REACTOR PHYSICS STUDIES FOR A PRESSURE TUBE SUPERCRITICAL WATER REACTOR (PT-SCWR)

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Abstract

Preliminary lattice physics and full core neutronic analysis have been performed for the pressure-tube supercritical water reactor (PT-SCWR). Current CANDU reactor physics codes (WIMS-AECL and RFSP) were used for modeling this reactor. A key challenge in the physics design of this reactor is the optimization of lattice parameters to achieve the appropriate balance between coolant void reactivity (CVR) and fuel utilization. A vertically-oriented, batch-fuelled reactor is considered, with an insulated pressure tube to accommodate the high coolant temperatures and pressures. The analysis shows the reactor physics conceptual feasibility of the design, although further optimization is required.

1. Reactor physics aspects of the PT-SCWR

The PT-SCWR concept addresses the objectives of the Gen IV next-generation reactor program of improved sustainability, economics, proliferation-resistance and safety [1, 2].

The use of supercritical light water as coolant in the PT-SCWR has a major impact on core neutronics. The hydrogen in the light water coolant is effective at slowing-down neutrons and it contributes to moderation by the heavy-water moderator. However, the hydrogen also absorbs neutrons, which impacts negatively on neutron economy and is a major contributor to the positive component of CVR. The sign and size of CVR depend on the balance between the opposing effects of moderation and absorption in the supercritical water coolant. The higher temperature of the coolant lowers the effectiveness of the coolant in thermalizing neutrons, resulting in a harder neutron spectrum. The supercritical water conditions result in a lower average coolant density, and there is a large variation in coolant density along the channel.

The use of supercritical water as coolant results in a very large increase in channel outlet temperature leading to high thermodynamic efficiency (>45%), which will improve fuel utilization (expressed in terms of electrical energy, rather than thermal energy) by about 50% over the conventional CANDU reactor. The use of supercritical water coolant also requires the use of special in-core materials, many of which have a negative impact on neutron economy. The material properties of Zircaloy, which has relatively low neutron absorption, will not be acceptable at SCWR temperatures. Suitable materials for SCWR fuel cladding (such as stainless steel) have much higher neutron absorption than zirconium-based materials and consequently will degrade neutron economy (and achievable burnup). One of the key challenges of PT-SCWR physics is thus to find the balance of materials, fuel and lattice design that enable the gain in thermodynamic efficiency to more than offset the losses due to in-core material requirements.

The ability to maintain a thin zirconium-alloy pressure tube at SCWR temperatures is enabled by thermally insulating the pressure tube from the high-temperature coolant. Hence, the concept of an “insulated pressure tube”, or “high-efficiency channel” (HEC) is essential in the PT-SCWR design. In addition, one option being considered is a re-entrant channel (REC) in a vertical core, in which the coolant flows down the inside of the pressure tube, keeping it cool, before rising up through the channel to remove the fission heat. The HEC and REC conceptual designs are similar, differing by the presence of a small coolant annulus (nominally 3 mm thick) adjacent to the pressure tube in the REC design, so the choice between these designs can be made on the basis of considerations other than neutronic.

The PT-SCWR allows for the traditional separation of coolant and moderator in the CANDU design, allowing each to run at different conditions (and allowing different materials for each). The PT-SCWR retains heavy water moderator and the high thermalization thereof. Heavy water has a longer slowing down length, so a larger pitch is required to fully thermalize the neutrons, and is preferred for achieving good neutron economy (and fuel utilization).

Thus, the use of supercritical water as coolant thus requires a careful optimization of the lattice cell to optimize neutron economy (fuel utilization) and CVR. This is perhaps the biggest challenge in PT-SCWR physics design.

An important characteristic of the lattice, which affects the spectrum, burnup and CVR, is the moderator-to-fuel ratio (M/F). Generally speaking, as M/F increases, both CVR and burnup increase. Several means of varying M/F have been explored thus far:

- varying the lattice pitch;
- using a calandria tube with a gas gap between the pressure tube and the calandria tube (which displaces moderator and reduces M/F);
- varying the thickness and porosity of the insulator between the fuel bundle and the pressure tube;
- using moderator displacement tubes (empty tubes at the corners of the square lattice that displace moderator);
- modifying the geometry of the fuel bundle, i.e. moving the fuel to the outside of a larger bundle and channel.

One concept being considered, which reduces the pressure difference between the inside and outside of the pressure tube, is use of a pressurized moderator. If a pressurized moderator is chosen, requiring a pressure vessel, then this will impact on core neutronics in several ways. A higher range of moderator temperatures (and densities) can be considered, which will affect M/F for a given lattice pitch as well as the burnup and

CVR. The preclusion of a postulated fast rod rejection accident would need to be addressed, for instance by using the current LWR approach. (In the current CANDU reactor, reactivity devices are located inside the un-pressurized moderator.) The size of the pressure vessel would also limit the number of fuel channels that could be accommodated in the vessel, the length of the fuel channels, or the lattice pitch – reducing the flexibility inherent in the modular PT-SCWR design.

Batch refuelling in a vertical reactor, with uni-direction coolant flow, and refuelling the entire channel has been chosen as the reference at this stage of the pre-conceptual design. This simplifies the sealing between the refuelling machine and the fuel channels, and has the potential of simplifying the feeder design and reducing the feeder length. From a physics perspective, batch refuelling reduces the achievable burnup for a given fuel enrichment. With 3-batch refuelling, the achieveable burnup is reduced by at least 25% compared to on-line refuelling. With uni-directional coolant flow and batch-refuelling, there will not be the averaging of neutronic properties from one channel to another, with the associated natural flattening of the axial power profile, to the same degree as with bi-direction on-power refuelling.

At this early stage in the pre-conceptual design, physics requirements have not been specified. Power output can be achieved by choice of the number of fuel channels, the length of the fuel channel, and the number and size distribution of fuel elements in the bundle. Ultimately, there will be safety and operational limits on the maximum fuel temperature, which can be met by varying these parameters.

