A NEXT GENERATION HEAVY WATER NUCLEAR REACTOR WITH SUPERCRITICAL WATER AS COOLANT

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Abstract

The supercritical water (SCW) cooled nuclear reactor is one of six candidate reactor concepts selected by the Generation-IV International Forum (GIF) for meeting GIF design goals, which includes enhanced safety, resource sustainability, economic benefit and proliferation resistance [1]. As a member of GIF, Canada is developing a pressure-tube type SCW reactor, which has the potential to fulfill all major GIF goals. Employing the existing supercritical water technology used in coal plants on the balance-of-power systems, the Canadian SCWR design effort focuses mainly on the core configuration to generate supercritical water at the temperature of 625°C matching closely the existing high-pressure turbine design (thereby increasing the thermal efficiency by 40% as compared to the conventional nuclear power plants).

This paper presents the pre-conceptual design of a heavy-water moderated pressure-tube type SCWR, which has evolved from the well-established CANDU reactor. It describes the core configuration, advanced thorium fuel cycle, the high-efficiency fuel channel design, passive moderator cooling system, and fuel design. These key components facilitate the fulfillment of the GIF design goals.

1. Introduction

AECL is designing the Canadian SCWR, which has evolved from the well-established pressurized-channel type CANDU®[1] reactor. The general concept is discussed in [2]. The Canadian SCWR will produce electrical energy as the main product, plus process heat, hydrogen, industrial isotopes, and drinking water (through the desalination process) as supplementary products, all within a more compact reactor building. Another potential application of the available co-generated process heat is the extraction and refining of oil sands, which is presently achieved using co-generation with natural gas turbines and process heat. The extraction and upgrading process requires: thermal power to lower the viscosity and extract the oil; electric power for separation and refining equipment; and hydrogen gas for upgrading the oil product prior to transport [3].

A National Program has been established in Canada to support R&D studies for the Canadian SCWR design [4]. It covers key areas of interest (such as thermal hydraulics, safety, materials, and chemistry) to participants in the Generation-IV International Forum (GIF) SCWR designs. Results generated by the program are contributed to the GIF SCWR project management boards (PMBs). For example, heat transfer correlations have been derived using experimental data primarily obtained from fossil-plant related studies (which were started as early as 1930s [5], [6]). Materials and chemistry studies have evolved based on operating experience from fossil-fired power plants to a) develop, and perform targeted testing of, materials for key components, in particular in-core reactor components that will be exposed to conditions (irradiation, water radiolysis) not encountered in a fossil-fired boiler, and b) develop a suitable water chemistry to minimize corrosion and corrosion product transport.

[1] CANDU – Canada Deuterium Uranium, a registered trademark of Atomic Energy of Canada Limited (AECL).

Some of the advanced features of the proposed Canadian SCWR are as follows:

1. (Improved) Passive Safety - Passive decay heat removal based on natural circulation and radiation cooling is used to mitigate accident scenarios. No-core-melt goal is likely achievable by using passive decay heat removal thus assuring that fuel melting does not occur, even if all emergency injection systems fail, and containment integrity is not challenged;
2. (Sustainable and Proliferation Resistant) Thorium Fuel Cycle – The Canadian SCWR is being specifically designed with the capability to operate with sustainable fuels, namely thorium-uranium-233, and thorium-plutonium reference fuel cycles, while burning up excess plutonium and significantly reducing spent fuel amounts and heat loads in a proliferation resistant fuel cycle;
3. (Improved) Economics – At SCW operating conditions, thermodynamic cycle efficiency increases significantly. Up to 50% increase in cycle efficiency as compared to current nuclear power plants is possible, resulting in reduced generating cost.

2. Thermodynamic Cycle

The proposed thermodynamic cycle of the Canadian SCWR closely matches the current advanced turbine configuration of SCW fossil power plants. High-pressure SCW from the reactor core is directly fed into SCW turbines. This direct cycle is also used in boiling-water reactors (BWRs) at lower pressures and temperatures. The direct cycle facilitates the implementation of high pressures and temperatures leading to improved thermodynamic efficiency. It also simplifies the system by eliminating the need to transfer energy to a secondary cycle via a steam generator and its associated components. The Canadian SCWR thermodynamic cycle is designed for high-pressure turbines operating at a pressure of 25 MPa and temperature of 625°C [7]. Variants of the SCWR thermodynamic cycle currently under consideration include a direct cycle either with or without the option for reheat channels, and a dual cycle which could also be used with or without the reheat option. These variants are discussed below.

The direct cycle is equipped with a moisture separator reheater (MSR) to reduce the steam moisture inside the low pressure turbines. A schematic diagram of the direct cycle is shown in Figure 1 [8]. The heat transport system (HTS) coolant (SCW) flows directly to the high pressure and intermediate-pressure turbines. Some moisture is anticipated at the exhaust of the intermediate-pressure turbine. An MSR is installed at locations between the intermediate pressure and low-pressure turbines. It separates the moisture from the steam and reheats the steam to ensure an acceptable moisture level at the inlet of the low-pressure turbine. The temperature and pressure of the coolant at various stages in the cycle are also shown in Figure 1.

It is envisioned that the Canadian SCWR thermodynamic cycle design will eventually take full advantage of the steam reheat option used in fossil power plants, raising the outlet steam temperature of the reheat channels to the 625°C range at a lower pressure of 6.2 MPa prior to entering the intermediate pressure turbine. This reheat core pass increases the efficiency further and eliminates the need for the MSR. Figure 2 illustrates the SCWR layout and thermal cycle with the reheat option and shows the temperature and pressure of the coolant at various stages in the cycle.

