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# PHYSICS ASPECTS OF THE PRESSURE TUBE TYPE SCWR PRECONCEPTUAL DESIGN

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## Abstract

One of the best options for meeting growing global energy needs is nuclear energy, since it is both emissions free and has the potential to be a sustainable energy source. The key areas for the development of future reactors are safety, sustainability, economics and security. Heavy water moderated reactors are an appealing option because of their improved neutron efficiency, which is advantageous from a sustainability standpoint. In an evolution of the current CANDU® reactors, the pressure tube structure and heavy water moderator are retained in an advanced reactor design, cooled by supercritical light water. The use of supercritical light water as the coolant enables a large increase in thermal efficiency and therefore provides improved economic benefits. The use of thorium-based fuel provides improved safety, and a non-proliferative and sustainable fuel cycle. The optimization of the SCWR (Super Critical Water-cooled Reactor) is determined in part through reactor physics calculations, with respect to fuel utilization and safety (e.g. coolant void reactivity). These and other aspects of SCWR physics will be discussed in this paper.

## 1. Introduction

The supercritical water-cooled reactor (SCWR) is an advanced reactor concept in which water coolant is under high pressure and temperature in a supercritical state. The primary advantages of the SCWR over current reactor technologies are enhanced safety and improved thermal efficiency [1]. Enhanced safety is achieved through passive safety features such as passive decay heat removal through the moderator. The use of supercritical water coolant provides a significant gain in reactor thermal efficiency, from about 33%, which is typical for a conventional CANDU reactor, to as much as 49%, which is the expected thermal efficiency of an SCWR using reheat channels.

The concept of a supercritical water (SCW)-cooled nuclear reactor is not new [2-7]. The earliest SCW-cooled, pressure tube, heavy water moderated, reactor concept was developed in 1964 [4]. Likewise, SCW has been in use in fossil-fired power plants since 1957 ([7], and references therein). Despite interest in SCWRs in the late 1950's and 60's, a prototype was never built.

In the early 1990's, the SCWR concept was revisited, and has stimulated renewed interest, in part due to the cost savings associated with increased thermal efficiency and the potential for passive safety [8]. In 2001 a cooperative international initiative, the Generation IV International Forum (GIF), was formed in order to carry out research and development on potential next generation nuclear energy systems, with specific focus on four general goals: safety, sustainability (e.g. fuel

® CANDU (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL) utilization), economics, and proliferation resistance [9]. Canada's primary contribution to this effort is the development of a CANDU-based pressure-tube (PT) supercritical water-cooled reactor (SCWR). This reactor is intended to have an operating pressure of 25 MPa and outlet temperature of 625 °C; the coolant densities and temperatures range between 615 kg/m$^3$ and 350 °C at the inlet and 68 kg/m$^3$ and 625 °C at the outlet, respectively [1]. The PT-SCWR is also refuelled off-line, based on a 3-batch cycle and uses thorium-based fuel.

The PT-SCWR differs from conventional CANDU reactors in having large variations in coolant density along the fuel channels, much higher temperatures, an advanced channel design, thorium-based fuels and offline refuelling. All of these factors will impact reactor performance, and optimization of the physics design with respect to these features is required in order to achieve safety and economic design goals. The SCWR physics design optimization also aims to achieve the four GIF goals: enhanced safety, improved economics, sustainability and proliferation resistance. Physics calculations support these goals through optimization of reactor physics safety parameters (e.g. reactivity coefficients and kinetic parameters), optimization of the fuel bundle and fuel channel design, and optimization of the overall fuel cycle. In this paper recent developments in SCWR physics design are discussed, specifically, the development of the fuel design and the optimization of thorium-based fuel cycles for SCWR.