Careful consideration will be required in specifying requirements for reactivity coefficients. Generally, they should be small. Power coefficient of reactivity (PCR) should be small and negative, as should the fuel temperature coefficient of reactivity. CVR should simply be small, as discussed below.

With the large change in coolant density along the channel, a small value and variation of CVR with burnup will help ensure control and stability during normal operation (e.g., changes to coolant density will have a small impact on reactivity). A small value of core-average CVR will provide a level of inherent safety during postulated accidents (that either increase or decrease the coolant density). With supercritical light water as coolant, absorption in the hydrogen will contribute a large positive component to CVR. To offset this positive component, a negative component must be introduced. This can be achieved either by having more moderation occurring in the coolant (smaller M/F) so that the loss of that moderation upon voiding results in a decrease in reactivity, or by increasing absorption somewhere in the lattice upon voiding. Several means of reducing M/F were previously mentioned. The low void reactivity fuel (LVRF) concept for CANDU reactors and the ACR-1000 places a burnable neutron absorber (BNA) in the centre of the fuel bundle, where the thermal flux increases slightly upon coolant voiding. The absorption in that absorber increases during coolant voiding, introducing a negative component to CVR. Both approaches, reducing M/F and use of BNA, can be used in the PT-SCWR. Moreover, with batch refuelling, a means of suppressing the excess reactivity is required between refuelling cycles. In LWRs, this is done through a combination of neutron absorber in the moderator and BNA in the fuel (which also suppresses flux/power peaking at the beginning of cycle.) In the PT-SCWR, we have an opportunity to use BNA for both suppressing the excess reactivity required for batch refuelling and for reducing CVR. This concept is the subject of ongong studies.

2. Reactor physics codes and models

A formal assessment of code requirements and a selection of codes against those requirements have not yet been performed. For the preliminary scoping studies on the physics of the PT-SCWR performed to date, current CANDU/ ACR-1000 methods and codes have been used, namely 2-group, coarse-mesh, finite-difference diffusion theory analysis for the core neutronics, with cell-averaged cross sections from a lattice code [3]. WIMS-AECL version 3.1 [4] was used for lattice calculations (deriving 2-group cell-averaged cross sections for RFSP) with ENDF/B-VI nuclear data, and RFSP version 3.5.1 for 2-group, finite difference diffusion theory core calculations [5].

Some key aspects of the PT-SCWR pertaining to reactor physics methods are as follows:

- The reactor is a pressure-tube reactor, e.g., the lattice cell consists of a cluster geometry (a fuel bundle inside of a fuel channel, surrounded by moderator).
- There is a large variation of coolant density along the channel (see Figure 1; note that this figure is for a 6-m fuel channel; subsequent analysis was done for a 5-m channel).
- The current reference design is batch-fuelled, with uni-directional coolant flow.
- Standard length 50-cm CANDU fuel bundles have been modelled. (Even if the fuel assembly for a vertical core consists of a single fuel string, it would need to be subdivided for physics analysis.)
- The current analysis does not consider axial shuffling of fuel.
- Although the fuel will be located in different channels (radial locations) in the core as a result of batch refuelling, there will not be large changes in the actual coolant density that a fuel bundle (or segment of a full-length fuel assembly) would see during irradiation. Hence, the fuel will burn up at more-or-less constant coolant conditions, determined by the axial location of the fuel bundle in the channel. This greatly simplifies the physics analysis.
- Interstitial moderator displacement tubes (running parallel to the fuel channel at the corners of the lattice cell) are considered in some configurations.
- Fuel types currently being considered are LEU and a homogeneous mixture of plutonium and thorium (Pu/Th), with a target 3-batch-fuelled discharge burnup around 40 MWd/kg.
- Reactivity devices are not considered in this phase of work.

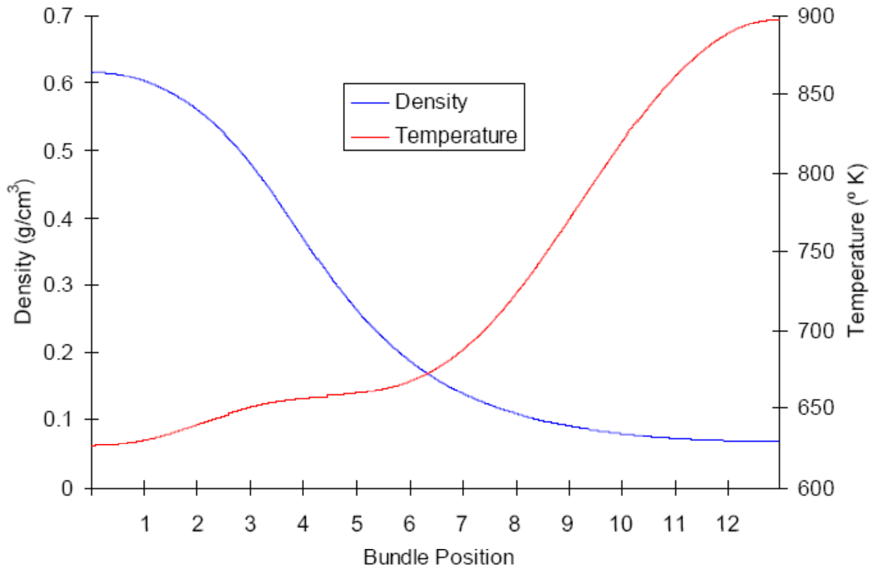

Figure 1 Axial coolant density and temperature distributions calculated using a symmetric cosine axial power distribution along a 6-m (12-bundle) fuel channel

The CANDU code suite is designed for this kind of fuel cluster geometry. Since reactivity devices are not being modelled during this phase of the analysis, the calculation of incremental cross sections is not needed. (DRAGON [6] is used for that purpose for physics analysis of the CANDU reactor and the ACR-1000.) Also, since the moderator displacement tubes run parallel to the fuel channel, they can be explicitly modelled in the 2-D WIMS-AECL lattice code.

