenton, 2009. "Discussion about HBS Transformation in High-Burnup Fuels", Nuclear Engineering and Technology, Vol. 41, No. 2, pp. 199-214, March 2009.

D. Baron, L. Hallstadius, 2012. "Fuel Performance of Light Water Reactors (Uranium Oxide and MOX)"; Chapter 40 in "Comprehensive Nuclear Materials", book edited by R. Konings, Elsevier, ISBN-9780080560274, 2012.

Z. Bhatti, B. Hyland, G. Edwards, 2013. "Minor Actinide Transmutation in Thorium and Uranium Matrices in Heavy Water Moderated Reactors", Proceedings, Global 2013, Salt Lake City, Utah, U.S.A., September 29–October 3, 2013.

P. G. Boczar, J. D. Sullivan, H. Hamilton, Y. O. Lee, C. J. Jeong, H. C. Suk, C. Mugnier, 1993. "Recovered Uranium in CANDU—A Strategic Opportunity", Proceedings of International Nuclear Conference and Exhibition (INC 93), Toronto, Ontario, Canada. October 3-6, 1993.

P. G. Boczar, P. J. Fehrenbach, D. A. Meneley, 1996. "CANDU Fuel Cycle Options in Korea", Proceedings, KAIF/KNS Annual Conference, Seoul, Korea, April 1996; also Atomic Energy of Canada Limited, Report AECL-11586, 1996.

P. G. Boczar, J. D. Sullivan, 2004. "Low Void Reactivity Fuel", Proceedings, 25th Annual CNS Conference, Toronto, 2004.

P. G. Boczar, J. T. Rogers, D. H. Lister, 2010. "Considerations in Recycling Used Natural Uranium Fuel from CANDU Reactors in Canada", Proceedings, 31st Annual Conference of the Canadian Nuclear Society, Montreal, Quebec, Canada, May 24–27, 2010.

P. Boczar, 2012. "CANDU Nuclear Reactor Design, Operation, and Fuel Cycles", Chapter 11, "Nuclear Fuel Cycle Science and Engineering", Woodhead, 2012.

B. Cox, 1985. "Assessment of PWR Waterside Corrosion Models and Data", EPRI Report NP-4287, 1985.

G. R. Dyck, T. Mochida, T. Fukasawa, 2005. "Application of Fluoride Volatility to the Recycling of LWR Spent Fuel into CANDU", Proceedings, Global 2005, Paper number 493, Tsukuba, Japan, October 9–13, 2005.

M. R. Floyd, 2001. "Extended-Burnup CANDU Fuel Performance", Proceedings, International

Conference on CANDU Fuel, CNS, Kingston. Also Atomic Energy of Canada Limited, Report AECL-CONF-01135, 2001.

W. J. Garland, 2005. EP4D03 course notes, Chapter: Core Composition Changes, McMaster University, http://www.nuceng.ca/ep4d3/ep4d3home.htm, 2005.

F. Garzarolli, W. Jung, H. Schoenfeld, A. M. Garde, G. W. Parry, P. G. Smerd, 1982. "Waterside Corrosion of Zircaloy Fuel Rods", Electric Power Research Institute, U.S.A., Report EPRI-NP-2789, 1982.

J. M. Gere, S. P. Timoshenko, 1997. "Mechanics of Material", PWS, 4th edition, 1997.

S. Girgis, N. Singhal, M. Tayal, 1990. "Influences of Load Following Partial Dryout and End-Flux Peaking on Temperature Distributions in CANDU Fuel", Proceedings, 11th Annual Conference, Canadian Nuclear Society, June 3–6, 1990.

N. F. Harrison, T. Janathasing, F. C. Dimayuga, 2010. "Post-Irradiation Examination of MOX Fuel with Varying Plutonium Homogeneity", Proceedings, 11th International Conference on CANDU Fuel, Sheraton Fallsview Hotel and Conference Centre, Niagara Falls, Ontario, Canada, CW-124920-CONF-004, October 17–20, 2010.

