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Review of R&D for supercritical water cooled reactors

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ABSTRACT

The Supercritical Water-Cooled Reactor (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point (374 °C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure vessel concepts proposed first by Japan and more recently by a Euratom partnership, and pressure tube concepts proposed by Canada, generically called the Canadian SCWR. Other than the specifics of the core design, these concepts have many similar features, like outlet pressure and temperatures, steam cycle options, materials, or heat transfer characteristics. Therefore, the R&D needs for each reactor type are common, which enables collaborative research to be pursued. The paper provides an overview on research and development performed so far on the SCWR within the Generation IV International Forum.

1. Introduction

Looking at the trend of coal fired power plants in the last 40 years, we observe a remarkable increase of net efficiency from around 37% in the 1970s to more than 46% today. The last 20 years since 1990, in particular, were characterized by an increase of live steam temperature beyond 550 °C, when boiler steels became available which allowed exceeding the former material limits. Along with the temperature increase, the live steam pressure went up to maximize the turbine power, finally exceeding the critical pressure of water. The next generation of coal fired power plant will even reach a net efficiency of ~50%, when live steam temperatures of 700 °C or more can be realized. In comparison with such development, the net efficiency of latest pressurized water reactors (PWR) of around 36% is still close to the efficiency of ~34% of the first generation of light water reactors.

Following this trend, Super-Critical Water-cooled Reactors (SCWRs) are a class of high temperature, high pressure, water-cooled reactors that operate above the thermodynamic critical point of water (374 °C, 22.1 MPa). The GIF Technology Roadmap (GIF, 2002) has identified several of the key technical advantages of the SCWR compared to conventional water technologies that make it attractive for consideration as a Generation IV Nuclear Reactor System. The main thrusts of these advantages translate into improved economics because of the increased thermodynamic efficiency and plant simplification opportunities afforded by the high-temperature, single-phase steam.

Other key advantages of the SCWR include:

- SCW fossil-fired plants (SCW-FFPs) are well known in the electricity-production industry and, in many cases, vendors of nuclear products are also manufacturers of components for SCW-FFPs.
- No turbine development is required for outlet temperatures <625 °C; SCW-FFPs are already operating at these conditions.
- Advanced fuel cycles can be considered, but development of a new fuel type is not essential before a prototype reactor could be built.
- The SCWR is an evolution from existing water reactors (i.e., pressurized water reactors (PWRs), boiling-water reactors (BWRs) and pressurized heavy-water reactors (PHWRs)); thus, today's nuclear expertise is relevant and can be leveraged in the development of the concept.
- Many existing utilities around the world will be comfortable with SCWR technology since they currently operate both nuclear power plants and SCW-FFPs.
- Projections of the cost of the SCWR will be more accurate than those of other systems under development, since cost models will be based in a large part on proven systems.

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2. Past experience

Supercritical water cooled reactors were studied already in the 1950s and 1960s as summarized by Oka (2000). In particular, we like to mention the following early studies:

- A water moderated, supercritical steam cooled reactor was designed by Westinghouse in 1957, in which 7 fuel rods in cylindrical, double walled cans formed the fuel assemblies to insulate the superheated steam from the liquid moderator water at 260 °C. An indirect steam cycle was favoured for this concept to avoid activity in the turbines.
- A heavy water moderated reactor, cooled with light water, was designed by General Electric in 1959 for a thermal power of 300 MW with a once through steam cycle. The coolant passed the core four times, reaching an outlet temperature of 621 °C.
- A graphite moderated and light water cooled pressure tube reactor was designed by Westinghouse in 1962, called the Supercritical Once Through Tube Reactor (SCOTT-R) for an electric power of 1000 MW with a thermal efficiency of 43.5%. The low pressure tank containing the graphite moderator was cooled with Helium.
- A pressurized water reactor with a closed loop primary system at supercritical pressures had already been proposed in 1966.

A supercritical water cooled reactor, however, has never been built in the past. Instead, a boiling water reactor with a nuclear superheater was built in Grosswelzheim, Germany, which could be considered as an early, evolutionary step from boiling water reactors towards an SCWR. The HDR (Heissdampfreaktor) was intended to reach 500 °C core outlet temperature in its final stage, and the prototype built from 1965 to 1969 with 100 MW thermal power was designed for a reduced temperature of 457 °C of superheated steam at 9 MPa reactor inlet pressure as an introductory step. Schulenberg and Starflinger (2012) summarized the key design features.