<u>WIMS-AECL</u>. WIMS-AECL version 3.1 is a modern, state-of-the-art lattice code that has been substantially improved for use in ACR-1000 physics analysis. It calculates the properties of a lattice cell in many energy groups, in two dimensions, as a function of burnup. It has a multi-cell capability that allows for calculation of cell-averaged cross sections for the cell of interest, which can be surrounded by neighbouring cells of different compositions. While this multi-cell capability might have been needed for bi-directional cooling and fuelling (in which fresh fuel at one end of a channel with higher coolant density would be located adjacent to four neighbouring channels having burned fuel with low coolant density) it is not needed for uni-directional coolant flow with batch refuelling without axial shuffling (since the coolant density will be more-or-less uniform radially).

The lattice code is used in scoping studies to estimate the achievable discharge burnup for a particular fuel composition and lattice configuration. For on-line refuelling, the core comprises fuel from fresh to discharge burnup. Fuel is added (nearly) every day to maintain criticality of the core. The excess reactivity added during refuelling is only a few milli-k (mk), and is compensated by the zone control units. Except for the initial start-up core, there is usually very little burnable absorber in the moderator. The lattice code does not model the reactivity devices in the core (in the case of CANDU reactors, including adjuster rods, zone controllers, flux detectors and structural materials) – these are modelled in the 3-D transport code DRAGON as incremental cross sections which are added to the 2-group cell-averaged cross sections in RFSP. The discharge burnup for on-line refuelling is estimated in WIMS-AECL as that value of burnup for which the average value of k_{inf} equals the excess reactivity in the core (from leakage and absorption in reactivity devices and structural materials not explicitly modelled in the lattice code). For CANDU 6 reactors, that value is around 0.045 (45 mk) and the average discharge burnup with natural uranium fuel is ~ 7.5 MWd/kg.

With batch refuelling, enough excess reactivity needs to be added to the fresh fuel to keep the reactor critical until the reactor is fuelled in the next cycle. Of course, the reactor is exactly critical at all times between refuelling cycles, and that excess reactivity is suppressed by adding a dissolvable neutron absorber (boron) in the moderator, which is progressively removed during the cycle as the fuel reactivity depletes during burnup. BNA may also be added to the fuel, which can also suppress excess power peaking at the start of the cycle. Fuel management with batch refuelling needs to consider the cycle length (time between refuelling), fresh fuel enrichment and the discharge burnup. For a given enrichment, as the cycle length increases (and as the fraction of the core refuelled decreases), the burnup decreases. The core loading pattern also affects the peak ratings and the burnup: an out-in loading pattern decreases the peak ratings at the expense of increased leakage from the core (and lower burnup). This is discussed in [7] and illustrated for a 900 MWe French PWR in Figure 2.

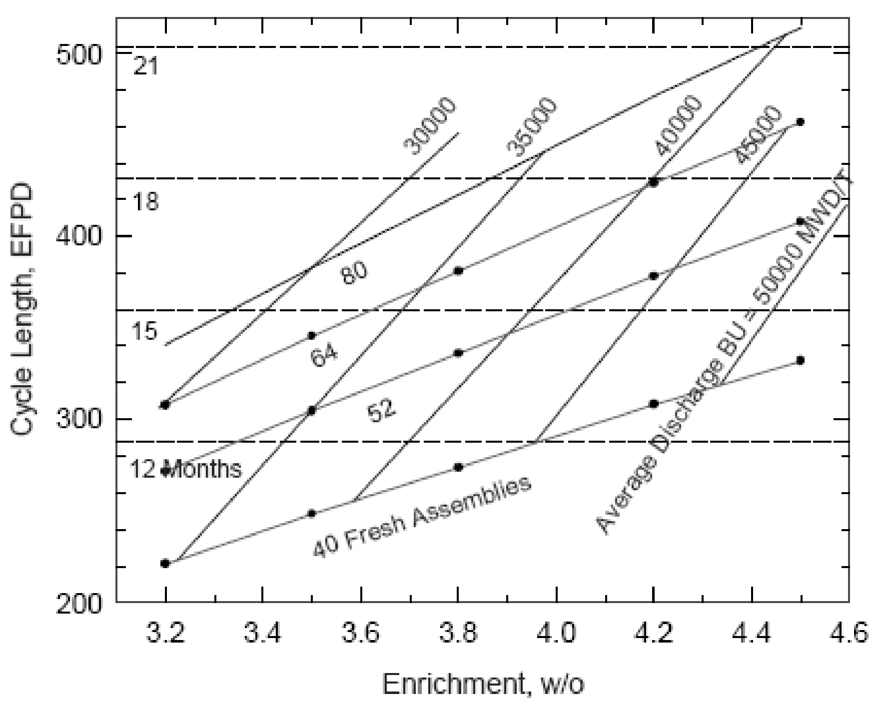

Figure 2: Cycle length and average discharge burnup vs. reload enrichment and batch size for a French 900 MWe PWR (out-in pattern); from Figure 5 in [7]

Batch refuelling can be estimated using a lattice code using the linear-reactivity model. If B_1 is the burnup corresponding to k_{inf} equalling the excess reactivity not modelled in the lattice cell (such as 0.045), and B_∞ is the burnup corresponding to on-line refuelling (e.g., infinite-batch refuelling), and B_n is the burnup corresponding to n-batch refuelling, then $(B_\infty - B_n)/B_\infty = 1- n/(n+1)$. So the burnup corresponding to 3-batch refuelling is about 25% lower than that corresponding to on-line refuelling.

In this study, 3-batch refuelling was modelled, with about a 1-year cycle length.