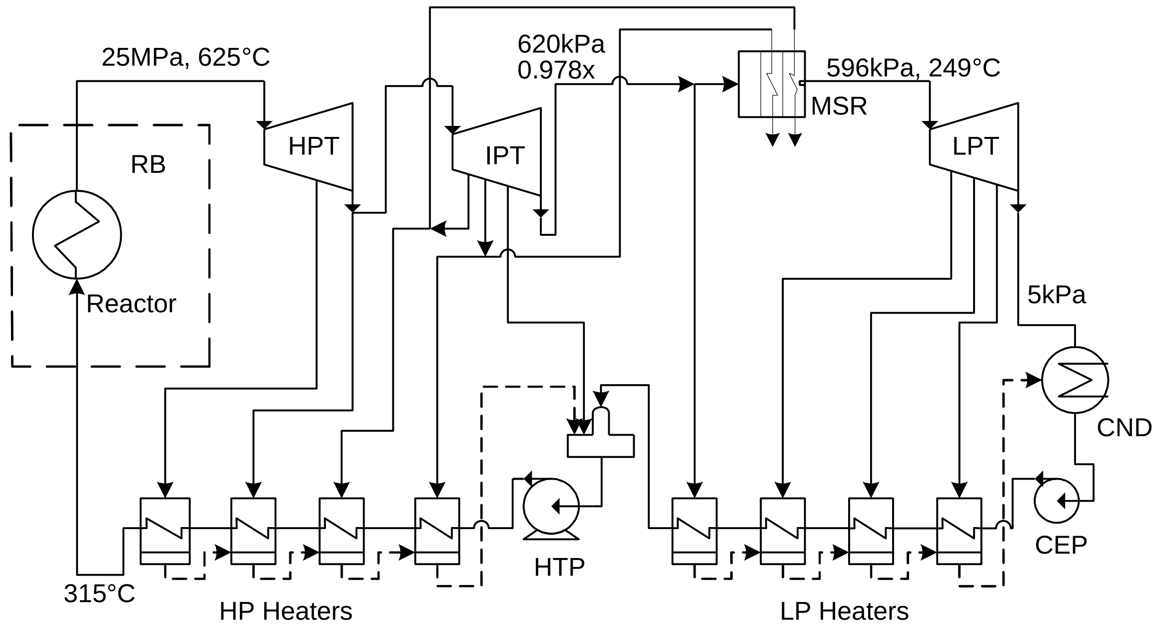

Figure 1 Schematic of Direct Cycle with an MSR in an SCWR Plant.

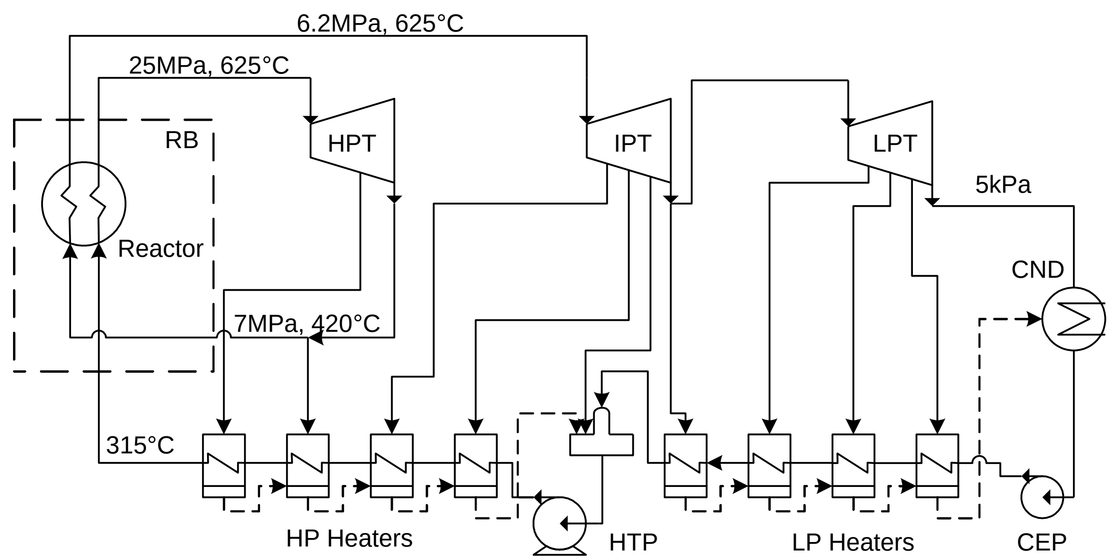

Figure 2 Schematic of Direct Cycle with Reheat in an SCWR Plant.

To match a SCWR to a reheat SCW turbine, the flow from the back end of the high-pressure turbine must be returned at a lower pressure through the core in the second pass. The steam is then reheated to the required superheat and fed to the intermediate-pressure section of the turbine. At the exit of a pressure-tube type reactor, the target HTS coolant temperature can be established by either extending the channel length or increasing the number of passes through the core. Superheat channels are placed at the periphery of the reactor core and have about 1.5 times lower heat flux compared to the average heat flux.

The sizes of the high-pressure and intermediate-pressure turbines are relatively small compared to the low-pressure turbine. This provides an opportunity to simplify the layout, with all high-pressure sections placed inside the reactor building, while the low pressure turbine can be located outside the main containment or reactor building.