## 2. Preliminary Fuel Design

The fuel design has undergone a series of changes which have led to improvements in safety, fuel performance and discharge burnup. Physics modeling for the fuel design was performed using the code WIMS-AECL [10] with an 89-group nuclear data library based on ENDF/B-VII [11]. The starting point for the SCWR fuel design was enriched uranium fuel in a CANFLEX-type bundle configuration, shown in Figure 1, using dysprosium in the centre pin as an absorber to suppress coolant void reactivity (CVR) [12]. Although use of a poison such as dysprosium is an effective strategy for CVR suppression, it does incur a penalty to the fissile utilization and exit burnup, which must be countered by increased fuel enrichment. For the SCWR, this presents a problem, since there are additional penalties to fissile utilization due to the use of steel-based in-core materials [13] and batch versus continuous refuelling. Thus, it is desirable to suppress CVR without reliance on the use of a neutron poison. An alternative method for CVR suppression is via removal of the central fuel element and innermost fuel ring, as discussed in [14]. This approach has the advantage of suppressing CVR, but also reduces the amount of fuel per bundle. A 54-element bundle, shown in Figure 1, based on a 61-element design, but with the central 7 elements removed and replaced with a large centre region filled with coolant or solid material such as zirconia, was investigated in [15] and compared to the previously used CANFLEX bundle. It was found that use of the 54-element bundle gave improvements to both CVR and fissile utilization in comparison to the CANFLEX bundle.

Subsequent investigations using the 54-element bundle revealed that the pin power distribution led to undesirably high powers in the outer fuel ring of the bundle, which could lead to fuel and cladding overheating, and fission gas release [16]. To mitigate this problem, a new bundle design, shown in Figure 1, was introduced, in which the outer ring of elements was subdivided into a larger number of smaller elements. Optimization of this 78-element bundle gave values for CVR and exit burnup similar to the earlier 54-element design, but with superior performance with respect to linear element ratings, as shown in Table 1.