The construction of the power plant started in Jan. 1965. First criticality was reached on Oct. 14, 1969, and the HDR power plant was connected to the electricity grid on the same day. Commercial operation started on Aug. 2, 1970, reaching up to 23 MW_e, but the core was damaged soon. It has been reported that the tubes of the superheated steam collapsed, but details have not been published. The reactor tests were finished and the reactor was shut down on April 20, 1971, 18 months later, having produced 6200 MWh electric power in total. It is, therefore, not a story of success but still a milestone in the development of light water reactors with increased temperatures.

3. Applications for SCWR technologies

The basic idea of using a supercritical steam cycle for nuclear power production can be applied to a number of different systems, as will be discussed with the following examples which had been worked out by the Generation IV International Forum within the last 10–15 years.

3.1. Pressure vessel type reactors with a thermal neutron spectrum

In Europe, a consortium of 12 organizations from 8 European countries started in 2006 to address this challenge by working out a design concept of such a reactor, which is referred to as the High Performance Light Water Reactor (HPLWR), with a core exit temperature of at least 500 °C at a supercritical system pressure of around 25 MPa. The design objectives were a core with a thermal neutron spectrum, a net electric power of 1000 MW and a net plant efficiency of around 44% for base load electric power production. Meanwhile, the conceptual design has been completed. Schulenberg and Starflinger (2012) reported about design details and supporting analyses.

The core of the HPLWR has to solve a lot more design challenges than simply an increase of the core exit temperature by around 200 °C. Assuming a typical feedwater temperature of supercritical fossil fired power plants of 280 °C, the enthalpy rise in the core would exceed the one of conventional light water reactors by almost a factor of ten. A conventional LWR core design with a single stage coolant heat up from bottom to top would result in peak cladding temperatures beyond any reasonable cladding material limits, if all power and mass flow non-uniformities, uncertainties and tolerances as well as allowances for operation are taken into account. Ideas to solve this issue can be found at coal fired boilers. There, the coolant is typically heated up in three steps, namely the evaporator (which means the transition from liquid like to steam like conditions at supercritical pressures) and a first and second superheater with higher temperatures but lower powers when approaching the envisaged boiler outlet temperature. Intensive coolant mixing between each step eliminates hot streaks of the preceding step before entering the next one. Schulenberg and Starflinger (2012) proposed a thermal core concept in which the evaporator assemblies are placed in the centre of the core, followed by first superheater assemblies with downward flow surrounding them, and second superheater assemblies with upward flow at the core periphery where the fissile power is low anyway because of neutron leakage.

Like in pressurized water reactors, the reactor internals include the core barrel with its core support plate and the lower mixing plenum, the steel reflector, the steam plenum with adjustable outlet pipes and the control rod guide tubes. The core barrel is composed of a cylindrical part with flange and the lower core support plate with orifices as shown in Fig. 1. The circular lower mixing plenum, which is welded to the bottom of the lower core plate, homogenizes the water flow from the downcomer before it enters through the plate into the lower part of the evaporator of the core. An annular mixing chamber underneath the core support

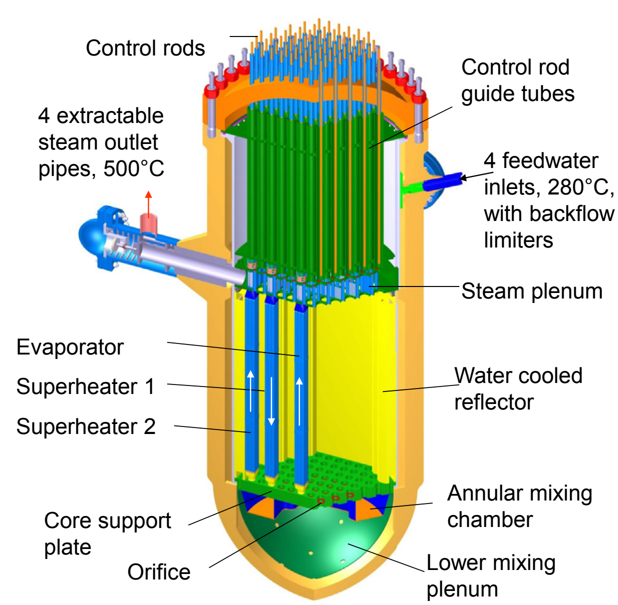

Fig. 1. HPLWR reactor pressure vessel and core structures (Schulenberg and Starflinger, 2012).

plate, inside the lower mixing plenum, mixes the coolant from superheater 1 before it enters superheater 2.

