THE RIGHT KIND OF SMR FOR TODAY AND TOMORROW

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Abstract

Canada has derived its nuclear energy for over 60 years from natural uranium fuel in its CANDU reactors. Even though it only consumes 0.74% of the mined uranium, the CANDU reactor is still the most fuel-efficient of thermal reactors. The potential of adding small modular reactors to Canada's fleet makes it imperative that the nuclear power paradigm in this country be re-examined. Should one continue with the CANDU design to exploit as little as 0.74% of the world's most energy-rich fuel, or extract even less with the use of all other thermal SMRs? Or is it time to change to a more efficient uranium fuel use, over 100 times more fuel efficient, a use that can as a result hugely minimize used nuclear fuel waste with the acquisition of fast-spectrum SMRs and the adoption of recycling to close the fuel cycle? Philosophically the time to change has never been better, with the modular nature of the SMRs providing an entry that is doable, affordable, and expandable as needed.

1. Introduction

Energy means life, and for life to be sustainable, energy has to be sustainable.

Grandfather used to have a wood lot on his farm, where every year we would fell a good-sized beech tree that would serve as a source of wood to keep us warm during the winter. The lot was big enough that this energy source was sustainable; and it was carbon neutral.

This approach was replaced by coal, and oil, and natural gas, fuel substances not derived from a very recently grown tree but from very ancient vegetation and growths, now fossilized. As we are discovering, using this fuel source has Earth-wide consequences. It is not carbon neutral, and it is not sustainable on any time cycle short of millions of years.

Then there is uranium. It is the World's most compact energy-rich fuel. Uranium can be split, "fissioned", to release about 200 MeV of energy. That is 40 million times more than released from the chemical combination of a carbon atom with two oxygens to produce carbon dioxide.

All heavy fuel atoms, like uranium, thorium or plutonium, release that much energy when fissioned. There are two uranium isotopes, U-235 at 0.72% and U238 at 99.28%. U-235 is easily split with "thermal" neutrons, neutrons whose energy has equilibrated with the motions of atoms that surround them. Consequently U-235 is the focus and choice of fuel in virtually all of about 440 nuclear power reactors, "thermal" reactors, worldwide [1]. With so little of U-235 in uranium, it is therefore a precious isotope.

The U-238 isotope is more difficult to split except with neutrons of very high energy. However, about half of the energy in a thermal reactor nevertheless comes indirectly from U-238, via its transmutation to the slightly heavier "transuranic" isotope plutonium-239 (Pu-239), which splits with thermal neutrons as easily as U-235 [2].

As a consequence of this heavy concentration on the small amount of U-235, less than 1% of the total energy of uranium is accessed in current reactors. More worrisome, the singular focus on this isotope, even its selective extraction from bulk uranium, means that 99% of the potential energy of the uranium is discarded, both as depleted uranium and as used nuclear fuel. Both of these stocks with still over 99% uranium in over 2 million tons by now, are looked at as unusable and even as hazardous highly radioactive "waste" from the view of thermal reactor engineering.

As a result of such inefficient fuel use, the OECD-Nuclear Energy Association estimates that, with the current practice in Canada of uranium mining and exporting around 16,000 tons per annum, Canada's economically accessible uranium reserves of over 500,000 tons of uranium will be depleted in about 30 years, by the year 2050 [3]. Moreover, World reserves in relation to World uranium demand will last only a few decades beyond that [3]. The stress on uranium reserves can only increase with the call for greater decarbonization in the nations' economic sectors such as transportation, industry, and home heating, which use fossil energy equivalent to over 16 times the current non-carbon nuclear power, worldwide [4],

One response is greater fuel efficiency in nuclear energy generation. At its best, increasing uranium use from <1% to all 100% would immediately relieve the pressure on uranium reserves by over a factor of 100, and provide copious non-carbon energy for many centuries.

Canada's intent to add small modular reactors to its nuclear fleet provides that opportunity. The choice of the right type of SMR is crucial, since not all have the capability to be fuel efficient.