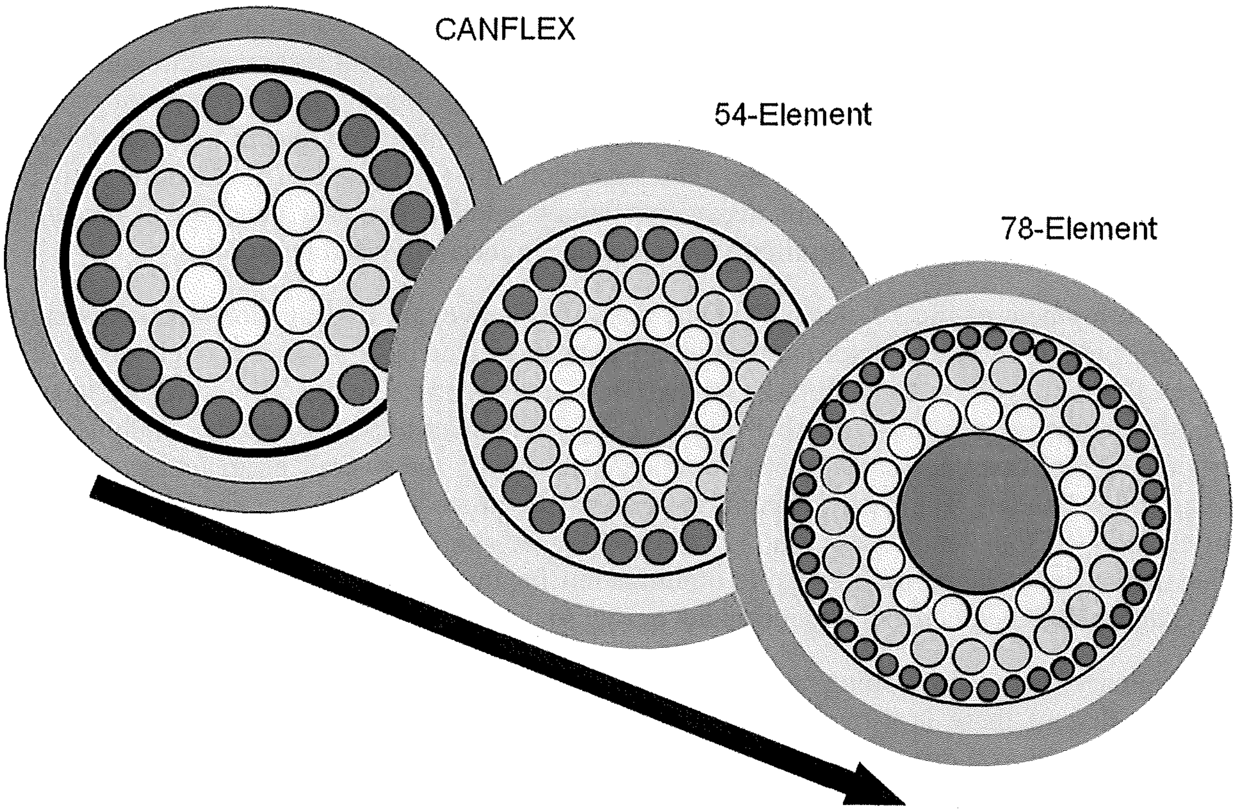


Figure 1 Evolution of the SCWR Fuel Bundle:
Left to right, the CANFLEX, 54-Element and 78-Element Bundle.

Table 1 Comparison of Performance Characteristics of the 54-Element and 78-Element Bundles

| Parameter | 54-Element | 78-Element |
|---|---|---|
| Exit Burnup (MWd/kg) | 42.1 | 43.1 |
| Coolant Void Reactivity (mk) | -2.4 | 0.1 |
| Maximum Linear Element Rating (kW/m) | 77 | 37 |
| Approximate Maximum Fuel Temperature (°C) | 2850 (possible melting) | 1400 |

## 3. Thorium-Based Fuel Cycles

Thorium-based fuel cycles are the present focus for SCWR development since the use of thorium-based fuel will yield improved performance over uranium-based fuel with respect to safety, resource sustainability, economic benefit and proliferation resistance. Thorium dioxide ($ThO_2$) chemical stability, fission product release, thermal conductivity and coefficient of thermal expansion are all superior to uranium dioxide ($UO_2$). Consequently, use of $ThO_2$-based fuels will enhance fuel safety (via reduced fuel failure) compared to $UO_2$-based fuels. Thorium is three times more abundant than uranium and is composed primarily of fertile Th-232. Thus, there is great potential for enhancing the sustainability of the nuclear fuel cycle by extending the availability of current resources through the use of thorium fuel cycles. The superior performance of $ThO_2$ at high burnup (e.g. because of reduced fission gas release) suggests the possibility of longer core residence times (compared to uranium-based fuels) which lead to potential economic gains. The formation of U-232 via (n, 2n) reactions with Th-232, Pa-233 and U-233 (and subsequent decay into strong gamma emitting daughter products) in thorium-based fuels leads to intrinsic proliferation-resistance in thorium-based fuel cycles.

Prior to the development of the 78-element bundle described above, an investigation was made using the 54-element bundle design to determine optimum parameters for both a once-through and U-233 recycle-based thorium cycle. Although these studies were performed using the 54-element design, the results are also indicative of the physics behaviour of the 78-element bundle, since the subdivision of the outer ring of fuel does not constitute a significant change in the overall arrangement of fissile material within the bundle. The studies were also performed with an alternate fuel channel design, the re-entrant channel (REC). As discussed in [17], because of the similarity in geometry and distribution of material within the channel, the fuel cycle optimization for the REC is also applicable to the high efficiency channel (HEC) which is the fuel channel design in use with the current SCWR design.

For the preliminary fuel cycle scoping work, two homogeneous thorium-based fuel cycles have been examined. The first of these is a once-through-thorium (OTT) fuel cycle, which uses a mixture of thorium and plutonium as a driver fuel. The second is a U-233 recycle (UR) fuel cycle, which uses a mixture of thorium, plutonium and U-233. Physics modeling for the fuel cycle optimization was performed using the code WIMS-AECL [10] with an 89-group nuclear data library based on ENDF/B-VII [11].

