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SYSTEMATIC APPROACH TO THE DEVELOPMENT OF THE CANADIAN SCWR REACTOR FOR DESIGN OPTIMIZATION

J. Li and M. Yetisir
Atomic Energy of Canada Limited, Chalk River, Ontario, Canada

A B S T R A C T

The Canadian Super-Critical Water-cooled Reactor (SCWR) is a Generation IV reactor that is being designed as part of the international initiative to design next generation nuclear reactors known as the Generation-IV International Forum (GIF). It is a heavy-water moderated pressure-tube type reactor that uses supercritical water as coolant (which allows ~40% higher thermodynamic efficiency than current light-water reactors), employs passive safety systems, uses an insulated fuel channel design, and burns thorium as fuel for sustainability and proliferation resistance.

The pressure tube design feature provides more flexibility to optimize the reactor efficiency in addition to enhancing reactor safety in comparison with other reactor designs. The operating pressure (25 MPa) and temperatures (typically 450ºC to 625ºC) of SCW reactors are significantly higher than those in existing light-water reactors, presenting design challenges that require innovative solutions. This paper provides a summary of current status of the mechanical design of the Canadian SCWR, discusses some of the design challenges and proposes path forward for future R&D to deal with these challenges. Also the paper discusses a variety of technologies that may be employed to achieve optimized reactor efficiency and increased plant reliability as well as plant economics.

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 [1], [2]. Some of the advanced features of the proposed Canadian SCWR are as follows [1]:

1. Passive Safety – Passive decay heat removal based on natural circulation and radiation cooling is used to mitigate accident scenarios. The 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 Fuel Cycle – The Canadian SCWR is specifically being designed with the capability to operate with sustainable fuels, namely thorium-uranium-233, and thorium-plutonium reference fuel cycles, while producing spent fuel with reduced content of plutonium and actinides, thereby improving resistance to proliferation and reducing heat loads in spent fuel storage areas.

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

3. Improved Economics – At SCWR operating conditions, thermodynamic cycle efficiency increases significantly. More than 40% increase in cycle efficiency as compared to current nuclear power plants is possible, resulting in reduced generating cost. In addition, the proposed cogeneration presents the potential that the Canadian SCWR may be operated continuously at its full capacity, which would significantly improve its operating economics. During times of low electricity demand, an increased fraction of energy could be used for cogeneration processes, such as hydrogen generation.

2. Current Status of Canadian SCWR design

The Canadian SCWR employs a direct steam cycle in which the supercritical water is fed directly into the high pressure turbine. Operating conditions at the turbine inlet (25 MPa and 625ºC) are selected to match those of current advanced SCW fossil power plants. A direct-steam cycle is used with a theoretical cycle efficiency approaching 50% [1].

Because of Canada's expertise in pressure-tube reactors, i.e. CANDUs®, Canada is working on a pressure-type SCWR while Europe and Japan are working on pressure-vessel type SCWRs. Figure 1 illustrates the current Canadian SCWR core design [2]. The proposed design uses a pressurized inlet plenum attached to a traditional channel-type core, and differs from traditional pressure-tube heavy water reactor (HWR) designs in three major features: (1) it uses an inlet plenum instead of inlet feeders, (2) adopts a vertically oriented reactor core, and (3) refuels off-line. This design eliminates the challenge of the stresses and safety issues of connecting an on-line fuelling machine to a pressure tube at SCW conditions. The water enters the inlet plenum through inlet nozzles (inlet pipes are not shown) and then enters the fuel channels that are connected to the tubesheet at the bottom of the inlet plenum. Inlet conditions are specified to be subcritical at a pressure of 25 MPa and a temperature of 350ºC. As the coolant flows vertically downward through the fuel channels, it gradually becomes supercritical with the energy released by nuclear fission. At the outlet, fuel channels are connected to smaller diameter outlet pipes that transport the supercritical water to common outlet headers at an average 625ºC temperature. This temperature is chosen specifically to match existing SCW turbines in coal power plants.

The reference fuel channel concept is called the high efficiency channel (HEC) [3]. The fuel bundle is contained within a fuel channel that consists of a pressure tube, insulator, and liner tube. In this concept, the pressure tube is in direct contact with the heavy water moderator. The pressure tube is made from a zirconium alloy, Excel® [4], which was developed by AECL in the 1970s and 1980s. The low pressure and temperature of the moderator is achieved by thermally insulating the inside of the pressure tube from the primary coolant. The reference design utilizes a porous yttria-stabilized zirconia (YSZ) insulator separated from the bundle by a thin perforated liner tube.