> What prevents thermal reactors from accessing more of the energy in uranium?
> Can alterations in fuel composition improve thermal energy yields substantially?
> Can a redesign of reactors assist?
> Are there limits imposed by the physics of nuclear energy?
> Are there alternative avenues offered by the physics of nuclear energy?

The brief answers are:
1. physics
2. no
3. yes
4. yes
5. yes.

The longer examinations are below, concluding with the only path forward that makes sense: recycling the fuel through fast-spectrum reactors, i.e. using neutrons with higher energies. Thermal reactors of any sort, even as good as the CANDUs, don't have the capabilities.

2. The Fundamental Physics of Thermal Reactors

The fundamental aim of the physics and engineering of nuclear energy is the maximization of the dual beneficial effect of a neutron:

1.) splitting a heavy fuel atom to release its nuclear energy and 2.) maintaining enough of the 2 to 3 new neutrons emitted in this event in order to continue a chain reaction.

Very little thought has been given to the fact that only a very small proportion of uranium is actually used, or more importantly that a very large latent energy fraction of the uranium is left unused. The potential to rectify this omission is the underlying concept in this paper. That idea, too, should be part of the physics, the engineering, and the politics.

To achieve the first benefit requires merely a relatively “easily splittable” or “fissile” isotope of fuel atom, such as U-235, U-233, Pu-239, Pu-241, etc.

The second benefit requires more care, since all atoms, whether fuel, coolant, or structure, absorb neutrons. Therefore to conserve neutrons for fission, the coolant and structural materials have been chosen with very low neutron absorption coefficients or “cross sections”. Atoms and molecules such as zirconium, carbon, sodium, fluorine, or deuterium in “heavy water” are examples of the latter. Normal water also has a relatively low absorption cross section, although about 600 times higher than heavy water [2]. Enriched U-235 has to compensate for that difference.

For fuel atoms there is not much choice. Every fuel isotope has two intrinsic cross sections that consume neutrons, one for splitting, or “fission”, and one for absorption, also called “radiative capture”. Uniquely among naturally occurring atoms, uranium has a natural isotope, U-235, that has a large thermal fission cross section, at 596 barn (1 barn = 10^{-24} cm^2), that makes it easily “splittable”. U-235 also has a sizeable absorption cross section, at 98 barn. U-238 has an almost negligible thermal fission cross section, 0.0000403 barn, while its absorption cross section is also relatively small, at 2.63 barn [2]. Thorium, another potential nuclear fuel, has no naturally occurring fissile isotope. Its one major isotope, Th-232, is like U-238, with a negligible thermal fission cross section at 0.0000536 barn, and an absorption cross section of 7.23 barn [2]. It cannot fuel a reactor by itself, but requires the assistance of U-235 or uranium-produced Pu-239.

In any thermal neutron interactions with uranium, the characteristics of the isotope U-235 will predominate, with its relative concentration being decisive in the overall outcome. Some modulation can be effected by changing the relative concentration of U-238 in the fuel, but its characteristics are usually neglected *a priori*. They should not be; they affects fuel sustainability.

2.1 The CANDU Case

The characteristics of the CANDU reactor, heavy-water-cooled and –moderated, will be examined in some detail in light of its use of unchanged natural uranium as fuel. The underlying properties and quantities for the components of the core of this reactor are shown in Table 1.

The relative fuel concentrations in column 1, shown as a percentage in moles of the total core composition, are in the natural fuel ratio of 0.72% U-235 and 99.28% U-238.