### 3.1 The Once-Through-Thorium Cycles

Optimization of the once-through-thorium (OTT) cycle was performed via the determination of exit burnup and CVR, as a function of lattice pitch and Pu enrichment. Fuel for the OTT was composed of a mixture of Pu driver fuel and Th-232. As shown in Figure 2, exit burnup increases with both increasing LP and increasing [Pu].

The increase in exit burnup with increasing LP is due to a shift in the neutron spectrum. When LP increases, neutrons encounter a longer path through the moderator, resulting in greater thermalization of the neutron spectrum. This shift of neutrons to lower energies leads to an increase in thermal fissions of Pu-239, Pu-241, and U-233, when present. The increase in fissions results a higher overall reactivity of the lattice, leading to a higher burnup for larger LP.

The increase in exit burnup with increasing [Pu] is due to positive reactivity insertion from the addition of fissile material. An increase in [Pu] corresponds to an increase in the proportion of fissile material in the fuel. The higher concentration of fissile isotopes leads to more fissions, again leading to a higher overall reactivity and consequent higher burnup.

As shown in Figure 3 the CVR becomes more positive with increases in both LP and [Pu]. The relationship between CVR and LP is essentially the same as in the case of the Advanced CANDU Reactor (ACR) lattice [18], in which the light water coolant acts as both an absorber and moderator. On voiding, loss of neutron absorption in the coolant leads to a positive contribution to CVR, while loss of moderation leads to a negative contribution. When the lattice pitch is reduced, there is less neutron moderation in the moderator, and the moderation in the coolant becomes more significant. Thus, reducing the LP drives the lattice toward an under-moderated state and negative CVR, while increasing the LP has the opposite effect.

The relationship between CVR and [Pu] is more complex and is a result of several competing effects which originate with the hardening of the neutron spectrum that results from coolant voiding. Neutron absorption in Th-232 in the epithermal energy region increases with voiding, leading to a negative contribution to CVR, which decreases with increasing [Pu]. On the other hand, voiding leads to increases in fast fissions of Th-232 leading to a positive contribution to CVR, which is reduced with increasing [Pu]. Pu-239 and Pu-241 have large capture cross sections at energies of approximately 0.3 eV and 0.25 eV, respectively, which will lead to positive contributions to CVR (since a shift to a faster spectrum will lead to a decrease in neutron absorption), which will increase with increasing [Pu]. On the other hand, these isotopes have large fission cross sections at the same energies, which should lead to concomitant negative contributions to CVR. However, these latter contributions could be balanced by positive contributions resulting from increases in fast fissions of both isotopes. A quantitative evaluation of the relative importance of these various contributions to CVR would require a dedicated study, such as described in [19], which is beyond the scope of the present work.

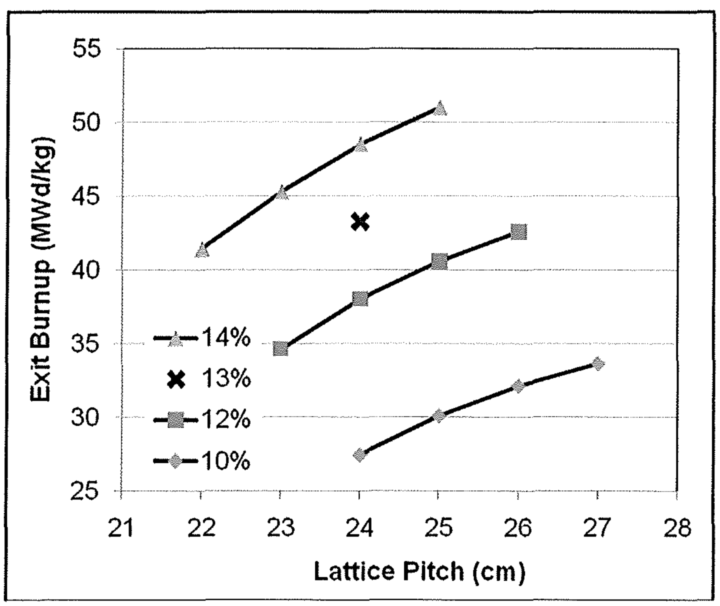


Figure 2 Channel-Averaged Exit Burnup (MWd/kg) for the OTT cycle as a function of Lattice Pitch (cm) and [Pu] (wt%).