One of the possible benefits of using the HEC design is that in the event of a LOCA without emergency core cooling, the heat in the fuel is radiatively transferred to the liner tube and conducted to the moderator, maintaining the fuel cladding below its melting point. The heat transferred to the moderator is removed via a flashing-driven passive moderator system [5].

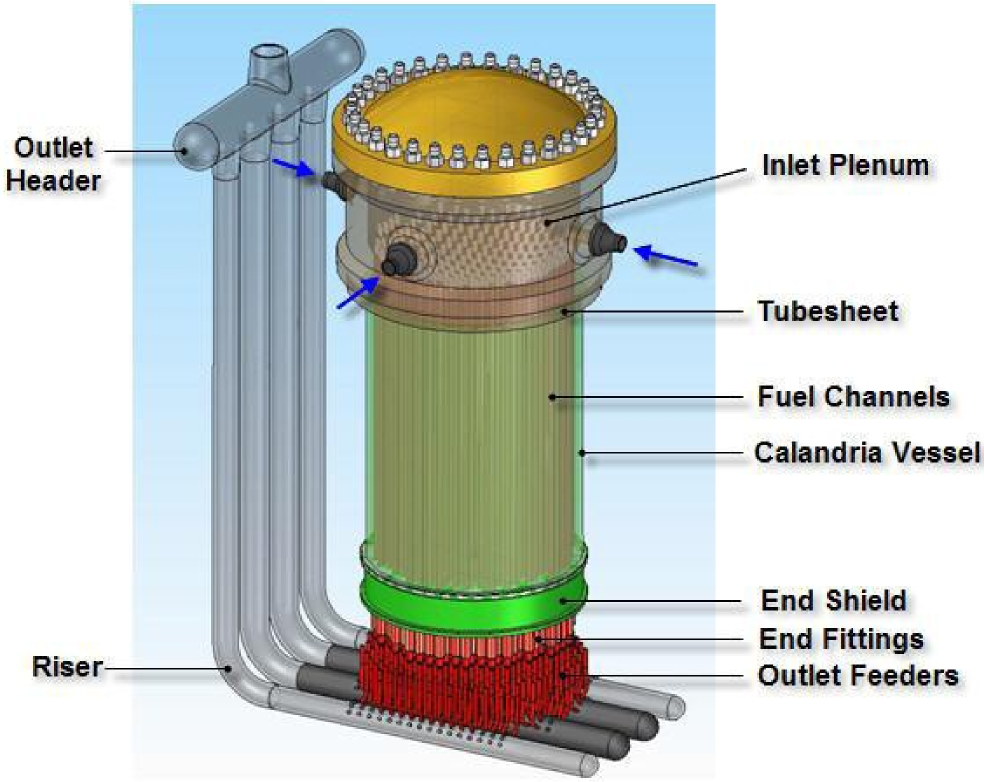

Figure 1 Canadian SCWR

3. Design Challenges

The Canadian SCWR is in the pre-conceptual design stage. There are number of design challenges that require further R&D. Some of these challenges are discussed below:

3.1.1 Fuel Channel Connections

At high pressure and temperatures such as those proposed for the Canadian SCWR, leak-tight operation of the plant may be challenging. Too much leakage at joints would impose significant operational challenges and can affect plant safety by obscuring safety-significant leakages through small cracks. Of particular interest are the connections of fuel channels at the inlet plenum and at the outlet feeders. Because the pressure tube material Excel®, a zirconium alloy, cannot be welded to dissimilar materials without loss of ductility at the weld region, direct welding of the pressure tubes to the inlet plenum and to the outlet piping is not feasible. In traditional CANDUs, pressure tubes are roll expanded to stainless steel end-fittings that are connected to the inlet and outlet pipes. This technology is being explored for use in the Canadian SCWR, but the ability of rolled joints to maintain leak-tight operation at supercritical conditions has not been demonstrated yet. Thermal expansion and the deflection of the inlet plenum combined with the increased thickness of fuel channel and higher operating pressures will likely require different rolling parameters than what is used in current CANDUs and, hence, require R&D to establish the feasibility of rolled joints. In parallel with the rolled-joint technology, other fuel channel connection technologies including bolted and co-extruded joints are being explored. Because the inlet temperature is 350ºC, only 40ºC higher than the current CANDU 6 design, there is a good chance of success for the inlet fuel channel connection. At the outlet, the temperature of the supercritical water is 625ºC, significantly higher than past CANDU experience. At the outlet connection where the fuel channel is connected to a small-diameter outlet pipe, the fuel channel insulator is extended into the pipe by about 1 m so to maintain a low temperature at the joint. This configuration reduces the mismatch of thermal expansion of the fuel channel material and the outlet piping material and, thus, ensures leak-tight operation.