The relevant results are shown in bold in columns 4 to 6. The relative instantaneous amounts of U-235 consumed per neutron absorbed in fission are 55.2%. Since each nucleus of U-235 that is fissioned yields on average 2.46 new nascent fission neutrons [2], there will therefore be 1.36 new neutrons (column 6) for each neutron used for all absorptive interactions within the core. Since only one (1) neutron is required to continue the chain reaction, the excess of neutrons

	Existing CANDU Core Operation						**“Break-Even Operation”**			
	1	2	3	4	5	6	7	8	9	10
Isotope	Moles	**Cross Sections**		**Fission**	**Absorp’n**		**Moles**	**Fission**	**Absorp’n**	
	%	**Fission barn**	**Absorp’n barn**	**%**	**%**	**Net Neutron Yield**		**%**	**%**	**Net Neutron Yield**
U238	4.02	0.’’’’403	2.63	0.’’’51	**33.2**		4.037	0.’’’74	**48.2**	
U235	0.0295	596	97.6	**55.2**	**9.0**	**1.36***	0.0153	**41.4**	**6.8**	**1.02***
O(fuel)	8.105	0	0.’’’368	0	0.0094		8.105	0	0.014	
D_2O	84.09	0	0.00116	0	0.31		84.09	0	0.44	
Zr-alloy	3.75	0	0.191	0	2.25		3.75	0	3.24	
CO_2	0.00315	0	0.00432	0	0.’’’’43		0.00315	0	0.’’’’62	

Table 1. Analysis of the Proportion of Fission Neutrons Created and Fissile Atoms Used versus Fissile Atoms Replaced in a CANDU Reactor Core Under Current Operating Conditions and Under a Hypothetical “Break Even” Condition by Changing the Isotopic Ratios between U235 and U238.

* U-235 fission neutron yield = 2.46 [4]

Note: Symbol ’’’ indicates zeros

permits neutron losses during neutron moderation from high energies, escape of neutrons out of the core, absorption in fission products created during operation, as well as permitting reactor operation until U-235 fuel atoms are depleted sufficiently for refuelling to become necessary.

In total there are 55.2% + 9.03% U-235 atoms consumed per neutron in fission and absorption (radiative capture) for a total of 64.23% fissile atoms used per neutron entering the core (columns 4 and 5). In contrast there are only 33.2% new fissile atoms created via neutron absorption in U-238 and the subsequent transmutation of this isotope to fissile Pu-239. Thus a reduction in fissile atom numbers occurs in this configuration for the CANDU core, and fissile fuel replenishment will be required when the total neutron yield can no longer maintain neutron equilibrium (also known as criticality), after about one year of operation. At this time only 0.74% of the uranium has been consumed.

Since there is initially an excess of neutrons, it is of interest to see if an adjustment in the relative concentration of U-235 and U-238 can create a balance between fissile U-235 nuclei used in fission and absorption, and replenishment of fissile nuclei via absorption (radiative capture) in U-238 followed by a transmutation to Pu-239. However, for continuing reactor operation the relative number of neutrons created (the net neutron yield) should remain sufficiently above 1.0.

This attempt is shown in columns 7 to 10 in Table 1.

To create relatively more fissile Pu-239 isotopes it is necessary to increase the relative concentration of U-238. Since U-238 is already above 99%, this requires a compensatory decrease in U-235 concentration, and a corresponding smaller number of fissions. Balance between fissile consumption and fissile creation can be achieved if U-235 is decreased from 0.72% to 0.37%, with an increase of U-238 from 99.28% to 99.63% (corresponding to change in the relative number of moles of fuel isotopes between columns 1 and 7).

From a fuel point of view this would permit such a CANDU configuration to operate indefinitely with a conversion ratio of 1.0, by extracting fission products and replenishing purely with a source of plentiful U-238. However, the power per pressure tube in the reactor would be reduced with the smaller number of energy-creating fission events per neutron from 55.2% to 41.4%.

Worse still, the new nascent neutron yield, while still above 1.0, is now reduced from 1.36 to 1.02 (columns 6 and 10). This value is not sufficiently high to furnish enough neutrons to permit the obligatory losses during neutron moderation and the losses from the escape of neutrons out of the core, nor will it permit extended operation of the reactor. In fact, in practical terms the reactor core would not achieve neutron equilibrium at all to operate. Indeed, an operational CANDU replenishes its fuel when the fissile content has reached a low of 0.50%, still higher than the theoretical 0.37% above.