I. Hastings, P. G. Boczar, C. J. Allan, M. Gacesa, 1991. "Synergistic CANDU-LWR Fuel Cycles", Korea Nuclear Society, Annual Conference, 1991. Also Atomic Energy of Canada Limited, Report AECL-10390, 1991.

B. Hyland, B. Gihm, 2010. "Scenarios for the Transmutation of Actinides in CANDU Reactors", Proceedings, 18th International Conference on Nuclear Engineering (ICONE18), Paper ICONE18-30123, Xi'an, China, May 17–21, 2010.

IAEA, 2003. "Status and Advances in MOX Fuel Technology", International Atomic Energy Agency, Vienna, Austria, Technical Report Series, Report TRS 415, 2003.

IAEA, 2005. "Thorium Fuel Cycle", International Atomic Energy Agency, Report IAEA-TECDOC-1450, 2005.

IAEA, 2010. "Review of Fuel Failures in Water-Cooled Reactors", International Atomic Energy Agency, Vienna, IAEA Nuclear Energy Series, report number NF-T-2.1, ISBN 978-92-0-102610-1. 2010.

Y. Jioa, D. Li, M. Chen, Z. Meng, J. Wang, C. Cottrell, S. Kuran, 2009. "Test Irradiation of Recycled Uranium in Chinese CANDU Reactors", Proceedings, 30th Annual Conference of the Canadian Nuclear Society, Calgary, Alberta, Canada, May 31–June 3, 2009.

M. Karam, N. Harrison, J. Montin, 2010(a). "Post-Irradiation Examination of DUPIC Fuel", Proceedings, 11th International Conference on CANDU Fuel, Sheraton Fallsview Hotel and Conference Centre, Niagara Falls, Ontario, Canada, October 17–20, 2010.

M. Karam, F. C. Dimayuga, J. Montin, 2010(b). "Post-Irradiation Examination of CANDU Fuel Bundles Fueled with (Th, Pu)O_2", Proceedings, 11th International Conference on CANDU Fuel, Sheraton Fallsview Hotel and Conference Centre, Niagara Falls, Ontario, Canada, October 17–20, 2010.

W. C. H. Kupferschmidt, 2013. "Advanced Fuel Development at AECL: What does the Future

Hold for CANDU Fuels/Fuel Cycles?", Proceedings, 12th International Conference on CANDU Fuel, Canadian Nuclear Society, Kingston, Ontario, Canada, September 15–18, 2013.

J. R. Lamarsh, 1996. "Introduction to Nuclear Reactor Theory", Addison-Wesley, 1966.

S. J. Livingstone, M. R. Floyd, 2013. "Thoria Irradiation and Post-Irradiation Examination Experience at AECL", Proceedings, 12th International CANDU Fuel Conference, CNS, Kingston, 2013.

P. E. MacDonald, L. B. Thompson (eds.), 1976. "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behaviour", MATPRO-Version 09, TREE-NUREG-1005, 1976.

D. A. Meneley, 2006. "Transition to Large-Scale Nuclear Energy Supply", Proceedings, 27th Annual CNS Conference, Toronto, 2006.

P. Ottensmeyer, 2012. "CANDU Fuel Waste Re-Used, Recycled, Eliminated: $48 Trillion Carbon-Free Electricity via Fast-Neutron Reactors", Engineering Dimensions, Professional Engineers of Ontario, Volume 33, No. 4, July/August 2012.

R. D. Page, 1976. "Canadian Power Reactor Fuel", Atomic Energy of Canada Limited, Report AECL-5609 (1976).

C. G. S. Pillai, P. Raj, 2000. "Thermal Conductivity of ThO_2 and $Th_{0.98}U_{0.02}O_2$", Journal of Nuclear Materials, Vol. 277, pp. 116-119, 2000.

P. Reid, M. Gacesa, M. Tayal, 2008. "The ACR-1000 Fuel Bundle Design", Proceedings, 10th International Conference on CANDU Fuel, Canadian Nuclear Society, 2008.