RFSP. RFSP version 3.5.1 is a 2-group, finite-difference diffusion theory core code. It can handle all kinds of simulations needed for CANDU reactor core design and safety analysis, including time-average simulations for reactor design, time-dependent refuelling simulations, both slow (xenon transients) and fast (such as LOCA) kinetics calculations, control and shutdown system modelling, calculations of harmonic modes, flux detector responses and flux mapping. RFSP can also couple with a thermalhydraulics code to examine the interaction between physics and thermalhydraulics behaviours, both for steady-state and transient conditions.

RFSP can also be used to model batch refuelling. Just as it can model the refuelling of individual channels during a time-dependent refuelling simulation, it can also model the batch refuelling of the entire core. For a 3-batch refuelling scheme, one third of the core is replaced with fresh fuel, one third of the fuel channels that had been irradiated for a single cycle are moved to new locations, and one third of the channels that had been in for two cycles are moved to their final locations. The batch refuelling requires the definition of the core-loading pattern, e.g., the location of fresh fuel, and the movement of once-burned and twice-burned fuel channels.

A description of the standard code set and its capabilities is given in [3]. Extensive benchmarking and validation has shown the adequacy of the toolset for CANDU reactor and ACR-1000 modelling. Because of the similarities of the PT-SCWR to these reactors, the same modelling approach can be expected to be suitable for this reactor as well. This assessment has not been done, and will be needed before detailed analysis is done. The suitability of the standard calculational approach will depend to some extent on details of the PT-SCWR, such as the lattice pitch. (Fuel channels will be more closely coupled with a tight lattice pitch, and the properties of one cell more dependent on the properties of its neighbours.) Some of the approximations that will need to be assessed include the adequacy of 2-group, coarse-mesh diffusion theory and the need for discontinuity factors. It is noted that such an assessment could be done with a new multi-group calculational scheme that can be linked with RFSP [8].

3. Lattice cell studies for a CANDU–sized fuel bundle

The first studies [9, 10] were done using the same size of fuel bundle as is currently used in CANDU reactors (and planned for ACR-1000). The LEU fuel bundle was a CANFLEX-ACR bundle, with 42 elements of the same size arranged in rings of 7, 14 and 21 elements, and a large central element containing 30 wt% Dy_2O_3 in a ZrO_2 matrix to reduce CVR (Figure 3). Pu/Th studies were done using a standard CANFLEX fuel bundle, again with Dy_2O_3 in a ZrO_2 matrix in the central element. An insulated Excel (zirconium-based alloy) pressure tube configuration was modelled in a high-efficiency HEC channel, with variations to the lattice pitch, insulator thickness, insulator porosity, and pressure tube thickness (which varied M/F) as well as the fuel enrichment. Stainless steel was used for the fuel cladding and for the liner protecting the insulator. Table 1 and Figures 4 and 5 show sample results for LEU fuel. These studies provided insights into the physics of the PT-SCWR, and represented the first step in the optimization process.

Key conclusions from these first studies are as follows:

- The axial variation in coolant void is not strongly sensitive to the axial power distribution (as tested using a cosine and flat axial power distributions).
- There is a large penalty in neutron economy (and fuel utilization) through the use of supercritical water coolant. This penalty increases with steel cladding and liner (the penalty is about 1.6% in U-235 enrichment for steel cladding and liner). Further work will be done on materials having lower neutron absorption (such as a zircaloy fuel cladding with a protective coating).
- As expected, for a given enrichment, both fuel burnup and CVR increase with increasing M/F; e.g., fuel utilization and CVR cannot be simultaneously optimized. This is the case with both LEU and Pu/Th fuel.
- With LEU fuel, Case c in Table 1 yields a burnup of 44 MWd_{th}/kg (corresponding to on-line refuelling) with an enrichment of 5%, and CVR of ~3 mk. This corresponds to a uranium utilization of 184 Mg NU/GW_e (assuming 45% thermal efficiency and on-line refuelling), compared to 152 Mg NU/GW_e for NU (with 32% thermal efficiency).
- With Pu/Th fuel, an average plutonium concentration of ~9.2% gives a burnup of 40 MWd_{th}/kg (assuming on-line refuelling) and a CVR near zero (with 10% Dy as BNA in the central element). This compares to 4.9% Pu for Pu/Th fuel in a CANDU 6 reactor, with a burnup of 45 MWd_{th}/kg and CVR between 2.7 and 5 mk [11].
- Burnups corresponding to 3-batch refuelling would be at least 25% lower than those for on-line refuelling.