3.1.2 Outlet Pipe Design

At the reactor outlet conditions of 25 MPa pressure and 625ºC temperature, material creep could be a significant degradation mechanism and has to be considered by the design. At 625ºC, the ASME code stress allowables for common materials reduce significantly (for example, ferritic steels have allowable stress of about 50 MPa or less at this temperature), resulting in significantly increased wall thickness. At these conditions, pressure vessel design is very challenging. Hence, smaller diameter pipes are used at the reactor outlet. As shown in Figure 1, in the reference design, small diameter outlet pipes are connected to larger diameter riser pipes. The consequence of a break of a riser has not been evaluated yet, but due to the riser's connection to multiple channels, consequences could be serious. Hence, an alternative design using only small diameter piping is considered. In this design option, there are 336 outlet pipes attached to 336 individual fuel channels. In supercritical coal plants, ferritic steels, such as P91 and P92, are used for such applications, but these materials are not recommended for temperatures higher than 600°C. Austenitic stainless steels have higher allowables and can operate at higher temperatures, but the coefficient of thermal expansion is almost 50% higher than those for ferritic steels, resulting in higher stresses. Nickel based superalloys, which are considered for ultra-supercritical coal plants, can operate at much higher temperatures and can be candidate materials for outlet piping. Because of increased surface area, heat loss through these feeders can be significant. Hence, an insulated cabinet is needed. This concept is shown in Figure 2.

Coating or cladding with insulation material onto the inside surface of outlet piping is considered to improve corrosion resistance and heat loss. This technique will be applied for transporting hot water/steam at >530°C for long distance (in kms) from the reactor to the hydrogen production plant.