Thus, unfortunately, a fuel state for a conversion ratio of 1.0 is not possible for this CANDU reactor design, whether full-sized or as a CANDU-3 SMR.

2.2 The Ubiquitous Light Water Reactor

The results for a similar analysis carried out with a model core of a light water reactor consisting of an array of 17 x 17 fuel rods in typical PWR fuel assemblies [5] are shown in Table 2. To compensate for the greater neutron absorption of the light water coolant compared to the heavy water coolant used above, the concentration of U-235 in the fuel has been increased to 3.0 %, corresponding to a relative number of moles U-235 of 0.48% in the entire core (column 1).

The greater efficiency of moderation by hydrogen in light water as compared to deuterium in heavy water permits a reduction in moderating water from a relative 84 mole% in the CANDU core to 41.5 mole% in the PWR core.

The resulting net new neutron yield is a high 1.75 neutrons created for every neutron consumed in all interactions in the core (Table 2, column 6), easily accommodating neutron operating losses, higher power levels, or longer operating times before refuelling becomes necessary.

However, the disparity between fissile U-235 used in fission and absorption and new fissile Pu-239 atoms created via transmutation from radiative capture of neutrons in U-238 is large. For every 82.6 fissile U-235 atoms used (columns 4 + 5), only 10.1 fissile Pu-239 atoms are created from absorption of neutrons (radiative capture) in U-238 (column 5). The dominance in neutron utilization by the high concentration of U-235 leaves very few neutrons to interact with U-238 and transmute it to fissile Pu-239.

In spite of the high net neutron yield, at 1.75, attempts to find conditions for the establishment of an equilibrium between fissile atoms used and fissile atoms created in the core met with failure.

	PWR Core Operation						**“Break-Even Operation”**			
	1	2	3	4	5	6	7	8	9	10
Isotope	Moles	**Cross Sections**		**Fission**	**Absorp’n**		**Moles**	**Fission**	**Absorp’n**	
	%	**Fission barn**	**Absorp’n barn**	**%**	**%**	**Net Neutron Yield**		**%**	**%**	**Net Neutron Yield**
U238	15.66	0.’’’’403	2.63	0.’’’155	**10.1**		16.1	0.’’’57	**37.0**	
U235	0.485	596	97.6	**71.0**	**11.6**	**1.75***	0.061	**31.8**	**5.2**	**0.78***
O(fuel)	32.3	0	0.’’’368	0	0.0029		32.3	0	0.0094	
H_2O	41.5	0	0.666	0	6.79		41.5	0	0.31	
Zr-alloy	10.1	0	0.191	0	6.03		10.1	0	2.25	
He	0.056	0	~0	0	~0		0.056	0	~0	

Table 2. Analysis of the Proportion of Fission Neutrons Created and Fissile Atoms Used versus Fissile Atoms Replaced in a PWR Reactor Core Under Normal Operating Conditions and Under a Hypothetical “Break Even” Condition by Adjustment of Isotopic Ratios between U235 and U238.

* U-235 fission neutron yield = 2.46 [4]

Note: Symbol ’’’ indicates zeros

The establishment of such a theoretical fuel equilibrium between fissile U-235 and transmutation of U-238 (Table 2, U-235 col’s 8 + 9 versus U-238 col. 9) had the result that the fraction of new neutrons created, 0.78, was well below the number required for neutron equilibrium (col. 10).

Thus such a thermal PWR cannot achieve an operating “break even” conversion ratio of 1.0.

2.3 High Temperature Helium-Cooled Reactor

The third type of reactor analysed is a TRISO-fueled reactor cooled with helium, moderated with carbon. The design of the core is reminiscent of the Xe-100 SMR proposed by X-energy. The core is effectively filled only with 6 cm pebbles composed of a carbon core filled with tens of thousands of uranium oxide spheres surrounded by shell layers of pyrolytic graphite and silicon carbide. Carbon, silicon, helium and even oxygen have very low radiative capture cross sections. This results in a core whose properties are uniquely dominated by the uranium isotopes alone.