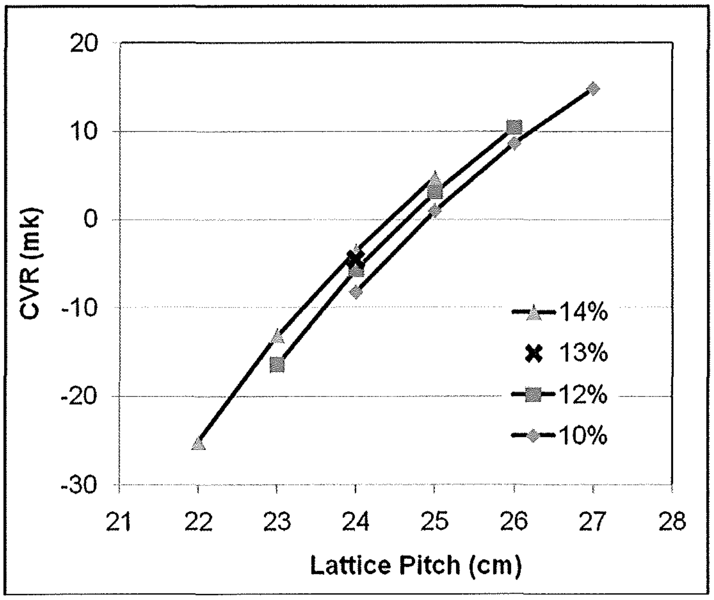


Figure 3 CVR (mk) for the OTT cycle as a function of lattice pitch (cm) and [Pu] (wt%).

These trends in exit burnup and CVR present a challenge since it is desirable to increase exit burnup while minimizing the CVR. As can be seen from the results in Figure 3, changes to LP have a larger influence on CVR (approximately +10 mk change in CVR for +1 cm change in LP) than [Pu] (approximately +2 mk change in CVR for +1% change in [Pu]), while changes to [Pu] have a much more profound influence on exit burnup (approximately +10 MWd/kg for +1% [Pu]) than changes to LP (approximately +3 MWd/kg for +1 cm change in LP). These results suggest that the best way to maximize exit burnup while minimizing CVR is to decrease the LP while increasing [Pu]. The combination of an LP of 24 cm and [Pu] of 13% leads to near optimal values for exit burnup and CVR, 43 MWd/kg and -4.5 mk, respectively. For the HEC option, adjustment to approximately the same fuel-to-moderator ratio requires an increase in LP to 25 cm. WIMS-AECL calculations for the HEC channel with an LP of 25 cm and [Pu] of 13% wt give a CVR of -6.9 mk, but an exit burnup of only 40 MWd/kg. Increasing [Pu] to 14% wt gives near optimal values for exit burnup and CVR, 45 MWd/kg and -5.9 mk, respectively.

### 3.2 U-233 Recycle

For the U-233 recycle studies, the fuel was composed of a mixture of Pu driver fuel, U-233 and Th-232. As was observed in the OTT cycle, exit burnup increases with both increases in lattice pitch and increases in [Pu] or [U-233]. Likewise, CVR becomes more positive as both lattice pitch and [Pu] or [U-233] increase. The physics phenomena driving the variation in exit burnup and CVR for the UR cases are expected to be essentially the same as those discussed in the OTT cases and are not discussed further. It was possible to achieve a cycle that was self-sustaining in U-233 using fuel with initial concentrations of 8% Pu and 2.1% U-233, and a lattice pitch of 25 cm. The corresponding exit burnup and CVR were 44 MWd/kg and -6.4 mk, respectively. Assuming that the reactors and fuel bundle design used for the OTT cycle are the same as for the U-233 recycle, approximately 1.5 OTT reactors are required to produce enough U-233 to begin the U-233 recycle in one reactor.