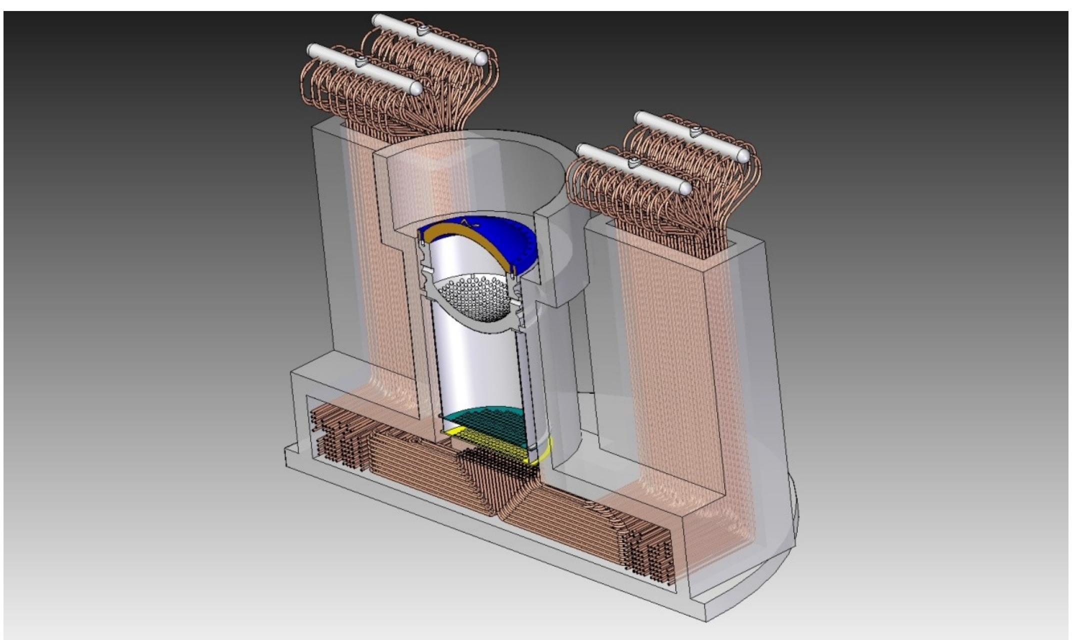

Figure 2 Canadian SCWR with Small-Diameter Outlet Piping

3.1.3 Leak-before-break

To avoid a catastrophic failure of a fuel channel(s) and to support the leak-before-break argument, small leaks from fuel channels must be detected. In traditional CANDUs, this is accomplished by monitoring the moisture level in the annulus gas between the pressure tube and the calandria tube. Since SCWR fuel channels do not have calandria tubes, an alternative concept is needed for the Canadian SCWR. A possible solution for this problem is the use of acoustic detection techniques. A small leak of the supercritical water to the moderator is expected to produce a measurable noise when the superheated water bubbles collapse in the subcooled moderator, producing a distinct noise, which is expected to have a narrow-band and high-frequency spectrum. By properly positioning multiple acoustic detectors and using triangulation, the location of a leak can be determined. Work is underway for the estimation of critical crack length, expected leak rates as a function of crack length and sensitivity of leak detection to support the leak-before-break behaviour of fuel channels. An alternative technique is thermal imaging that detects and tracks a localized temperature increase in the moderator side of the reactor core, which would indicate a leak from a pressure tube when it happens. There are technical challenges to implement any of these techniques with respect to the high radiation field and the reactor core configuration with large number of pressure tubes.

3.1.4 Downward Flow

The direction of the flow is downward in the Canadian SCWR. There are few main reasons for this. The main reason is that a complete break of a feedwater inlet piping does not drain the inlet plenum immediately. Guillotine break of an inlet nozzle is a large LOCA and it is considered to be the worst accident for the Canadian SCWR. Following such a break, water inventory in the inlet plenum provides immediate core cooling while shutdown and other safety systems take effect. The other reasons are the ease of refuelling, the ease of pressure tube replacement and the ease of maintenance and inspection. Also, the heat transfer deterioration that occurs at about the critical temperature does not seem to occur with downward flow, providing higher margins of safety.

The penalty of downward flow is the possible hydraulic instabilities that can occur at some flow conditions. Vertical channel flow instabilities are well-known in two phase flows and they are caused by the significant density difference of the liquid and gas phases [6], which are about two orders of magnitude different for water at atmospheric conditions. Such instabilities are significantly worse in downward flow. In the SCW reactor, because of a large variation of water density across the core, the possibility of density-wave instabilities and coupled thermal-hydraulics and neutronics instabilities exists. Such instabilities are studied [7] for Canadian SCWR operating conditions and fuel channel geometry. Because of the smoother change in density with the supercritical fluid vs. two phase flow, hydraulic instabilities are expected to be more benign at supercritical conditions. However, if instabilities are found to be an issue, design modifications such as orificing at the fuel channel inlet or allowing some coolant to flow axially through a central flow channel in the fuel assembly can mitigate hydraulic instability.

Another issue with the downward flow is the possibility of flow reversal in an accident following loss of feedwater flow and due to the buoyancy forces introduced by steam and water formation in the fuel channel as a result of reactor depressurisation. Consequences of flow reversal and the conditions leading to flow reversal are not well known yet and need to be studied as part of the safety systems design. To deal with the possibility of flow reversal, multiple (and redundant) feedwater pumps and inlet nozzles will be used. An auxiliary feedwater system with lower flow rate will be used. Also, a high-pressure emergency core flooding system may be needed. To maintain "passive" functionality of such a system, it will likely be powered by a high-pressure gas system, either by nitrogen gas or by steam at high pressure.

3.1.5 Calandria Design

The major safety feature of the Canadian SCWR is the passive moderator cooling system. This safety system has to be available under all accident scenarios to remove decay heat and to prevent fuel-melt. In order to maintain the passive moderator cooling functionality, the calandria vessel has to be designed to accommodate a pressure tube burst. A slow pressurization of the calandria vessel can be accommodated through the use of rupture disks connected to the moderator system. They would be positioned higher up in the calandria vessel to ensure that natural circulation of the passive moderator cooling system will be maintained. Following a pressure tube burst, shutdown systems would quickly shut the reactor down and automatically initiate primary-side depressurization. The supercritical fluid escaping from the burst fuel channel would be pressurizing the calandria vessel, but upon contact with the cool moderator, supercritical water would (at least partially) condense and reduce the volume and the pressure of escaping steam. The rate of pressurization of the calandria vessel after a pressure tube break and the time required for the rupture disks to respond to the pressure increase is yet to be determined. It is expected that the rupture disks will respond fast enough while the calandria vessel pressure is increasing and hence will not require a very thick wall.