Table 3 shows that with a fuel enriched to 15% U-235 the new neutron yield is 2.04 (Table 3, column 6), close to the theoretical limit of 2.11 for pure U-235. Inversely related to this dominance of U-235 cross sections is a rather feeble signal from the 85% U-238 in the fuel,

HT-Gas-Cooled Core Operation							“Break-Even Operation”			
	1	2	3	4	5	6	7	8	9	10
Isotope	Moles	Cross Sections		Fission	Absorp’n		Moles	Fission	Absorp’n	
	%	Fission barn	Absorp’n barn	%	%	Net Neutron Yield		%	%	Net Neutron Yield
U238	0.434	0.’’’’403	2.63	0.’’’’316	**2.06**		0.508	0.’’’603	**39.33**	
U235	0.077	596	97.6	**83.0**	**13.6**	**2.04***	0.00193	**33.8**	**5.53**	**0.831***
O(fuel)	1.03	0	0.’’’368	0	0.00068		1.03	0	0.0111	
C	94.2	0	0.00358	0	0.611		94.2	0	9.94	
Si	2.40	0	0.161	0	0.700		2.40	0	11.39	
He	1.86	0	~0	0	~0		1.86	0	~0	

Table 3. Analysis of the Proportion of Fission Neutrons Created and Fissile Atoms Used versus Fissile Atoms Replaced in a High-Temperature Gas-Cooled Reactor Core Under Normal Operating Conditions and Under a Hypothetical “Break Even” Condition by Adjustment of Isotopic Ratios between U235 and U238.

* U-235 fission neutron yield = 2.46 [4]

Note: Symbol ’’’ indicates zeros

which only delivers about 2 fissile Pu-239 atoms via radiative capture of neutrons (column 5) as a very much insufficient replacement for every 97 fissile U-235 atoms used.

Attempts to achieve a balanced response of the two effects, of fissile use and of fissile replacement, again meet with failure. The net neutron yield, 0.831, drops well below neutron balance in the core (below criticality) before parity of fissile use and fissile creation from U-238 transmutation is established.

3. Theoretical Limits

The above examples of three different types of thermal reactors indicate that it is impossible to produce an operating thermal reactor that can maintain its fissile fuel complement. One can establish equations that indicate the limits of the interplay between cross sections and relative fuel compositions which corroborate this conclusion.

It can easily be shown that for the ideal case of a core fuelled with a combination of a pure fissile and fertile pair of isotopes, like U-235 and U-238 and negligible other absorptions (the CANDU and the Xe-100 come close), the fraction “FR” of fissile renewal versus fissile consumption is given by Equation 1:

$$\mathbf{FR} = \frac{(1 - n_p) \times \sigma_r(c)}{n_p \times (\sigma_p(f) + \sigma_p(c))} \tag{1}$$

where "n_p" is the fraction of the fissile isotope U-235 or Pu-239 that powers the reactor, "$\sigma_r(c)$" is the radiative capture cross section of the fertile isotope U-238 that replenishes the fissile content, and "$\sigma_p(f)$" and "$\sigma_p(c)$" are the fission and capture cross sections of the fissile isotope.

Similarly, the coefficient for neutron replacement, "NR", is given by Equation 2:

$$\mathbf{NR} = \frac{\sigma_p(f) \times \overline{\nu}}{\sigma_p(f) + \sigma_p(c) - \sigma_r(c) + \sigma_r(c) / n_p} \tag{2}$$

where $\overline{\nu}$ (nu-bar) is the yield of new neutrons in the fission of any isotope, and the other terms are defined above for Equation 1.

The values of FR and NR calculated at thermal energies via the two equations are shown graphically for the fissile and fertile pair of U-235 and U-238 in Figure 1. What is remarkable is the apparent theoretical capability of achieving an operating state in which both factors, neutrons and fissile isotopes simultaneously have the capability of being maintained and renewed, since the cross-over of the two curves is at 1.0 or above. That suggests that with recycling all of the energy potential in uranium, both from U-235 and from every U-238 atom can theoretically be extracted in a thermal reactor.