Other than the direct cycle configuration, an indirect (or Dual) cycle can also be implemented using a heat exchanger that separates the primary side fluid from the secondary side fluid [9]. Overall thermal efficiencies for various configurations are compared in Table 1 for a typical SCWR of 2540 MW(th) output. A maximum calculated efficiency of 50% is achievable using the direct cycle with reheat option. This represents an improvement of about 40% in efficiency over current LWR designs (about 33% efficiency).

Table 1 Summary of typical predicted efficiencies (25 MPa/625°C/2540 MW(th)).

Cycle	Direct	Direct	Dual	Dual
Option	Reheat	MSR	Reheat	MSR
Reheat (MPa/°C)	6.1/625		6.1/625	
Efficiency %	50	49	49	47

3. Pre-Conceptual Core Design

The pre-conceptual Canadian-SCWR maintains a modular design with separated coolant and moderator, as in current CANDU reactors. For this reactor, the current CANDU practice of on-line refuelling is extremely challenging because of the significantly higher operating pressure and temperatures. Therefore, batch refuelling has been adopted, which leads to a simplified vertical core design with vertical fuel channels, each containing a fuel assembly. Figure 3 illustrates schematically the pre-conceptual Canadian SCWR core design.

The pre-conceptual Canadian SCWR design consists of 336 fuel channels, each housing a 5 m long fuel assembly. It is designed to generate 2540 MW of thermal power and about 1200 MW of electric power (assuming a 48% thermodynamic cycle efficiency of the plant). The average fuel channel power is 6.5 MW(t) and the core radial power profile factor is estimated to be 1.28. The lattice pitch is selected to be 250 mm based on recent optimization results for the fuel to moderator ratio to achieve a negative void coefficient, and high fuel burnup [10]. Figure 4 illustrates the cross-section layout of the Canadian SCWR core and Figure 5 shows the layout of fuel channels. Some fuel channels at the outer region of the core could be used for the reheat option.

As shown in Figure 4, the light water coolant enters the inlet plenum, through inlet nozzles (inlet pipes are not shown) and then enters the fuel channels, which are connected to the tubesheet at the bottom of the inlet plenum. A plenum is feasible for the core inlet due to the relatively low coolant temperature despite the high pressure. The top of the inlet plenum is removable for refuelling. The tubesheet at the bottom of the inlet plenum is machined to form a square array of holes about the same size as the pressure tubes. The pressure tubes, made of a zirconium alloy, are attached to the tubesheet using a well established rolled joint technique [11], which provides a leak-tight connection.

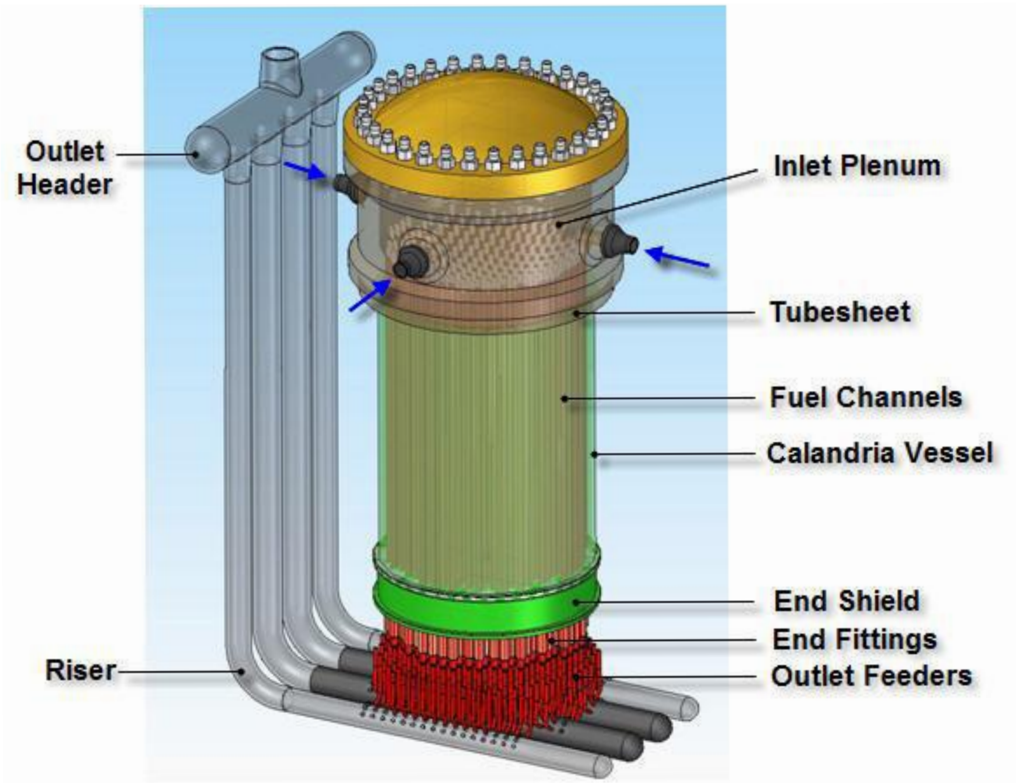

Figure 3 Schematic Diagram of the Pre-Conceptual Canadian SCWR Core Design.