3.1.6 Passive Moderator Cooling System

One of the inherent safety characteristics of the Canadian SCWR pressure-tube reactor is the separation of the high pressure primary circuit from the lower pressure moderator, which provides a large heat sink in case of severe or extreme accidents. The effectiveness of long-term cooling is ensured by a natural-circulation driven moderator cooling system to remove decay heat from the fuel in a large-break loss-of-coolant event (see Figure 3). This system could passively reject decay heat to the ultimate heat sink without fuel melting. Following an accident, radiation heat transfer from the fuel elements is conducted through the insulator and the pressure tube to the moderator. The ability of the moderator cooling system to remove heat is enhanced by flashing and condensation of the heavy-water moderator in the moderator recirculation loop. For flashing to occur, the temperature of the moderator has to be close to the saturation temperature while the moderator at the outside surface of pressure tube remains at subcooled condition. In present CANDU reactors, the moderator is maintained subcooled at about 20 to 30°C to avoid nucleate boiling while minimizing heat loss from fuel channels to the moderator. At this level of subcooling, the passive moderator cooling system (PMCS) would not flash and would function at a significantly reduced capacity. Hence, a significantly larger and/or taller passive moderator cooling system would be needed to remove moderator heat at normal operating conditions. An alternative to this option is to supplement the passive moderator system with an active moderator cooling system (AMCS). At normal operating conditions both PMCS and AMCS work together to maintain the moderator temperature at 20 to 30°C subcooled. At accident conditions, both systems are used if there is grid power or backup diesel power available. However, if the AMCS becomes disabled because of a loss of grid and backup power, the moderator temperature will increase close to the saturation temperature, enabling PMCS to function at its maximum capacity. At these conditions, the reactor would be in shutdown state and the PMSC alone would be able to remove decay heat to the ultimate heat sink. Another function of the active moderator system is to drive the moderator purification system, which will require a significant pressure head that may exceed the capabilities of a natural-circulation loop.

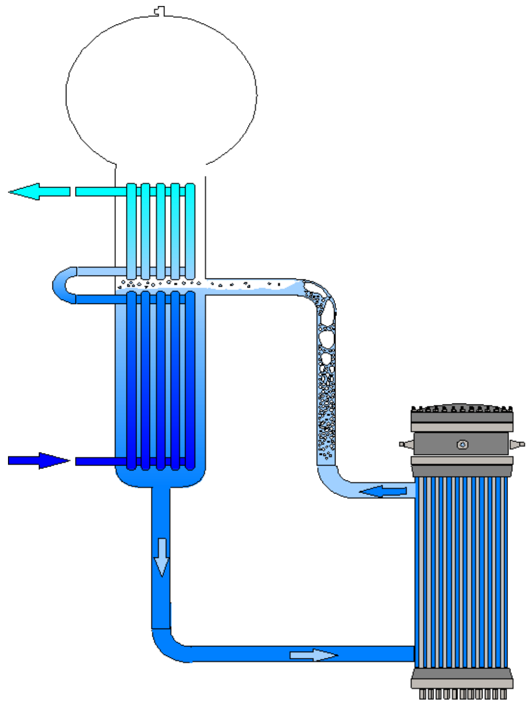

Figure 3 Schematic Diagram of the Passive Moderator Cooling System

4. Design Optimisation and New Technologies

The Gen IV reactor provides opportunities to implement technologies that require long development time and have not been considered by the risk-avert nuclear industry. Some of these technologies have been used in other industries, but have not found their way into nuclear plants. Some of these technologies and their potential uses are listed below.