The superheated steam is collected and mixed above all fuel assembly clusters in the steam plenum. This is a leak tight box, which is resting on support brackets of the RPV. An inner part, combining the evaporator outlet with superheater 1 inlet, is separated from an outer part at superheater 2 outlet. Four extractable steam pipes are positioned at the height of the steam plenum to guide the superheated steam through the outlet flanges of the RPV to the steam lines. The steam plenum can be moved in and out of the core barrel using guide rails in its upper part.

To replace spent fuel assemblies, the steam plenum is lifted out of the core barrel using four mounts welded to its top plate. For that purpose, the four outlet pipes are pulled out radially, such that the nozzles of the outlet pipes do not obstruct the lift path any more, while the steam plenum still rests on the protruding support brackets.

The HPLWR core design concept assumes that 50% of the coolant supplied through 4 inlet flanges to the reactor pressure vessel (RPV) is taken first as moderator water to run upwards to the closure head, then downwards through control rod guide tubes and through the central water boxes inside the housed assemblies, to be released through the foot pieces of the assembly clusters to the gap volume between the assembly boxes. From there, it rises upwards to serve again as moderator water outside the assembly boxes. It is collected at the top of the core to cool the radial core reflector with a downward flow, before it is mixed with the remaining 50% of the coolant in the lower mixing plenum underneath the core. The following three heat-up steps comprise an evaporator region formed by 52 assembly clusters in the core centre, where the coolant changes its density from liquid like to steam like conditions, followed by an upper mixing chamber above the core. Another 52 assembly clusters with downward flow surround the evaporator region and serve as the first superheater. After a second mixing in an annular mixing chamber underneath the core, the coolant is finally heated up to 500 °C in a second superheater region formed by 52 assembly clusters at the core periphery. The core arrangement is shown in Fig. 2.

Beyond 390 °C, the coolant density is less than 200 kg/m^3, hardly enough to produce a thermal neutron spectrum. Therefore, colder feedwater is foreseen as moderator water to run inside moderator boxes in the fuel assemblies and in gaps between assembly boxes. With an estimated pressure drop of up to 1 MPa from reactor inlet to its outlet, and with the aim to minimize the mass of structural material in the core to limit the neutron absorption, the fuel assemblies are small, with 40 fuel pins each and a single moderator box in their centre, to enable a small wall thickness of moderator and assembly boxes. To ease handling during maintenance, 9 assemblies are grouped to a cluster each with common head and foot piece. The clusters can be exchanged between evaporator and superheater positions. The fuel rods have an outer diameter of 8 mm with fuel pellets over an active core height of 4.2 m. Wire wraps are proposed as spacers to improve coolant mixing in both flow directions. The clusters can be disassembled at their foot piece to exchange single fuel rods for repair. Control rods with B_4C shall be inserted from the top of the core. They run inside 5 of the 9 moderator boxes of each cluster.

For illustration, Fig. 3 shows cut out view of a single fuel assembly. The assembly box and the water box are made of a stainless steel sandwich construction with an internal honeycomb structure filled with Zirconia to improve the thermal insulation and to reach the envisaged stiffness of less than 0.5 mm deflection towards the fuel rods under an outside pressure load of 500 kPa. Details of the box design are shown in Fig. 3. A venting hole per honeycomb, open to the colder side, is reducing the pressure load acting on the honeycomb structure. The corner pieces are made of solid stainless steel structures to reduce peak stresses there.