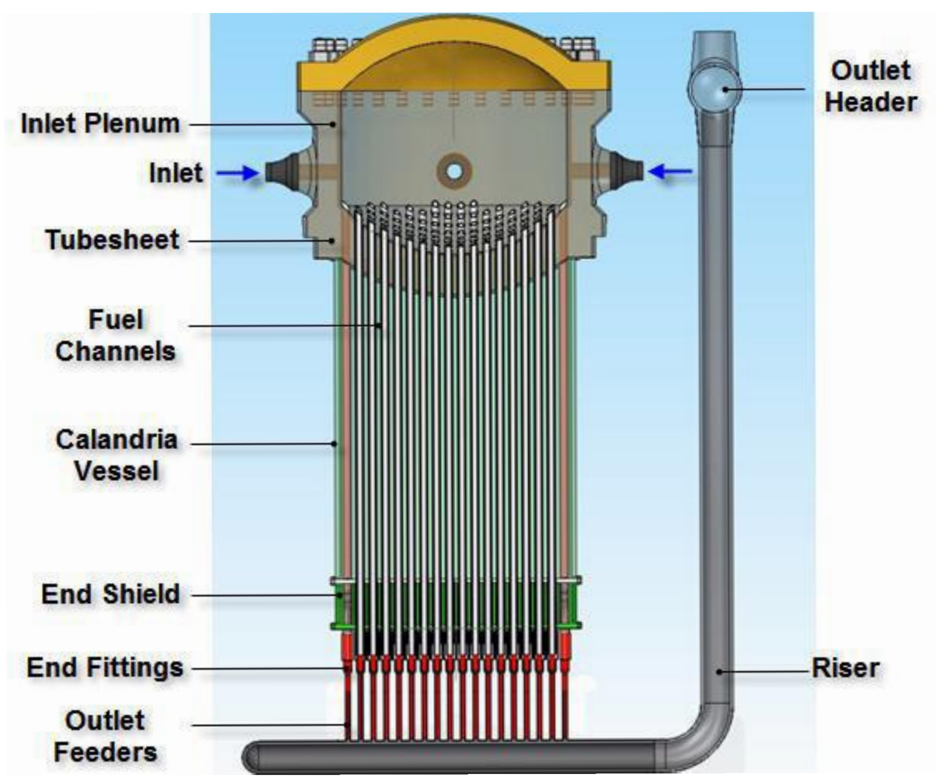

Figure 4 Cross Sectional View of the Pre-Conceptual Canadian SCWR Design.

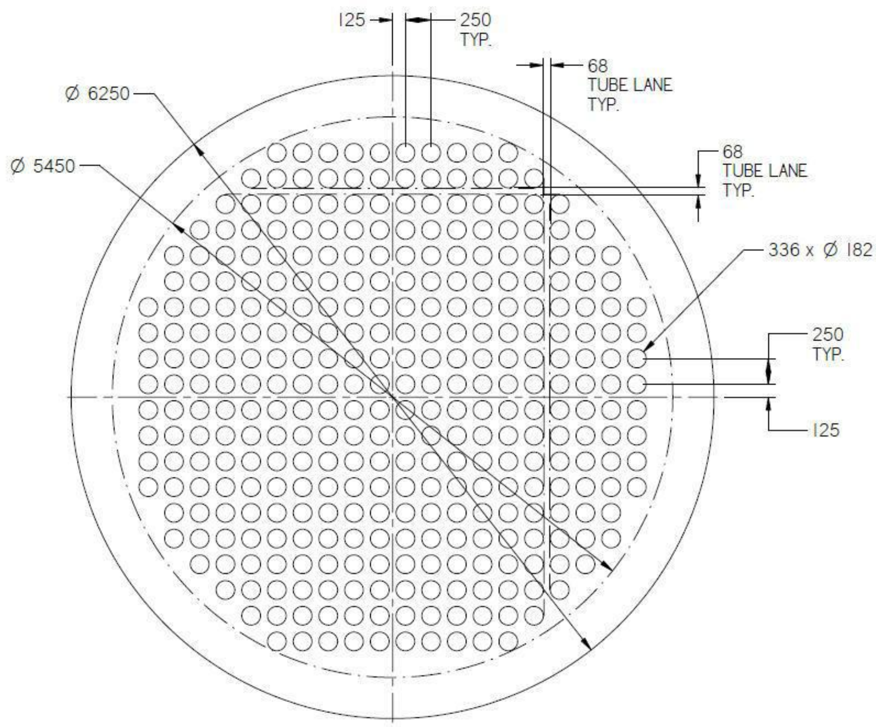

Figure 5 Cross-Section Layout of Fuel Channels in the Canadian SCWR Core.

Because the pressure losses in fuel channels are significantly larger than in the inlet plenum, inlet coolant will tend to divide reasonably uniformly into the fuel channels. Finer control of the flow rates in each individual channel would be achieved using orifices in the fuel channels for the purpose of achieving a more uniform exit temperature distribution. Light water coolant from inlet plenum flows down the fuel channel, cooling the fuel assembly and extracting heat. A down-flow configuration for the fuel channel is selected to simplify the refueling process while avoiding the plenum from getting exposed to hot and potentially oxidizing exit coolant conditions. The coolant exiting from the fuel channels is collected in the outlet header at an average temperature of 625ºC chosen specifically to match SCW turbines being used and developed for thermal power plants.

The calandria vessel holds the low-pressure and low-temperature heavy-water moderator surrounding the fuel channels. It is a relatively low-pressure tank containing the fuel channels, moderator, reactivity control mechanisms and emergency shutdown devices. Control and shut-down rods are installed from the side of the calandria vessel. When necessary, these rods would be inserted downward due to gravity at an angle toward the other side of the vessel (a spring-assisted inserting movement is also being considered). A second shut-down system would also be installed providing gadolinium injection at various levels of the calandria vessel. The conceptual development and positioning of the control rods, shut-down rods, and gadolinium injection nozzles are in progress. The end shield at the bottom of the reactor is a neutron reflector filled with spherical steel balls as in current CANDU designs.