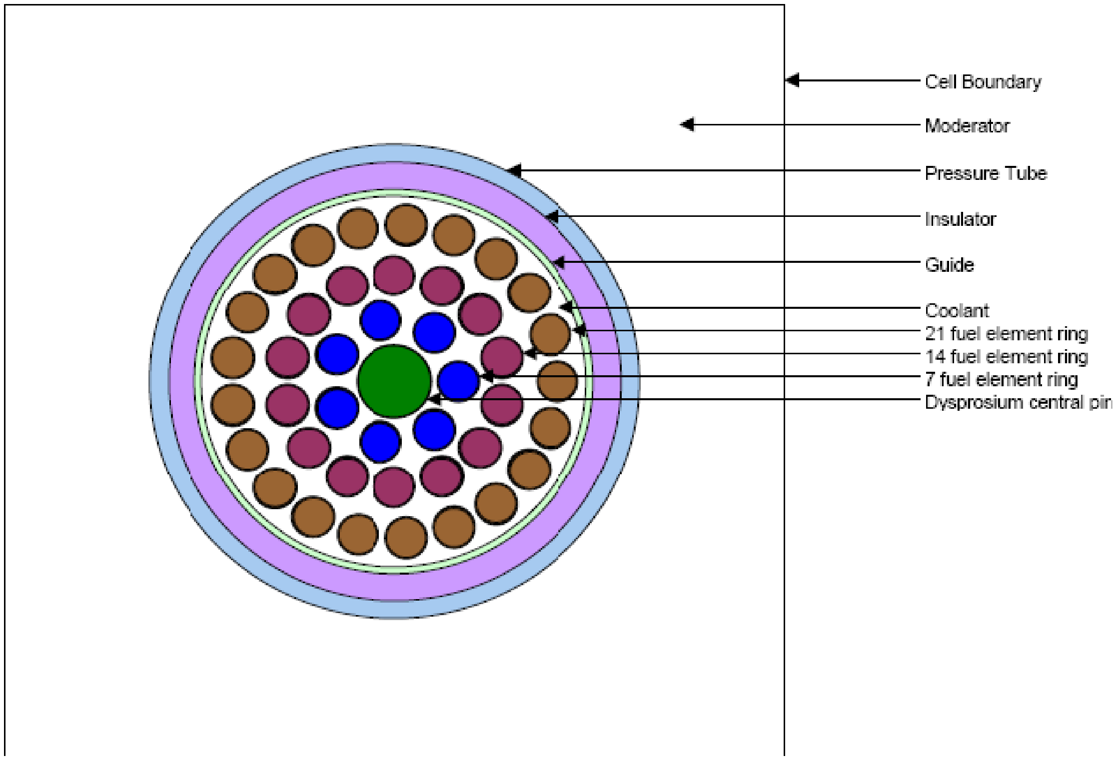

Figure 3 Schematic of HEC design with CANFLEX-ACR fuel

Table 1 Sample results for HEC design with CANFLEX-ACR LEU fuel

Case	Lattice Pitch (mm)	Insulator Porosity (%)	Insulator Thickness (mm)	U-235 enrichment (%)	Pressure Tube Thickness (mm)	Burnup* (MWd_{th}/kg)	CVR (mk)
a	200	50	3.5	4	6	18	-10.2
b	245	70	8.75	3.50	8.5	17	2.8
c	245	70	8.75	5	8.5	44	3.4
d	245	40	8.75	5	8.5	39	7.0
e	245	70	8.75	5	12.2	34	2.5
f	245	70	16.6	5	8.5	34	-11.0
g	245	99.9	8.75	5	8.5	40	-0.4
h	290	90	14	5	6	30	8.1

* Burnup corresponds to on-line refuelling; for 3-batch refuelling, discharge burnup would be at least 25% lower.

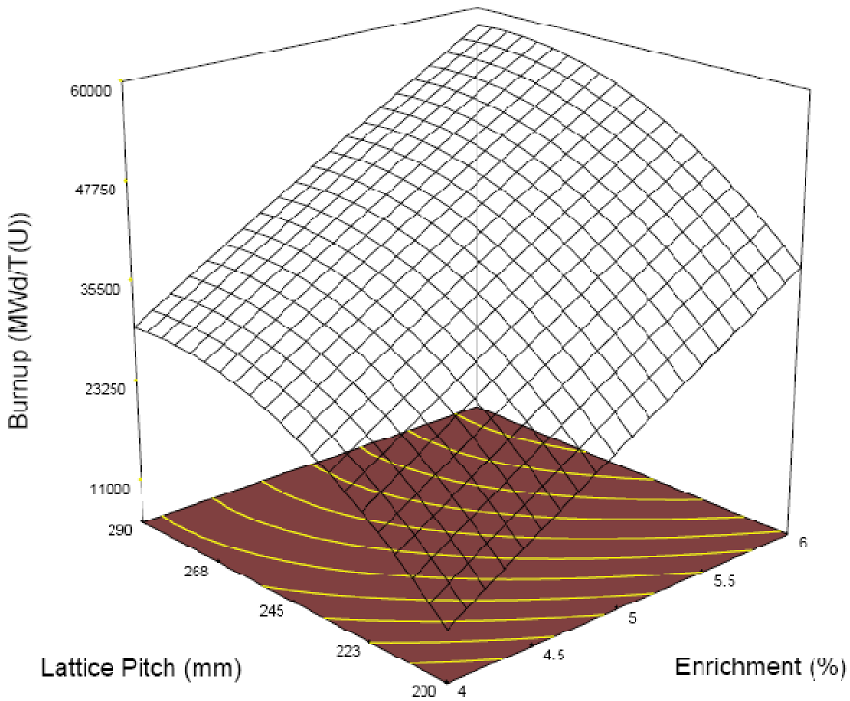

Figure 4 Variations of burnup with lattice pitch and U-235 enrichment for HEC design with CANFLEX-ACR fuel

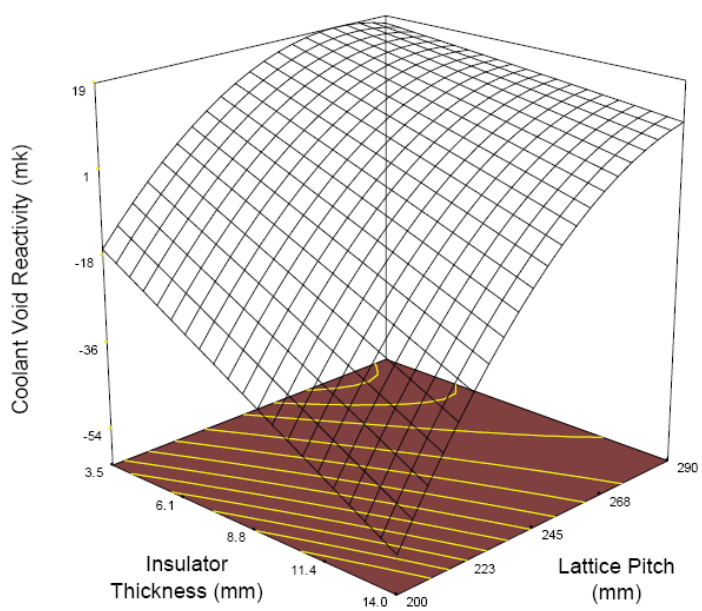

Figure 5 Variations of CVR with insulator thickness and lattice pitch for HEC design with CANFLEX-ACR fuel