The safety system of the HPLWR is very similar to existing boiling water reactors. In principle, active (e.g. low pressure coolant injection system) and passive components (e.g. containment condenser) have been used as part of the overall strategy dealing with accidents and transients. The analyses showed that the behaviour of the HPLWR reacts very benignly in case of reactivity induced accidents. For transients, the investigated cases never lead to an unfavourable behaviour of the reactor core. In case of accidents, like loss of coolant accidents, the reactor behaves like a conventional boiling water reactor, which means that an emergency core cooling system must be foreseen in the safety concept. However, a significant difference in the safety approach compared to existing light water reactors must be noted: Due to the missing recirculation inside the reactor pressure vessel, the coolant flow must always be maintained. Simply maintaining the water inventory in the vessel does not lead to successful core cooling. Therefore, automatic depressurization of the pressure vessel has been foreseen as a passive, fast responding system to remove heat in the short term in case of several accident scenarios.

Aiming at a net electric power of around 1000 MW and a net efficiency of almost 44%, the target thermal power of the reactor

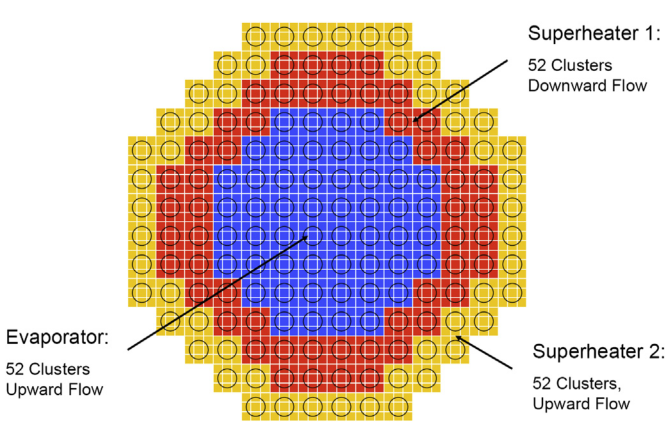

Fig. 2. Arrangement of evaporator and assembly clusters in the HPLWR core (Schulenberg and Starflinger, 2012).

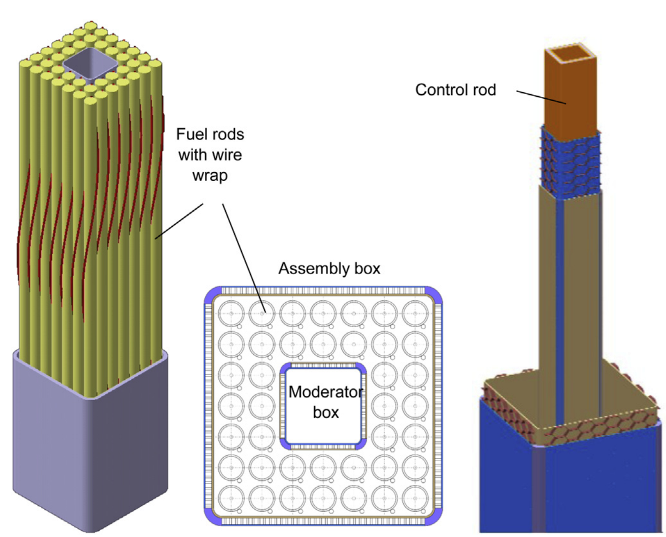

Fig. 3. HPLWR assembly design with wire wrapped fuel rods (left) and honeycomb structures of the assembly and moderator box (right). A square control rod is inserted from top (Schulenberg and Starflinger, 2012).

core needs to be 2300 MW. Early cycle studies indicated an optimum thermal efficiency at a feedwater temperature of 280 °C. The target core outlet temperature was chosen as 500 °C which is still rather low for a once through steam cycle with single reheat, compared with latest fossil fired power plants, but appears to be challenging enough with regard to available fuel cladding materials. Their peak temperature limit was targeted at 630 °C which is not only a challenge for oxidation and corrosion protection, but also for their creep strength and resistance to stress corrosion cracking. The fuel centreline temperature is a function of the linear power of the fuel rod. The latter one has been limited to 39 kW/m under nominal operating conditions. To be competitive with respect to latest pressurized water reactors, the target burn up should be at least 60 GWd/t_{HM}. Like with boiling water reactors, boric acid cannot be used to compensate the excess reactivity at the beginning of a burn-up cycle, so that burnable absorbers like Gd must be used instead. The target power and temperatures result in a coolant mass flow rate of 1179 kg/s. A constant feedwater pressure of 25 MPa has been foreseen for all load conditions keeping some margin from the critical pressure of 22.1 MPa.