The reference fuel-channel design consists of a pressure tube, an insulator, and a liner tube (Figure 6). This fuel channel design is called the high-efficiency channel (HEC) [12]. The current reference fuel channel dimensions are listed in Table 2. The pressure tube is designed to withstand the high coolant pressure, but directly contacts the moderator, thereby maintaining it at a low temperature (~100^{o}C). This allows the use of the zirconium alloy Excel for the pressure tube. This alloy has superior properties at low temperatures compared to other zirconium alloys. A stainless-steel liner is placed between the fuel bundle and the insulator, minimizing potential damage to the insulator by the bundle. The insulator thermally protects the pressure tube from the higher temperature bulk fluid flowing through the fuel bundle. It is made of Yttrium-Stabilized Zirconia (YSZ), which is refractory, has low neutron absorption properties and excellent resistance to neutron damage. One of the possible benefits of using the HEC is that in the event of a loss of coolant accident (LOCA) without emergency core cooling, the fuel may not melt because of passive heat rejection through the insulator into the moderator [13]. That is, the heat in the fuel will be radiatively transferred to the liner tube and conducted to the moderator, maintaining the fuel cladding below its melting point. To achieve this safety goal the fuel channel requires further optimization of its geometry, strength, corrosion resistance, creep, thermal resistance and mechanical properties to ensure sufficient decay heat removal during accidents conditions while minimizing heat loss during normal operating conditions.

The fuel channel outlets are connected to small diameter outlet feeder pipes through transition pieces that are rolled into the pressure tube on one end and welded to the outlet feeder pipes on the other end. The rolled joint is placed inside the calandria vessel and is in direct contact with the moderator to keep it cool. The YSZ insulator extends about 300 mm beyond the rolled joint to ensure that there is a reasonable temperature gradient between the region where the HTS coolant exits past the insulator and the region/elevation of the rolled joint. An expansion bellow at the end of the fuel channel allows for axial thermal expansion and creep growth of the fuel channel.

Outlet feeder pipes provide flexibility to the fuel channel/outlet pipe structures to accommodate any differential growth of the fuel channel due to material creep and thermal expansion. The large diameter outlet pipes, called risers, are anchored such that their downward thermal expansion is about the same as the thermal expansion of fuel channels. This configuration minimizes in-service stresses of both the outlet piping and the fuel channels. Outlet flow from risers is collected in the headers before it is transported to the turbines. Possible candidate materials for the outlet feeders include the T/P91 and T/P92 class of high-temperature alloys commonly used in conventional supercritical coal plants. However, flow accelerated corrosion may be an issue with these class of materials. As an alternative, nickel based high temperature super alloys are considered.

Since the SCWR is a water-cooled reactor, a key challenge will be to define a water chemistry control strategy. The properties of water change significantly as it passes through the critical point, and while there is a large water chemistry experience base in the fossil-fired SCW power generation industry, the significant structural and operational differences between a fossil-fired boiler and an SCWR core make the direct adoption of fossil plant practices problematic. The key chemistry issues are a) water radiolysis, which can produce oxidizing species that can increase the corrosion of reactor components in-core and immediately downstream of the core, and b) corrosion product deposition on in-core materials, which can lead to overheating and fuel failure [14]. These issues are the subject of a number of research programs now underway in Canada.

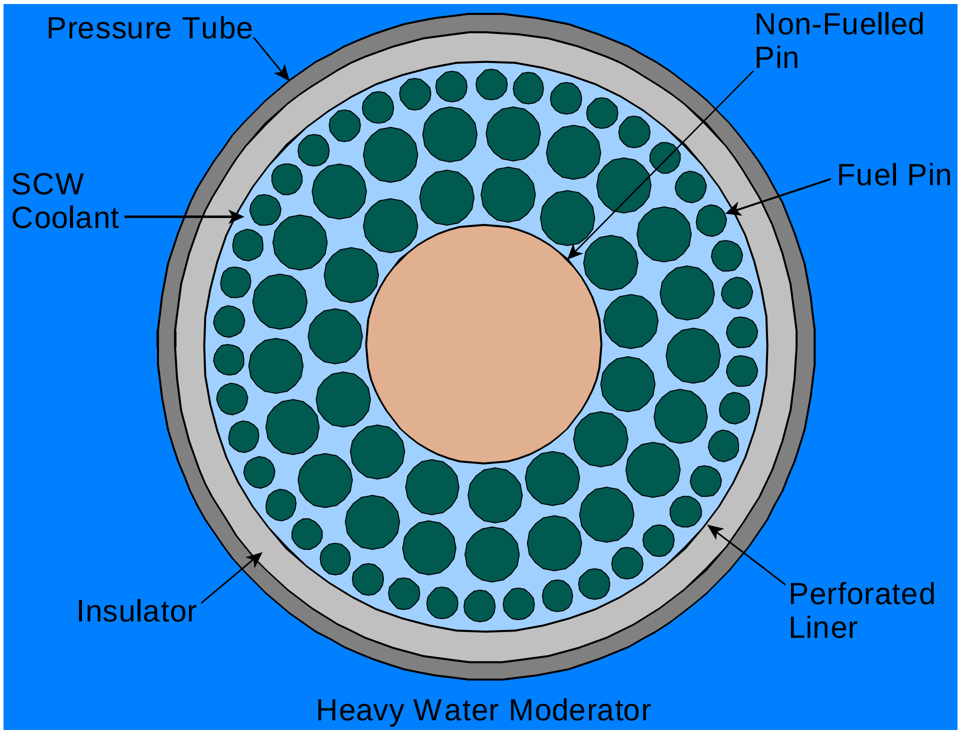

Figure 6 Schematic Diagram of the High Efficiency Channel.