4. Lattice cell studies for a larger–size fuel bundle and pressure tube

In order to improve the fuel utilization for the CANFLEX-ACR-based bundle under SCWR conditions, a significantly different design was considered. The modified bundle design (Figure 6) has 54 fuel elements, a fuel bundle rubber band radius of 6.4 cm (vs 5.0 cm with CANFLEX-ACR), and a large centre pin (filled with either air, solid material or stagnant coolant) to displace coolant. Moderator displacement tubes are optionally located at the corners of the square lattice. The figure shown is for the REC design, which is very similar in design to the HEC design. There are small differences in the thickness of the insulator in the HEC and REC designs to accommodate the 3 mm coolant annulus in the REC design. Both have the same pressure tube inner radius (7.7 cm) and thickness (0.9 cm). Some differences from the previous studies are that the thickness of the stainless steel liner was reduced from 2 mm to 1 mm, and the fuel cladding thickness reduced from ~4 mm to 3 mm. Table 2 summarizes the lattice and bundle parameters.

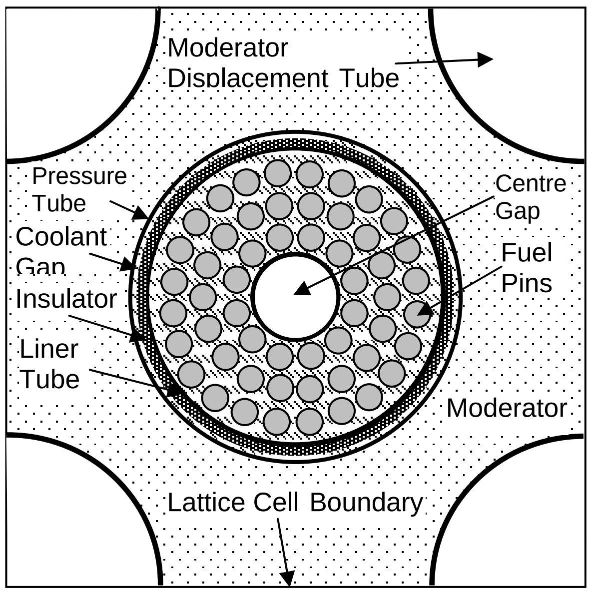

Figure 6 Schematic of the reference REC design with a larger fuel bundle size and moderator displacement tubes

The lattice-derived burnup for the reference REC design is ~41 MWd/kg (for on-line refuelling; 31 MWd/kg for 3-batch refuelling). Figure 7 shows the variations of lattice k-infinity and CVR as a function of burnup. CVR varies between -7 mk and -25 mk. While this is a more efficient lattice design than the CANFLEX-ACR bundle, it is still not optimized, as CVR is more negative than required.

Table 2 Specifications of the lattice parameters for the reference REC design with a larger fuel bundle size and moderator displacement tubes

Parameter	Value
Lattice Pitch	27 cm
Elements per bundle	55
Elements in rings 1, 2, 3	12, 18, 24
Pitch circle radius, ring 1	2.88 cm
Pitch circle radius, ring 2	4.33 cm
Pitch circle radius, ring 3	5.80 cm
Radius of central pin	1.9 cm
Outer radius of central pin cladding	2.0 cm
Radius of pins in ring 1, 2 and 3	0.61 cm
Outer radius of ring 1, 2 and 3 pin cladding	0.64 cm
Liner Tube inner radius	6.8 cm
Bundle length	49.5 cm
Liner Tube thickness	0.1 cm
Insulator inner radius	6.9 cm
Insulator thickness	0.5 cm
Outer coolant layer thickness	0.3 cm
Pressure tube inner radius	7.7 cm
Pressure tube thickness	0.9 cm
Moderator displacement tube inner radius	7.12 cm
Moderator displacement tube thickness	0.08 cm

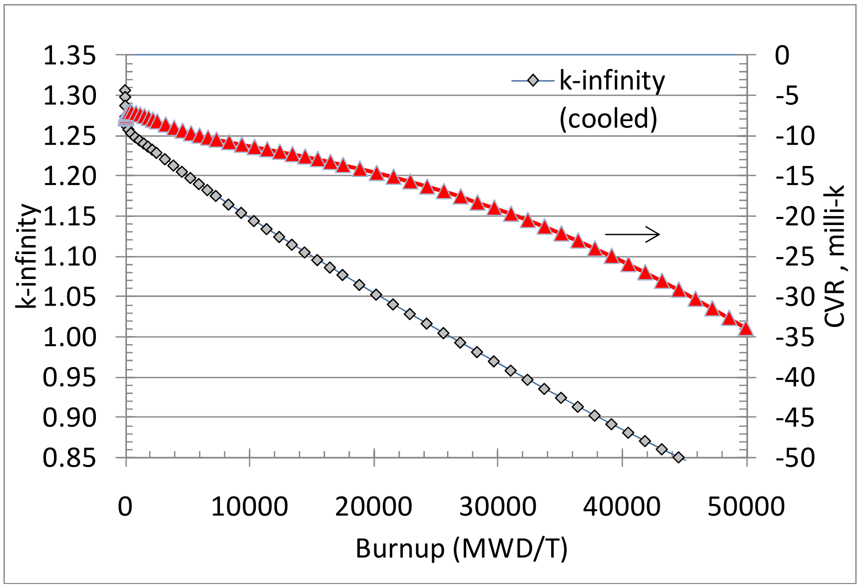

Figure 7 Lattice k-infinity and CVR as a function of burnup for the REC design with a larger fuel bundle size and moderator displacement tubes

5. Full Core Analysis

Assumptions for the key core parameters are given in Table 3. The REC reference fuel lattice was used (Figure 6), with uniform 4% enriched LEU (without BNA or reactivity devices). Five fuel types were used in the RFSP full-core model, assuming five different thermal-hydraulic parameters in the bundle axial direction. These values for the coolant temperature and density were applied to ten bundle positions along the fuel channel as given in Table 4, corresponding to a cosine power distribution.