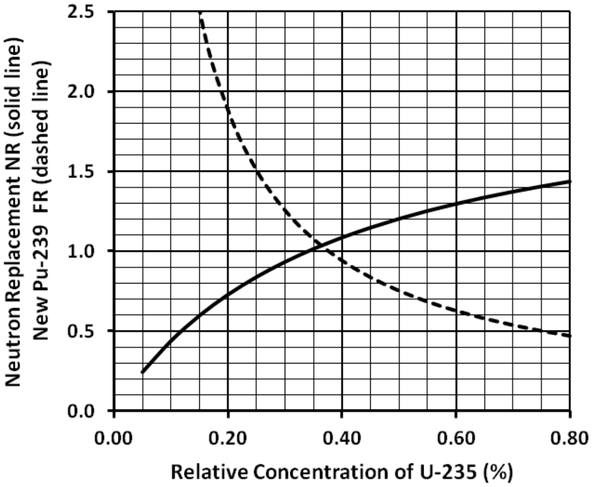

Figure 1. Relationship at thermal energies between the increase in the relative concentration of U-235 in U-238 and the capability to replace neutrons in the reactor core, and to replace fissile isotopes as the result of transmutation of fertile U-238 to Pu-239.

However, this state is achieved when both renewal capabilities are just barely possible, i.e. with a simultaneous value of 1.02, just above 1.0. In a thermal reactor newly created neutrons have to undergo a reduction in energy, or "moderation", from their nascent high energy state of an average of 2 MeV. During this process about 10% or more of the neutrons are lost to absorption with in fuel, coolant and structural components of the core.

The CANDU reactor in practice operates to the right of that cross-over point in Fig. 1, with a starting fuel composition of 0.72% U-235, and its fuel cycle ends with the total fissile percentage of U-235 + Pu239/241 at about 0.50. The PWR and the high temperature TRISO-fuelled reactors are far off this scale, to the right, with a fissile starting composition of 3.0% and 15% in the above examples.

In Equations 1 and 2 the core was idealized, containing only fuel, for the results in Figure 1. Every material item in the core that absorbs neutrons creates results that are worse, i.e. with a lower cross-over of the two curves.

Therefore one can effectively state that no thermal reactor operating with uranium, full-size or in the form of an SMR, can achieve a conversion ratio of 1.0, i.e. an overall replacement of its fissile content. At refuelling, additional extraneous fissile material is required for all of them.

A reactor core fuelled with a combination of thorium and U-233 can achieve a simultaneous renewal value of about 1.09 for both characteristics. Nevertheless, it is questionable whether a core design can be found in which thermalization of nascent neutrons can be achieved with a smaller loss of neutrons than 10%.

4. Alternative Reactors: Fast Neutron Spectra

The examination of Equations 1 and 2 indicate that at thermal energies there is no way to change outcomes other than changing relative composition of isotopes. The neutron cross sections are fixed. An approach to change fissile isotopes from U-235 to Pu-239 also met with failure. The rather promising higher fission neutron yield of Pu239 of 2.88 vs U-235 at 2.46 was expected to help, but its effect is undone by the higher unproductive neutron absorption via a radiative capture cross section of 272 barn for Pu-239 versus only 98 barn in U-235.

However, since nascent fission neutrons start their lives with an average energy of about 2 MeV, it is instructive to examine the renewal capabilities of neutrons and fissile isotopes at high energies. Cross sections at high neutron energies, both for fission and for radiative capture have quite different values compared to thermal energies.

Nascent fission neutrons have a broad spectrum of energies, with tails from about 3 to 5 MeV at the high end and around 0.1 MeV at the low end [7]. In addition, that spectrum shifts to lower energies due to elastic and inelastic scattering. Therefore calculations of renewal of fissile isotopes and of fission neutrons were carried out at neutron energies of 0.1 MeV, 0.3 MeV and 1.0 MeV to sample the effect of changing values in fission and radiative capture cross sections.