Table 2 Pre-Conceptual HEC Dimensions.

	Fuel Channel Dimensions
Liner Inside Diameter (mm)	136
Liner Thickness (mm)	0.7
Insulator Thickness (mm)	10
Pressure Tube Thickness (mm)	12
Pressure Tube Outside Diameter (mm)	181.4

4. Advanced Fuel Cycles

The GIF goals for the development of next-generation reactors include enhanced safety, resource sustainability, economic benefit and proliferation resistance. Each of these goals can be addressed through the implementation of thorium fuel cycles. In particular, 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. Recent studies of thorium-based fuel cycles in contemporary CANDU reactors demonstrate the possibility for substantial reductions in natural uranium (NU) requirements of the fuel cycle via the recycle of U-233 bred from thorium [15-18]. As thorium itself does not contain a fissile isotope, neutrons must be provided by adding a fissile material, either within or outside of the thorium-based fuel. This fissile isotope is typically enriched uranium, U-233 (which is bred from an earlier thorium cycle) or reactor-grade plutonium.

Thorium fuel cycles are categorized by the type of added fissile material and are also significantly influenced by the way in which the fissile and fertile materials are distributed within the fuel bundle and within the core. The simplest of these fuel cycles are based on homogeneous thorium fuel designs, where the fissile material is mixed uniformly with the fertile thorium. These fuel cycles can be competitive in resource utilization with the best uranium-based fuel cycles, while building up an inventory of U-233 in the spent fuel for possible recycle in thermal reactors. When U-233 is recycled from the spent fuel, thorium-based fuel cycles can provide substantial improvements in the efficiency of energy production from existing fissile resources. Options for once-through and U-233 recycle thorium fuel cycles are currently being investigated and optimized for the Canadian SCWR design [19].

The proposed refuelling scheme for the SCWR is a three-batch scheme, i.e., 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. Locations of these fresh, one cycle and two-cycle assemblies are determined by a fuel loading scheme. 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 [10].

At this stage of research and development, no reactivity devices have been modelled nor has any burnable neutron absorber (BNA) been added to fresh fuel or moderator for reactivity suppression. Figure 7 shows the refuelling scheme used for the analysis. This scheme produces a relatively even radial power distribution with power peaking factor of 1.28 and was used in the subsequent linear-element rating (LER) analysis. 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.

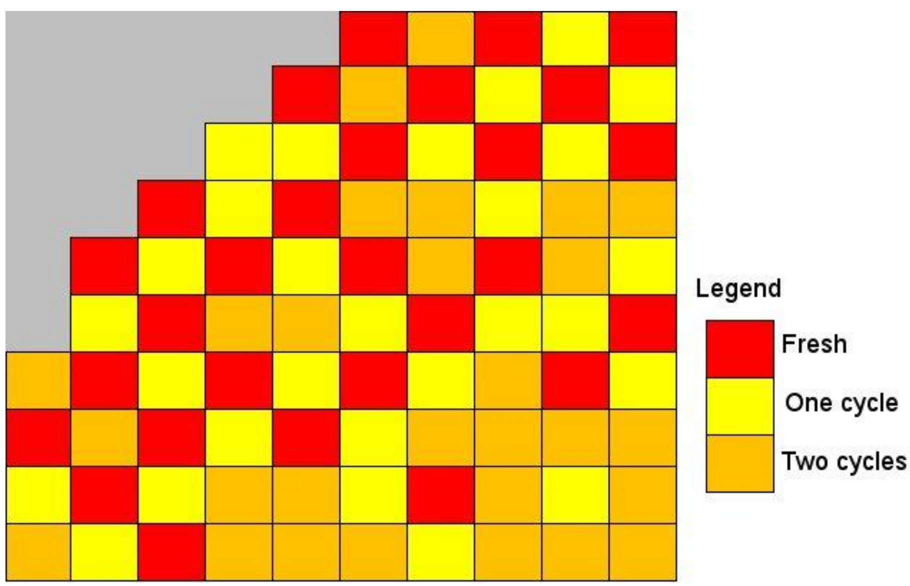

Figure 7 Quarter core fuel loading pattern.

The axial power profile used for the LER analysis of the high power channel is shown in Figure 8. This channel was chosen so that the highest LERs in the reactor would be computed. The power of this channel is 9648 kW at the beginning of cycle (BOC). The peak-power location shifts from the inlet at BOC to the outlet of the channel at the end of cycle (EOC).

The refuelling scheme and axial power profiles from the Reactor Fuelling Simulation Program (RFSP) code were used to create a power profile to feed back into the AECL version of the Winfrith Improved Multigroup Scheme (WIMS-AECL) code to calculate linear element ratings. Twelve power values were used in time, for each of five modeled axial positions, as shown in Figure 9.

More details of the physics aspects of the pressure tube type SCWR preconceptual design can be found in a related paper appearing in this conference [20].