RFSP was used to perform the 3-batch refuelling calculations. Beginning of cycle (BOC) and end of cycle (EOC) characteristics were determined for an equilibrium cycle obtained after six refuelling cycles. Various loading pattern designs were studied to reach acceptable channel power maps and bundle power maps at BOC and EOC for an equilibrium fuel cycle. The batch fuel loading pattern for the SCWR full-core is illustrated in Figure 8.

Table 3 PT-SCWR core parameters

Parameters	PT-SCWR
Fuel channels	300 vertical fuel channels
Power (both thermal and fission)*	2540 MW
Coolant	Supercritical H_2O, unidirectional flow
Coolant pressure	25 MPa
Coolant inlet temperature, density	367 ºC, 0.55 g/cm^3
Coolant outlet temperature, density	597 ºC, 0.08 g/ cm^3
Coolant nominal temperature, density	402 ºC, 0.19 g/ cm^3
Moderator	Low pressure/temperature D_2O
Lattice pitch	27 cm square
Core radius, height	335 cm, 495 cm
Average reflector thickness	65 cm
Refuelling method	3-batch refuelling

* Thermal and fission power assumed to be the same in this analysis.

Table 4 Coolant density and temperature used at the ten bundle locations along the fuel channel

Bundle Position	**Coolant Density (g/cc)**	**Coolant Temp. (°C)**	**Bundle Position**	**Coolant Density (g/cc)**	**Coolant Temp. (°C)**
10 (top of core)	0.08	597	5	0.19	402
9	0.08	597	4	0.35	387
8	0.08	497	3	0.35	387
7	0.08	497	2	0.55	367
6	0.19	402	1 (bottom of core)	0.55	367

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
A								2	1	2	2	1	2							
B							2	1	1	1	1	1	1	2						
C					1	3	1	2	2	3	3	2	2	1	3	1				
D				3	1	1	1	3	3	2	2	3	3	1	1	1	3			
E			1	1	3	2	3	2	1	3	3	1	2	3	2	3	1	1		
F			3	1	2	3	1	3	3	1	1	3	3	1	3	2	1	3		
G		2	1	1	3	1	3	2	3	3	3	3	2	3	1	3	1	1	2	
H	2	1	2	3	2	3	2	1	2	2	2	2	1	2	3	2	3	2	1	2
J	1	1	2	3	1	3	3	2	3	2	2	3	2	3	3	1	3	2	1	1
K	2	1	3	2	3	1	3	2	2	2	2	2	2	3	1	3	2	3	1	2
L	2	1	3	2	3	1	3	2	2	2	2	2	2	3	1	3	2	3	1	2
M	1	1	2	3	1	3	3	2	3	2	2	3	2	3	3	1	3	2	1	1
N	2	1	2	3	2	3	2	1	2	2	2	2	1	2	3	2	3	2	1	2
O		2	1	1	3	1	3	2	3	3	3	3	2	3	1	3	1	1	2	
P			3	1	2	3	1	3	3	1	1	3	3	1	3	2	1	3		
Q			1	1	3	2	3	2	1	3	3	1	2	3	2	3	1	1		
R				3	1	1	1	3	3	2	2	3	3	1	1	1	3			
S					1	3	1	2	2	3	3	2	2	1	3	1				
T							2	1	1	1	1	1	1	2				1	--	Fresh Fuel
U								2	1	2	2	1	2					2	--	Once-Burnt Fuel
																		3	--	Twice Burnt Fuel

Figure 8 PT-SCWR full-core map and fuel loading scheme

The key parameters from the RFSP full-core calculations for the equilibrium cycle are summarized in Table 5. The discharge burnup (28.5 MWd_{th}/kg) is consistent, although lower, with that estimated from the linear reactivity model using WIMS-AECL (31 MWd_{th}/kg).

Table 5 Summary of RFSP full-core results

Parameter	
Excess reactivity at BOC, mk	95.2
Excess reactivity at EOC, mk	9.7
Cycle length, FPD	315
Discharge burnup, MWd_{th}/kg	28.5
Maximum channel power at BOC, kW	10,075
Maximum bundle power at BOC, kW	1,423
Maximum channel power at EOC, kW	9,780
Maximum bundle power at EOC, kW	1,160

Figures 9 to 11 show the RFSP-calculated quarter-core channel power distribution, normalized channel power at the middle row, and the normalized bundle power for the channel with maximum channel power at both BOC and EOC, respectively. The peak channel power factor, defined as the ratio of the maximum channel power to average channel power, is not itself a safety parameter, but it impact safety parameters. It is judged that a value of 1.4 would be acceptable for the analysis at this stage. The peak channel power factor is 1.19 at BOC and 1.15 at EOC (Figure 9).

The inlet-to-outlet asymmetry in bundle powers (Figure 11) results from the non-uniform coolant density and temperature in the axial direction. Compared with the power shape at BOC, the power shape at EOC is flatter. As expected, at BOC the bundle power peak occurs in the region of higher coolant density. Because the burnup occurs faster in this region, the maximum bundle power shifts to a region of lower density at EOC.

The results show the reactor physics feasibility of the PT-SCWR at this early pre-conceptual design phase. Further optimization of the lattice and core design is required to improve the fuel utilization (4% enrichment gives a burnup of 28.5 MWd_{th}/kg with 3-batch refuelling). One direction for further studies will be to determine to what extent the lattice cell can be optimized for fuel utilization (maximizing the burnup for a given enrichment), while using BNA for both suppressing the reactivity of the fresh fuel during the first cycle and for reducing CVR.