The results for the fissile/fertile pair of U-235 and U-238, and also for the Pu-239 and U-238 pair at high neutron energies are shown in Figures 2 and 3 respectively.

The interplay of cross sectional values in Equations 1 and 2 for the U-235/U-238 pair at high neutron energies (Fig.2) results in an outcome that is very similar to that shown in Fig. 1 at thermal energies. Discouragingly, the results show that for this pair of isotopes the simultaneous achievement of neutron equilibrium and initial conversion ratio (new fissile ratio) is barely at 1.0 or even below (Fig. 1) over virtually all of the neutron high energy range. Since cross sections are generally lower, and even lower for U-235 in comparison to U-238, the fissile concentrations to achieve neutron equilibrium are as high as 7%. This suggests that a fast-spectrum reactor operating with optimally enriched U-235, whether SMR or full size, will have difficulty

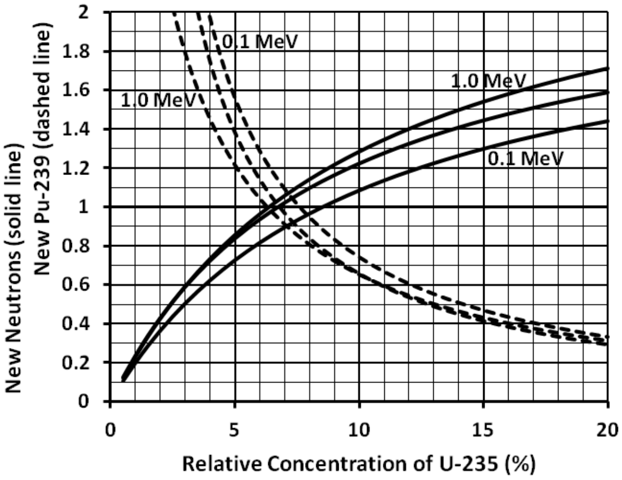

Figure 2. Graphical representation of Equations 1 and 2 for the creation of new neutrons from the fission of U-235 via neutrons at 0.1, 0.3 and 1.0 MeV, as well as the replenishment of fissile isotopes by Pu-239 via neutron capture in U-238

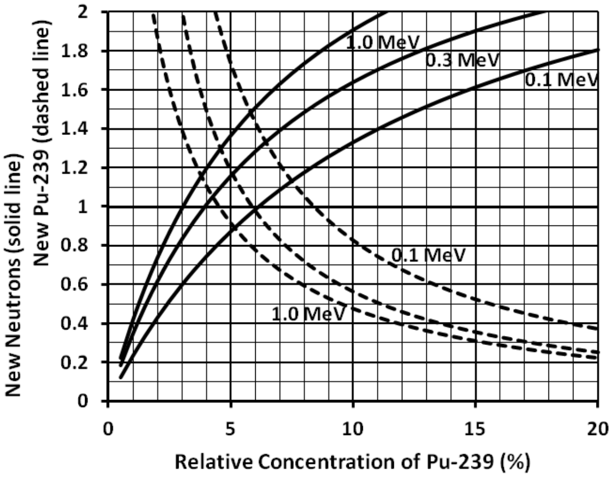

Figure 2. Graphical representation of Equations 1 and 2 for the creation of new neutrons from the fission of Pu-239 via neutrons at 0.1, 0.3 and 1.0 MeV, as well as the replenishment of Pu-239 via neutron capture in U-238

operating in a "break-even" mode, and will require extraneous fissile fuel replenishment in every fuel cycle.

Figure 3 shows that the situation is quite different if the reactor is fuelled with a combination of U-238 and the transuranic actinide Pu-239, i.e. with fuel that can be obtained by recycling existing used thermal reactor fuel such as used CANDU fuel. In Fig. 3 all high energy calculations indicate that a simultaneous fissile conversion ratio and overall neutron yield can be achieved that is as high as 1.13 to 1.17.

5. Choice of SMR

The fuel response of a generic fast-spectrum core fuelled with U-238 and the transuranic Pu-239 (Fig. 3) is very encouraging from several points of view, particularly from the point of view of fuel sustainability coupled with fuel independence for this country.