M. H. M. Roshd, H. C. Chow, 1978. "The Analysis of Flux Peaking at Nuclear Fuel Bundle Ends using PEAKAN", Atomic Energy of Canada Limited, Report AECL-6174, 1978.

M. Tayal, 1987. "Modelling CANDU Fuel under Normal Operating Conditions: ELESTRES Code Description", Atomic Energy of Canada Limited, Report AECL-9331, February 1987.

M. Tayal, 1989(a). "The FEAT Finite Element Code to Calculate Temperatures in Solids of Arbitrary Shapes", Nuclear Engineering and Design, Vol. 114, pp. 99–114, 1989.

M. Tayal, 1989(b). "Modelling the Bending/Bowing of Composite Beams such as Nuclear Fuel: The BOW Code", AECL-9792, Nuclear Engineering and Design, Vol. 116, pp. 149–159, North-Holland, Amsterdam, 1989.

M. Tayal, K. D. Hallgrimson, R. Sejnoha, P. N. Singh, R. DaSilva, 1993. "Elastic-Plastic Stress Distributions near the Endcap of a Fuel Element", Proceedings, 4th International Conference on Simulation Methods in Nuclear Engineering, Canadian Nuclear Society, Montreal, Canada. Also Atomic Energy of Canada Limited, Report AECL-10646, June 1993.

M. Tayal, B. Wong, Y. Shudoh, 1995. "Effect of Radial Power Profile on Endplate Integrity", Proceedings, 4th International Conference on CANDU Fuel, Canadian Nuclear Society, Pembroke, Canada. Also Atomic Energy of Canada Limited, Report AECL-Conf-00452, 1995.

J. Veeder, M. H. Schankula, 1974. "Bowing of Pelletized Fuel Elements: Theory and In-Reactor Experiments", Nuclear Engineering and Design, Vol. 29, pp. 167–179, 1974.

Wikipedia, 2015. "Breeder Reactor", http://en.wikipedia.org/wiki/Breeder_reactor, January 2015.

WNA, 2014. "Thorium", World Nuclear Association, http://www.world-nuclear.org/info/inf62.html, September 2014.

14 Relationships with Other Chapters

Chapter 8 provides an overview of fuel bundle configuration. Chapters 3 to 5 explain neutron physics that generates heat in the fuel. Chapters 6 and 7 explain how that heat is removed from the fuel bundle and illustrate the internal temperature distribution within a fuel rod. Chapters 14 and 15 explain chemical and metallurgical aspects that relate to corrosion of the fuel sheath. Chapter 13 explains the performance of fuel during postulated accidents. Chapter 17 describes the design and performance of fuel, focussing on the current natural-uranium cycle (i.e., low burnup). The current chapter (#18) describes a few selected fuel cycles, summarizes key aspects of fuel manufacturing related primarily to the natural-uranium fuel cycle, and explains a few selected aspects of fuel performance that become relatively more important for advanced fuel cycles. Finally, Chapter 19 describes interim storage and disposal of used fuel.

15 Acknowledgements

The authors thank Dr. Peter Boczar for permission to borrow heavily from his extensive publications on fuel cycles. Sections 2.1 and 2.2 of this chapter were provided by Dr. Dorin Nichita, Dr. Ben Rouben, and Dr. William Garland. Dr. William Garland also contributed Figures 1 and 2. Dr. D. Baron (EdF, retired) graciously provided material on high burnup structure. John Roberts and Dr. Craig Stuart contributed key material on fuel deposits. The above contributions are gratefully acknowledged with thanks.

In addition, the following reviewers are gratefully acknowledged for their hard work and excellent comments during the development of this chapter; their feedback has much improved it:

Peter Boczar

Bronwyn Hyland

Geoff Edwards

William Garland

Of course the responsibility for any errors or omissions lies entirely with the authors. Thanks are also extended to Simon Oakley and Media Production Services, McMaster University, for nicely drawing the figures, and to Diana Bouchard for expertly editing and assembling the final copy.