## 4. Core Physics Design

The present SCWR pre-conceptual core design consists of 336, 5 metre long fuel channels. The proposed refuelling scheme for the SCWR is a three-batch scheme. One third of the core is replaced with fresh fuel at the end of each operating cycle, another third of the core contains once-irradiated assemblies, and the remaining third contains assemblies that have been in core for two cycles. The locations of these fresh, one and two cycle assemblies are determined by the fuel loading scheme shown in Figure 4. A typical goal of designing such a scheme is to ensure an even power distribution radially across the core, that is, reducing the radial power peaking factor (PPF), defined as the ratio of maximum channel power to average channel power for the reactor. For the proposed reactor power of 2540 $MW_{th}$, the average channel power will be 7560 kW. At this stage, no reactivity devices have been modelled nor has any method, such as addition of a burnable neutron absorber to the fresh fuel or moderator, been employed to suppress the initial excess reactivity.

The full core was modeled using the two-group, three-dimensional neutron diffusion code RFSP version 3.5.1 [20]. Cell averaged cross-sections for the 54-element bundle design were used as input to the RFSP model and were determined using the code WIMS-AECL [10] with an 89-group nuclear data library based on ENDF/B-VII [11]. Axial properties of the full length fuel assemblies are determined by treating the fuel as ten 0.5 m length sections. This accounts for changes in neutronic behaviour due to variation in coolant properties along the fuel channel.

The results of RFSP calculations are summarized in Table 2 and show that the present fuelling scheme produces a relatively even radial power distribution with a radial power peaking factor of 1.28. It is expected that further refinement to the fuelling scheme, in combination with BNA addition to fresh fuel and reactivity devices will reduce the radial power peaking further. The axial power profile of the highest power channel is shown in Figure 5. This channel is a fresh fuel channel, C10 in the quarter-core map in Figure 4. The power of this channel is 9648 kW at BOC and the axial power peaking factor is approximately 1.4. Axial gradation of the fuel enrichment and axial distribution of BNA may be used to further reduce the axial power peaking factor.

Table 2: Summary of full-core model results

| **Parameter** | **Value** |
|---|---|
| Cycle Length | 610 FPD |
| Excess reactivity at Beginning of Cycle (BOC) | 96 mk |
| Excess reactivity at End of Cycle (EOC) | 9 mk |
| Maximum Bundle Power (BOC) | 1311 kW |
| Maximum Bundle Power (EOC) | 1034 kW |
| Maximum Channel Power (BOC) | 9648 kW |
| Maximum Channel Power (EOC) | 8879 kW |