Any reactors fuelled with enriched U-235, whether thermal or fast-spectrum SMRs, will require continual enriched U-235 replenishment on refuelling, according to Figs.1 and 2. Canada has no U-235 enrichment facilities due to a deliberate political decision made early in our nuclear energy history. Thus if enriched U235 is to be used, Canada is immediately dependent on foreign countries, likely on nuclear weapons nations such as the USA, France, the UK, Russia or China. Even CAMECO's recent purchase in the USA into Global Laser Enrichment, an experimental enrichment approach, does nothing to lift our dependence on foreign supplies of such uranium and at worst hastens the depletion of Canada's uranium reserves to less than 30 years [3].

However, fuel with fissile Pu-239 as part of a mix of transuranics and uranium for fast-spectrum SMRs can be economically extracted locally from Canadian stockpiles of used CANDU fuel [6]. There is enough fissile content in Canada's 60,000 tons of used fuel to start such SMRs with a total power of 24,000 MWe. Moreover, the recovered uranium from such stockpiles can replenish any such fast-spectrum SMRs for many centuries.

Figure 1 suggests that any thermal reactor is at the very best only useful for niche applications rather than for the long haul. It is wasteful of fuel, perpetuating the current profligacy of "less-than-one-percent" uranium fuel use.

Recycling of relatively low-grade fissile used CANDU fuel for thermal reactors is too expensive as a continual approach and does not improve uranium fuel use substantially, as shown even for reprocessing of LWR fuel where it has been carried out. On the other hand, a one-time concentration and extraction from used CANDU fuel of an appropriate mix of transuranic actinides and uranium to provide a one-time starting fuel for fast-spectrum reactors does make economic sense [6].

Are there appropriate fast-spectrum SMRs among the designs submitted by vendors or waiting in the wings in the Canadian context? Several immediately come to mind that deserve further examination: the lead-cooled 3 MWe to 10 MWe Sealer reactor or its larger 55 MWe version from Leadcold fuelled with uranium oxide or uranium nitride fuel [8]; the sodium-cooled metal-fuelled 100 MWe ARC-100 from Advanced Reactor Concepts [9]; the molten-salt cooled and fuelled 300 MWe Moltex SSR-W [10]; and the sodium-cooled metal-fuelled 300 MWe GE-Hitachi PRISM [11].

6. Fuel Cycling

To utilize the high fuel-consuming capabilities of fast-neutron reactors, fuel cycling is key. Such a process must be efficient, inexpensive, effective and with a minimum of extraneous waste. One such concept is electro-refining (pyroprocessing) of used fuel into new fuel [12].

This concept is at its simplest with metal fuel (e.g. ARC-100 and PRISM), which can be purified directly by electrolytic refinement in molten salts [12]. In an electrolytic cell the used metal fuel becomes the anode at which the impure used metal fuel components are dissolved electrically into the molten salt electrolyte. An electric current carries the dissolved metal ions across to two cathodes at which pure metal or a pure metal mixture is deposited. The fission products remain in the molten salt electrolyte from which they are periodically removed, while the salt is returned to the electrolyte. A similar partial process is being investigated by Moltex for their SSR-W molten salt fuel.

7. Summary and Conclusion

The future of climate change mitigation requires centuries of massive amounts of clean, non-carbon energy. Expanded nuclear power has the capabilities to supply that energy, but only with a much more efficient use of uranium fuel. The fundamentals of physics, above, indicate that there are no thermal reactors of any size which are capable of delivering efficient fuel utilization. The nuclear industry must transition to fast-spectrum reactors which alone are capable, with recycling, of consuming existing stockpiles of radiotoxic used nuclear fuel, using stored depleted uranium, and consuming every radioactive heavy atom in the World's uranium reserves to produce the World's needed non-carbon energy for many centuries.

Such fast-spectrum reactors, some named above, are being examined currently in Canada's SMR Action Plan. We must choose them. There is no other sane option.

8. References

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