4.1 Improved Thermal Efficiency

4.1.1 Reheat

The direct steam cycle adopted for the reference Canadian SCWR design includes a moisture separator reheater (MSR) after the high-pressure (HP) turbine to remove moisture from the steam so that the blades of the intermediate-pressure (IP) turbine is not damaged. An increasingly-used feature in supercritical coal plants is to "reheat" the steam by returning the flow from the back end of the high pressure (HP) turbine to the core in the second pass at a lower pressure. The steam is then reheated to the required superheat and fed to the intermediate pressure (IP) section of the turbine. This configuration eliminates the need of the MSR and, also, further increases the thermal efficiency. For a channel reactor, the reactor exit temperature can be established by either changing the channel length, the flow rate, or the number of passes through the core, or some combination. Reheat steam cycle version of the design uses channels to superheat steam which are placed at the periphery of the reactor core and have about 1.5 times lower heat flux compared to the average heat flux.

A maximum calculated efficiency of 50% is achievable using the direct cycle with the reheat option. This represents a major improvement of 40% in efficiency over current LWR designs (at about 35% efficiency), satisfying the economic design goal of the SCWR. Hence there is an equivalent increase in the energy extracted from the same amount of fissile material, a result consistent with analyses of the fuel irradiation and reactor core physics for SCWR using special high-efficiency pressure tubes with an optimized lattice. The reheat capability is being explored as a design option for the final Canadian

SCWR concept. The economic impact of the reheat capability is yet to be evaluated as the addition of reheat channels increases the reactor core size which could otherwise be used for regular fuel channels.

4.1.2 Low-Grade Heat Utilisation

As roughly half of the heat generated by nuclear fission in the reactor core is not converted to electricity and simply discarded to the environment as low-grade heat (LGH). This is due to lack of efficient conversion methods of LGH to electricity. LGH is a thermal source that has too low a temperature to boil water, typically at temperatures below 80ºC. This constitutes more than 50% of power produced by the Canadian SCWR and hence has significant potential for various uses. In pressure-tube reactors with heavy water moderator, in addition to the cooling water systems at the balance of plant, the heat transferred to the moderator (~5% of the thermal power) is also lost. Hence, there is an additional incentive to utilise LGS.

Electricity production from LGH is possible but the conversion efficiencies are so small (less than few percent) that it is usually not considered economical. However, the utilisation efficiency of the LGH has been increasing over the years with the development of new materials and new technologies. Hence, this is a promising area for the design optimisation of the Canadian SCWR.

Two methods for converting LGH into electricity are described below.

1. Direct Heat-to-Electricity Conversion: Direct conversion technologies convert the heat energy directly into electricity without moving fluids and components such as pumps, turbines and condensers. Various technologies using the principles of thermionics, thermoelectricity and thermovoltaics are used for direct conversion of heat to electricity. Such devices are typically solid-state converters with low conversion efficiencies, but they do not emit any noise and can provide autonomous and reliable electric power supply wherever there is a heat source. A possible application of these technologies in nuclear plants is to power critical instruments during an extended station blackout. Typical thermoelectric efficiencies are just a few percent at LGH temperatures, but much higher conversion efficiencies have been reported recently in thermionic devices in controlled conditions [9].
2. Thermodynamic heat engines: At lower LGH temperatures it is possible to generate electricity using the same principles as the conventional power plants using a heat engine, but using a fluid that boils below 100ºC and has high molecular mass that results in higher fluid density and, hence, higher efficiencies. Organic fluid such as n-pentane or toluene gives higher cycle efficiencies and is used in organic Rankine cycle heat engines known as ORCs. ORC cycle efficiencies as high as 4% is reported for LGH at 80ºC with potential to increase the efficiency to 9% [10].

LGH can be used effectively for non-electrical applications such as district heating and desalination. These options should be considered for the effective use of the energy generated at the NPP.

4.1.3 Flow Distribution in Fuel Channels

Reactor power generation is not uniform across the core. The central channels produce about 20% more power and the peripheral channels produce about 20% less power than the average channel, resulting in a range of channel powers from about 6.3 MWt to 9.1 MWt per channel. If the same mass flow rate is used in all channels, the central channels would be limiting in terms of the peak cladding temperatures. By increasing the flow rate at central channels and decreasing the flow rate at peripheral channels, maximum cladding temperature at the central channels can be reduced. This can be accomplished by using orifices at the fuel channel inlet or outlet with the goal of achieving the same outlet temperature of 625ºC in all channels. Because the peak cladding temperature is one of the design limitations, this arrangement provides highest safety margin and makes it possible to operate the reactor with an average exit temperature of 625ºC. Orifice design requires a good understanding of the pressure losses in individual fuel channels and the fluid mechanics in the inlet plenum.