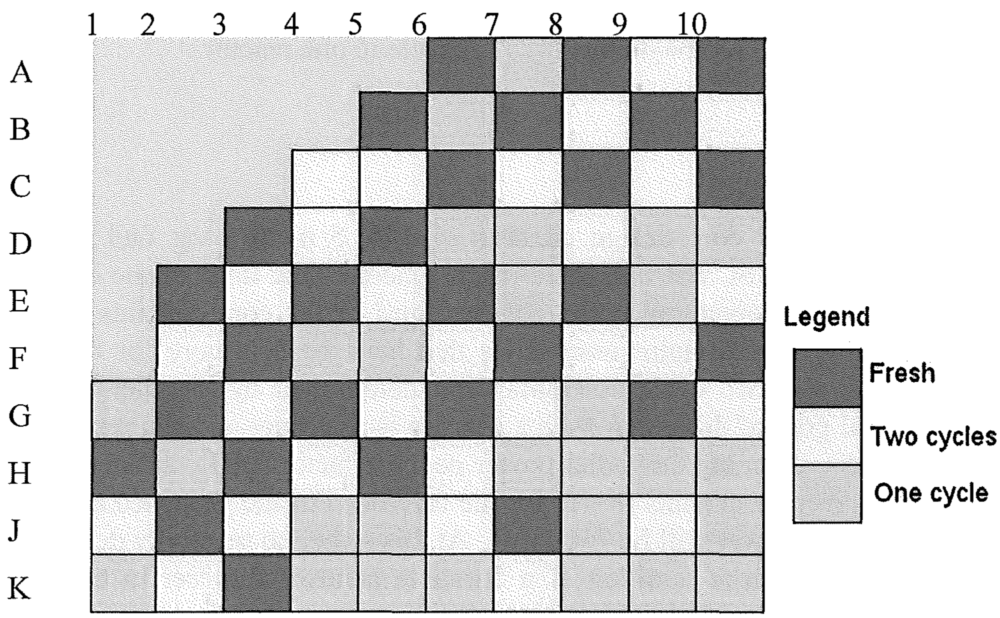


Figure 4: Quarter core fuel loading pattern

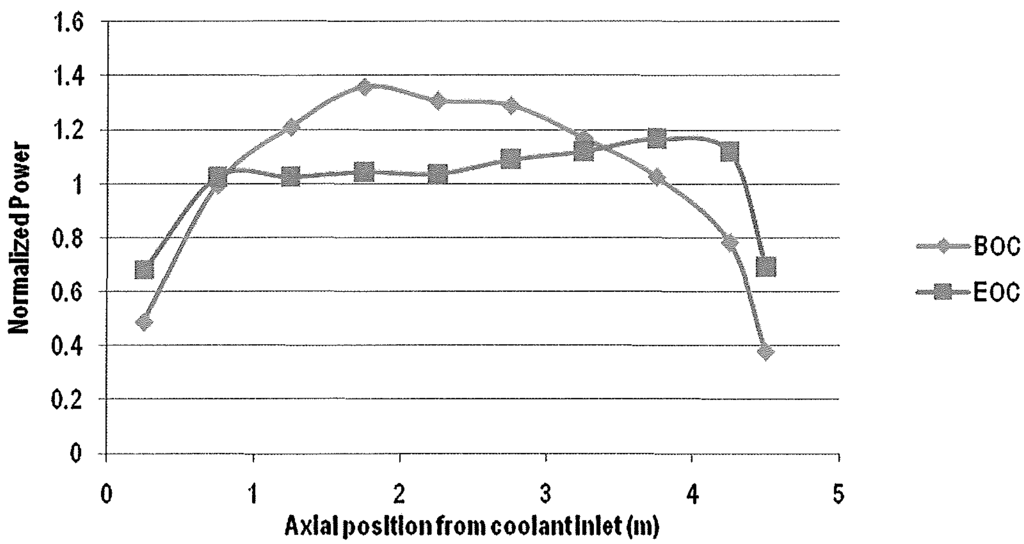


Figure 5: Axial power profile of maximum power channel

## 5. Summary

The fuel design has undergone a series of changes, starting with a CANFLEX-based bundle design, progressing to a 54-element bundle and finally a 78-element bundle, which is the design currently under consideration for SCWR. This series of changes has led to improvements in safety, fuel performance and discharge burnup. The homogeneous fuel cycle scoping work has provided fuel compositions and lattice geometries that provide optimum values for both exit burnup and CVR. However, as noted above, for the present homogeneous fuel cycles, approximately 1.5 OTT reactors are required to produce enough U-233 to begin the U-233 recycle in one reactor. The impact of axial and radial graded enrichment has not yet been examined and may lead to improvements in both U-233 production and fissile utilization. Preliminary optimization of the core loading scheme has also been performed, and further improvements to the power peaking factors may also be possible through graded fuel enrichment, the use of BNA and the inclusion of reactivity control devices.

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