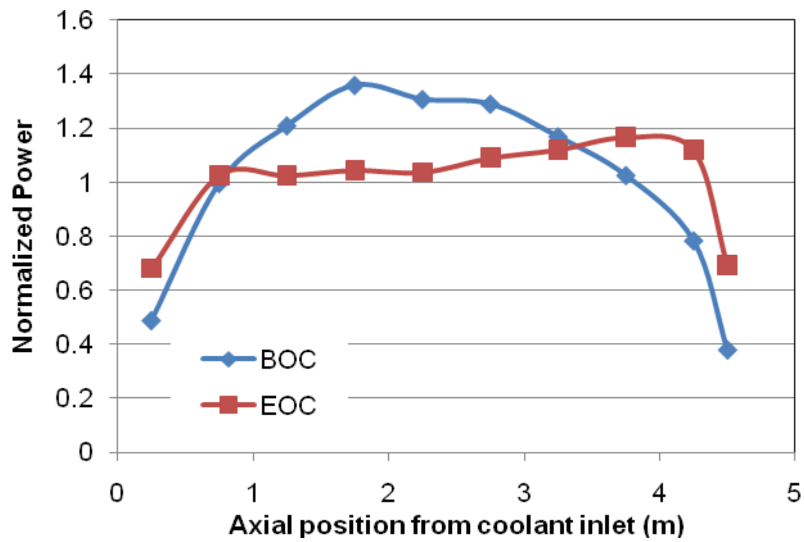

Figure 8 Axial power profile of maximum power channel.

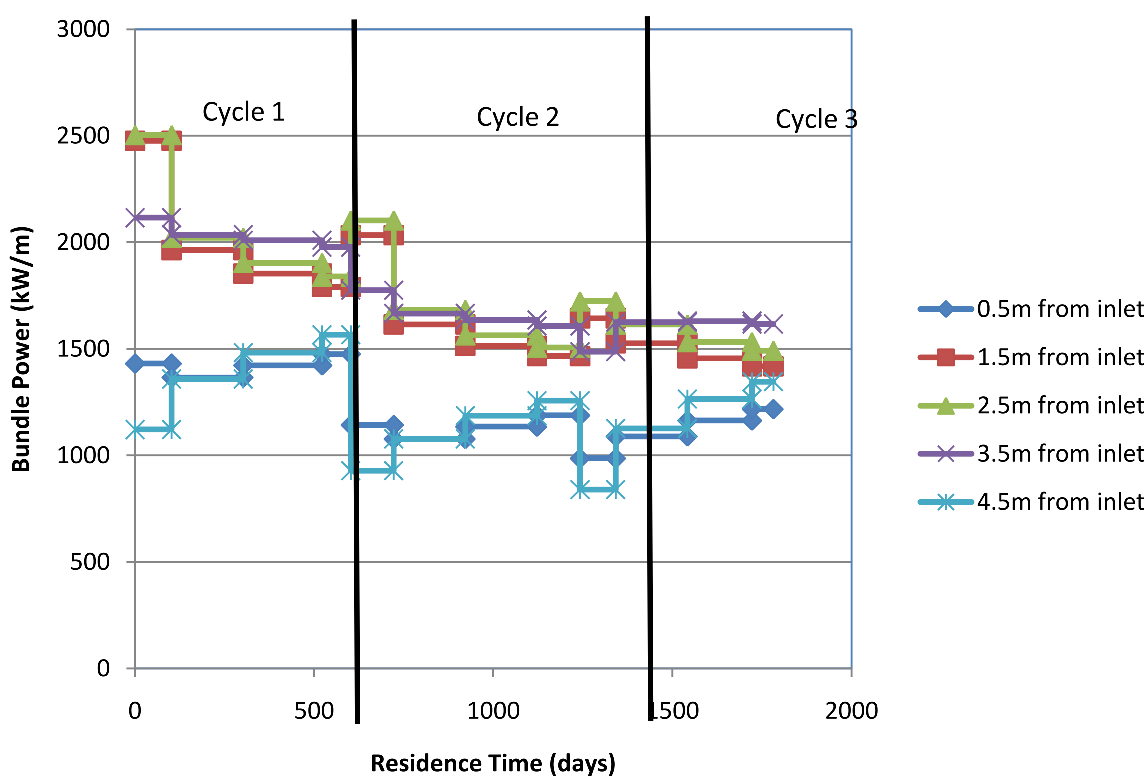

Figure 9 Power profiles applied to each axial position.

5. Conceptual Fuel Design

The limit on the SCWR outlet temperature is effectively set by the mechanical and corrosion properties of the fuel cladding. Properties such as the strength and corrosion resistance decrease with increasing temperature. The mechanical and corrosion properties of currently available zirconium-based fuel cladding alloys, which have been the mainstay of the current fleet of commercial nuclear power plants, are such that they cannot be used in an SCWR. As a result, various iron and nickel-based alloys are being considered for the fuel cladding of the Canadian SCWR, including austenitic stainless steels, ferritic / martensitic steels, and oxide dispersion strengthened (ODS) steels [22]. The cladding material must be able to withstand the peak cladding temperature, which will be some 20% higher than the average cladding temperature. While no alloy has yet been shown to possess all required properties for use at the proposed Canadian SCWR temperatures, there are a number of promising candidates. A number of research and development programs are underway to develop a suitable fuel cladding material for use in the Canadian SCWR. However, the use of these alloys results in some losses in fuel efficiency due to their higher neutron absorption compared to zirconium-based alloys. This is offset by the optimized bundle design where peak fuel cladding temperature is reasonable uniform across the fuel bundle leading to an increase in the allowable outlet temperature and hence efficiency [23], [24].