4.1.4 Moderator Heat Utilisation

Up to 5% of the thermal power generated in the reactor core is transferred to the heavy water moderator through a combination of conduction-convection heat transfer and gamma ray absorption by the heavy water moderator. In CANDU reactors, this heat is simply rejected to the environment. Utilisation of this low-grade heat (see Section 4.1.2) can increase the thermal efficiency of the plant. Another application for this heat can be the direct conversion of heat to electricity using solid-state thermoelectric devices to power critical instruments so that the condition of the core can be evaluated in case of an extended station blackout with the loss of diesel and battery power, as happened in Fukushima.

4.1.5 Improved Heat Exchangers – Micro-Channel Heat Exchangers

Micro-channel heat exchangers use diffusion-bonded plates with very small flow channels, typically smaller than 1 mm in diameter. At this scale, heat transfer from primary liquid to the secondary fluid can be significantly higher than those in traditional shell-and-tube heat exchangers resulting in significantly smaller (volumetrically, as much as an order of magnitude) compact heat exchangers without a significant pressure drop penalty. Because of very high heat exchange efficiencies in micro-channel HXs, the potential for improved overall thermal efficiency exists. Also, because of much smaller footprints, compact micro-channel heat exchangers allow smaller containment and turbine buildings that can reduce capital cost significantly.

4.2 Co-Generation

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 optional supplementary products. Another potential application of the available co-generated process heat is the extraction of bitumenfrom oil sands, which is presently achieved using co-generation with natural gas turbines and process heat.

4.2.1 Desalination

As discussed in Section 4.1.2, a significant amount of energy produced in the core is rejected to the environment. This low-grade heat can be used for desalination at locations where fresh water is not readily available. The worldwide demand for desalination is expected to double approximately every ten years in the foreseeable future [13]. Hence, the prospect of using nuclear energy for seawater desalination on a large scale is an attractive option in regions where desalinated water is needed. Two desalination processes are most commonly used to generate fresh water from sea water. These are (1) multi-stage flash distillation (MSF) process and (2) reverse osmosis (RO) process. The MSF process is heat energy intensive but does not require as much power as the RO process.

4.2.2 Hydrogen Production

AECL has been working on nuclear hydrogen production for the last eight years in collaboration with Canadian Universities and the U.S. National Laboratories. There are two types of production processes that have been investigated at AECL, which are Cu-Cl process and S-I process. The Cu-Cl process requires a heat source at 530°C from the reactor. The S-I process requires two heat sources at about >400°C and >850°C. Most of AECL effort in nuclear H_2 production has gone to the Cu-Cl process. The Canadian design would need to be optimized so that heat at 530°C is most efficiently supplied for H_2 production to achieve the optimal overall efficiency.

4.3 Improved Safety via Passive Components

Advanced nuclear power plants increasingly employ passive safety systems and passive components that activate without a need for external power. In addition to existing passive components, such as steam-powered pumps, gravity driven control rods and gravity-driven safety systems, many innovative passive components have been developed for advanced reactors. For example Generation III KERENA reactor employs an outflow reducer to reduce the discharge mass flow from the RPV in the event of a break in the emergency condenser return line [11]. A similar concept is used in the HPLWR design at the feedwater inlet line [12].

4.3.1 Outflow Reducer

In the Canadian SCWR design, it is expected that the break of an inlet pipe is one of the most critical accident scenarios. The rate of blowdown transient following an inlet line break has significant effect on the severity of such an accident while the reactor automatically shuts down and safety systems engage. The rate of the blowdown transient may be especially critical for those reactors relying on gravity driven passive cooling systems and for SCW reactors that have low water inventory. To slow the rate of blowdown some designs use outflow reducers that have very low hydraulic resistance in the "in" direction while providing much higher hydraulic resistance in the "out" direction. These components can significantly prolong the blowdown (and hence, maintain core cooling capability) while reactor decay power reduces to smaller and manageable levels. The uses of such fluidic devices are considered for the Canadian SCWR.

4.3.2 Heat Pipe Technology

A heat pipe is a heat-transfer device that uses the mechanisms of evaporation, condensation and mass transfer through capillary action and gravity to passively transfer heat from one location to another. Because the heat transfer rate for condensation and evaporation is very high, these devices can transfer up to two orders of magnitude more heat than could be conducted through an equivalent cross-section of copper. Also, because they contain no mechanical moving parts, heat pipes require no routine maintenance and operate in a highly reliable manner.