0	1	2	3	4	5	6	7	8	9	10
A						BOC	----	0.94	1.07	0.97
						EOC	----	0.91	1.01	0.93
B							1.05	1.11	1.14	1.14
							0.99	1.06	1.08	1.09
C					1.16	0.93	1.16	1.01	1.01	0.89
					1.07	0.89	1.10	0.98	0.99	0.89
D				0.92	1.13	1.18	1.19	0.90	0.89	0.97
				0.87	1.06	1.11	1.13	0.90	0.90	0.99
E			1.16	1.13	0.92	1.03	0.92	1.00	1.11	0.88
			1.07	1.06	0.89	1.01	0.92	1.02	1.13	0.91
F			0.93	1.18	1.03	0.92	1.14	0.88	0.87	1.10
			0.89	1.11	1.01	0.92	1.14	0.91	0.91	1.14
G		1.05	1.16	1.19	0.92	1.14	0.90	0.96	0.84	0.84
		0.99	1.10	1.13	0.92	1.14	0.93	1.02	0.91	0.92
H	0.94	1.11	1.01	0.90	1.00	0.88	0.96	1.09	0.94	0.93
	0.91	1.06	0.98	0.90	1.02	0.91	1.02	1.15	1.03	1.02
J	1.07	1.14	1.01	0.89	1.11	0.87	0.84	0.94	0.83	0.93
	1.01	1.08	0.99	0.90	1.13	0.91	0.91	1.03	0.93	1.04
K	0.97	1.14	0.89	0.97	0.88	1.10	0.84	0.93	0.93	0.94
	0.93	1.09	0.89	0.99	0.91	1.14	0.92	1.02	1.04	1.05

Figure 9 RFSP-calculated SCWR quarter-core channel power distribution

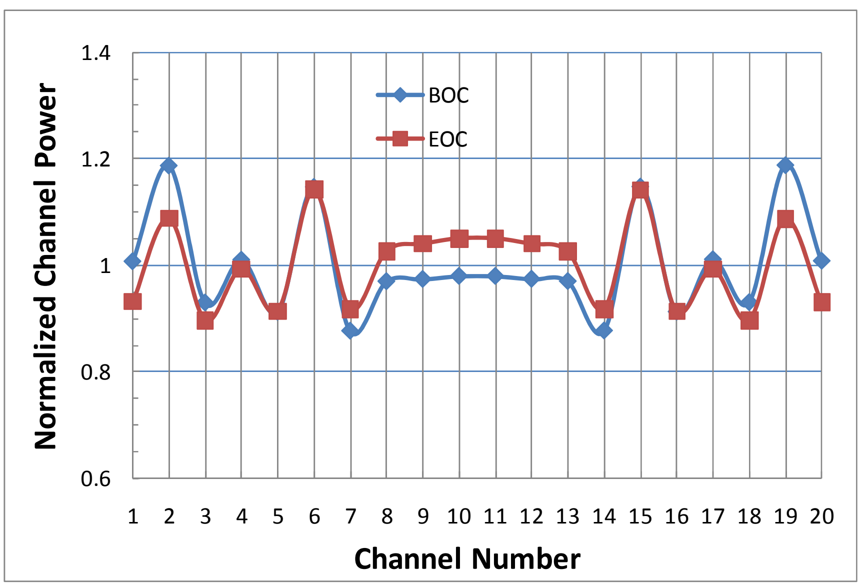

Figure 10 RFSP-calculated normalized channel power for row K

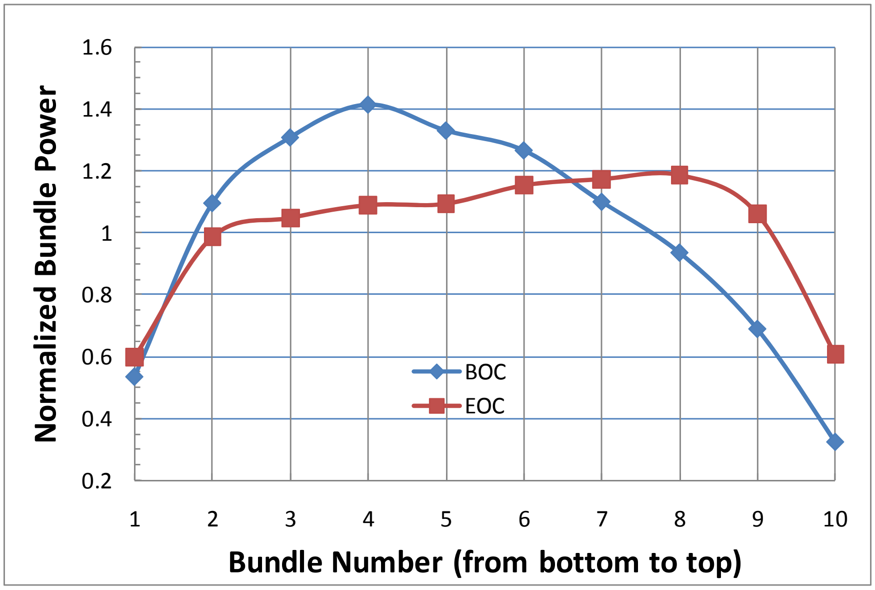

Figure 11 RFSP-calculated normalized bundle power for the channel with maximum channel power

6. Summary

Preliminary lattice physics and full core neutronic analysis have been performed for the PT-SCWR. Current CANDU reactor physics codes (WIMS-AECL and RFSP) were used for modeling this reactor. A key challenge in the physics design of this reactor is the optimization of lattice parameters to achieve the appropriate balance between CVR and fuel utilization. A vertically-oriented, batch-fuelled reactor is considered, with an insulated pressure tube to accommodate the high coolant temperatures and pressures. While the design needs further optimization, the analysis shows the reactor physics conceptual feasibility of the concept.

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