The Canadian SCWR fuel bundle design has three concentric rings of fuel with 15, 21, and 42 fuel elements (Figure 6). This bundle has a large non-fuel element in the centre. The removal of fuel from the central region of the bundle has the effect of significantly reducing the coolant void reactivity without requiring burnable neutron absorbers. The fuel composition for the bundle is 13% plutonium in thorium. Table 3 lists the dimensions for the current conceptual Canadian SCWR fuel design. Further optimisation of the fuel geometry is in progress to enhance the heat transfer.

Table 3 Geometry Parameters for Conceptual Canadian SCWR Fuel Design.

Parameter	Value
Elements per bundle	78
Elements in rings 1, 2, 3	15, 21, 42
Pitch circle radius, ring 1	3.655 cm
Pitch circle radius, ring 2	5.11 cm
Pitch circle radius, ring 3	6.295 cm
Radius of central pin	2.82 cm
Outer radius of central pin cladding	2.88 cm
Radius of pins in ring 1 and 2	0.62 cm
Outer radius of ring 1 and 2 pin cladding	0.68 cm
Radius of pins in ring 3	0.35 cm
Outer radius of ring 3 pin cladding	0.41 cm

6. Safety Systems

The safety concepts for the Canadian SCWR are generally similar to those developed for modern nuclear reactors, but specific considerations are necessary to cover the transition through the pseudocritical temperature. Passive safety concepts have been incorporated to support the "inherent safety" goals required in next generation nuclear reactors.

- The Canadian SCWR fuel is designed to exhibit a negative coolant void reactivity coefficient throughout the residence time in the core. Therefore, a large power pulse will not be encountered under the postulated large-break LOCA scenario.

- One of the inherent safety characteristics of the CANDU reactor is the separation of the primary coolant from the moderator. This feature provides a large heat sink (moderator) in case of LOCA within the primary HTS.

- One of the possible benefits of using the HEC is that in the event of a LOCA without emergency core cooling, the fuel may not melt because of passive heat rejection through the insulator into the moderator [13]. That is, the heat in the fuel will be radiatively transferred to the liner tube and conducted to the moderator, maintaining the fuel cladding below its melting point. Work is proceeding to optimize and demonstrate HEC performance for normal operating and accident conditions

- To ensure the effectiveness of long-term cooling, a passive moderator cooling system has been introduced to remove decay heat from the fuel in a large-break LOCA event (see Figure 10). This system could potentially meet the moderator heat removal requirements for both normal operating and accident conditions. The effectiveness of the passive moderator cooling system has been verified experimentally in a small-scale test facility [25]. A large-scale test facility is being designed to qualify the system.

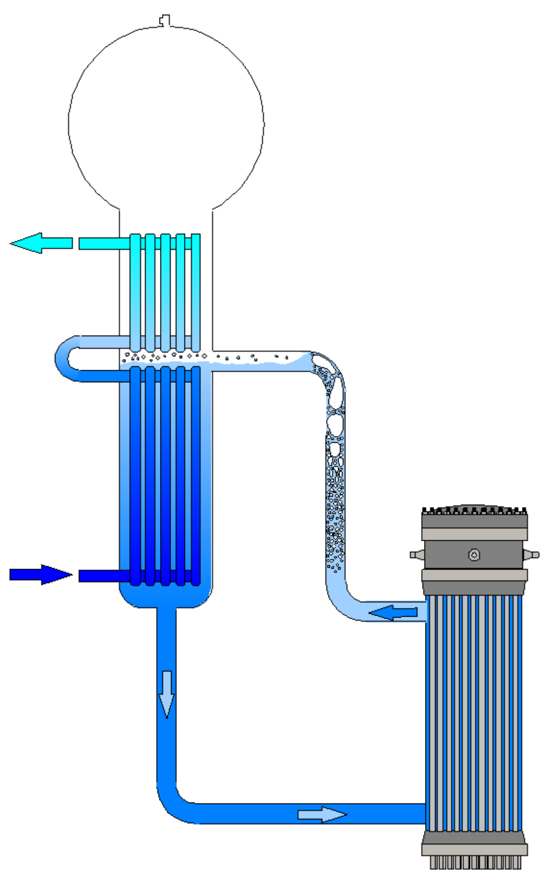

Figure 10 Schematic Diagram of the Passive Moderator Cooling System.

7. Summary

A pre-conceptual supercritical water cooled pressure-tube reactor design is presented. It has the potential to meet all GIF design goals (namely, enhanced safety, improved economic, enhanced sustainability, and improved proliferation resistance). Compared to present pressure-tube and other reactors, this concept offers the following advantages in mechanical design, safety and operations:

A) System Simplification

- Eliminates inlet feeders;
- Eliminates channel closure plugs (two for each fuel channel);
- Eliminates channel closure seal (two for each fuel channel);
- Allows batch fuelling with simultaneous multi-channel fuelling and convenient access through a hollow inlet plenum;
- Easier pressure tube replacement with convenient access to fuel channels by removing the inlet plenum head;
- Direct cycle option (no steam generator).

B) Passive Safety

- Enables a compelling safety case;
- Passive core cooling is possible through natural convection of low pressure moderator;
- Small break LOCA has a small impact due to the common coolant inlet;
- No-core-melt and "walk away" passive safety potential;
- No active pumps or power required for decay heat removal;
- Negative void reactivity coefficient and negative fuel temperature reactivity coefficient.

C) Performance and cost

- Provides up to 40% higher efficiency as compared to current nuclear generating stations;
- Uses known supercritical coal power plant materials and turbine.

Refinement of this pre-conceptual design via, optimization of the fuel, safety and layout is proceeding. Achieving the no-core melt design goal with passive heat removal would improve public confidence enhancing their acceptance of nuclear power.

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