Heat pipes have been used in space applications and have been shown to operate reliably for many years. An innovative application in nuclear technology was proposed for a small reactor concept developed by AECL to transfer core power to a secondary fluid through heat pipes [13]. Possible applications of the heat pipe technology in advanced nuclear power plants are to passively (and reliably) cool fuel bays and remove heat from the ultimate heat sink following a station blackout to maintain long-term cooling capability. These possibilities are being explored.

5. Summary and Conclusions

The proposed high operating pressure (25 MPa) and temperature (625ºC) of the Canadian SCWR present design challenges that require innovative solutions. In this paper, some of these challenges and possible solutions are presented. Innovative technologies are discussed that can be incorporated in the design to improve thermal efficiency and safety as well as reliability and economics.

6. References

[1] L.K.H. Leung, M. Yetisir, W. Diamond, D. Martin, J. Pencer, B. Hyland, H. Hamilton, D. Guzonas and R. Duffey, 2011 October, “A Next Generation Heavy Water Nuclear Reactor with Supercritical Water as Coolant”, The International Conference on Future of Heavy Water Reactors (HWR-Future), Ottawa, Ontario, Canada.

[2] M. Yetisir, W.T. Diamond, L.K.H. Leung, D. Martin, and R. Duffey, 2011 April, “Conceptual Mechanical Design for a Pressure-Tube Type Supercritical Water-Cooled Reactor”, Proceedings of ISSCWR-5 Conference, Vancouver, BC.

[3] C.K. Chow and H. F. Khartabil, 2007, “Conceptual Fuel Channel Designs for CANDU-SCWR”, Nuclear Engineering and Technology, 40(2): 139-146.

[4] E.F. Ibrahim and B.A. Cheadle, 1985, “Development of Zirconium Alloy for Pressure Tubes in CANDU Reactors”, Can. Met. Quart., Vol.24, p.273.

[5] H. Khartabil, 2008, “Review and status of the GEN-IV CANDU-SCWR passive moderator core cooling system”, Proceedings of the 16th International Conference on Nuclear Engineering (ICONE-16), Orlando, Florida, USA, May 11-15.

[6] I. Babelli and M. Ishii, 2001, “Flow Excursion Instability in Downward Flow System”, Nuclear Engineering and Design, Vol. 206, Pages 91–104.

[7] S. Yeylaghi, V. Chatoorgoon and L. Leung, 2011, “Assessment of Non-Dimensional Parameters for Static Instability in Supercritical Down-Flow Channels”, The 5th International Symposium on Supercritical-Water-Cooled Reactors, ISSCWR-5, Vancouver, British Colombia, Canada.

[8] Duffey, R.B., Pioro, I., Zhou, X., Zirn, U., Kuran, S., Khartabil, H., and Naidin, M., “Supercritical Water-Cooled Nuclear Reactors (SCWRs): Current and Future Concepts – Steam Cycle Options”, Proc. 16th International Conference on Nuclear Engineering (ICONE16), 2008 May 11-15, Orlando, Florida, USA.

[9] P. L. Hagelstein and Dennis Wu, 2007, “Thermal to Electric Conversion with a Novel Quantum-Coupled Converter” (MIT Research Laboratory of Electronics, Progress Report).

[10] M. M. Tahir, N. Yamada, and T. Hoshino, 2010, “Efficiency of Compact Organic Rankine Cycle System with Rotary-Vane-Type Expander for Low-Temperature Waste Heat Recovery”, International Journal of Civil and Environmental Engineering 2:1 2010.

[11] “The 1250 MWe Boiling Water Reactor”, Design Description of KERENA Reactor, AREVA, 2010 September.

[12] K. Fischera , T. Schulenbergb, E. Laurienc, 2009, “Design of a supercritical water-cooled reactor with a three-pass core arrangement”, Nuclear Engineering and Design, Vol. 239, Issue 4, pp 800–812.

[13] IAEA-TECDOC-1326, 2002, “Status of Design Concepts of Nuclear Desalination Plants.”

[14] K.S. Kozier, 1991, “The Nuclear Battery: A Very Small Reactor Power Supply for Remote Locations”, Energy, Vol. 16, No 1-2, pp. 583-591.