3.2. Pressure vessel type reactors with a fast neutron spectrum

In Japan, the Waseda University is developing the Super LWR with thermal neutron spectrum and the Super FR concept with a fast neutron spectrum, both in collaboration with the University of Tokyo. Research and development for technologies of SCWR, like thermal-hydraulics, materials and water chemistry studies, are applicable to both concepts. The objective of the Super LWR design study is to develop power reactor concepts meeting the challenges of market economy by reducing capital cost based on the experience of LWR and of supercritical coal fired power plant technologies. The objective of Super FR design study, on the other hand, is to develop a fast reactor concept of lower capital cost than thermal reactors like current LWR or the Super LWR.

The advanced features of Super LWR and Super FR are addressing the GIF design criteria:

3.2.1. Economy

Super LWR and Super FR are developed based on the experience of mature LWR and supercritical fossil-fired power plant technologies. Capital cost reduction of nuclear power plants is the most important goal for competing with the fossil-fired power plants such as combined cycle gas turbine plants in the electricity market. The capital cost of a Super LWR will be lower by 20–30% compared with an LWR due to the elimination of steam generators of a PWR or recirculation and steam-water separation systems of a BWR.

The Super FR has the same plant system as a Super LWR. It is expected to be in lower capital cost than a Super LWR and a LWR due to the high power density of a fast reactor, whereas a thermal reactor also needs a moderator.

Starting from a two-pass concept, a single coolant path core concept is being developed today for both Super LWR and Super FR. The internal structure of the reactor pressure vessel is as simple as in a PWR.

3.2.2. Sustainability and proliferation resistance

A breeder version of the Super FR shows the high breeding capability meeting the growth of energy demand of advanced countries by adopting newly devised closely packed fuel assemblies. The sustainability of the Super LWR and Super FR is on the line of the MOX fuel utilization in LWRs without abandoning the experience of light water coolant technology.

The transmutation of long lived fission products is possible with a Super FR by utilizing thermal neutron spectrum regions of the zirconium, hydrides layer of the Super FR that is introduced for negative coolant void reactivity.

An ultra-long operating cycle without refuelling for proliferation resistance is possible for the Super FR. The coupled fast and thermal neutron core of the Super FR adopting zirconium hydride layers shows unique long burn-up characteristics because of the increase in neutronic coupling between fast spectrum seed regions which are separated by blankets with zirconium hydrides.

3.2.3. Safety

The safety concept of the Super LWR and the Super FR is based on the experience of LWR. Safety characteristics of the Super LWR and Super FR are well understood by analyses with computer codes. The fuel, core and safety system and method for the analysis are similar to those of LWR. In addition to active systems, passive systems such as isolation condensers (IC) and passive containment cooling systems (PCCS) are easily applied to the Super LWR and the Super FR. Severe accident measures of LWRs such as core catcher and in-vessel retention are easily adapted to the Super LWR and the Super FR.

The Super LWR is a supercritical pressure light water cooled and moderated reactor with a reactor pressure vessel (RPV). It was developed by the research group of University of Tokyo since 1989 and is continued now at Waseda University. The design and analysis of the Super LWR is summarized in the monograph, "Super Light Water Reactors and Super Fast Reactors" by Oka et al. (2010). It includes an overview of the design and analysis methods of core design, safety system, plant dynamics and control, plant startup and stability, fast reactor design and research and development, referring to results up to 2009.

A two pass coolant flow scheme (two pass core) was adopted in the past. The upper core structure, however, turned out to be complex due to the downward coolant flow path from the upper dome to the water rods, and the refuelling scheme was complicated. Therefore, a new single pass coolant flow scheme is now being developed at Waseda University to simplify the design (Wu et al., 2013).

The fast core option, the Super FR, is a light water cooled, fast reactor with a reactor pressure vessel and operating at supercritical pressure. Reducing the capital cost of fast reactors to even smaller costs than LWRs is important target for closing the nuclear fuel cycle. Fast reactors are reducing the amount of spent LWR fuels and radioactive waste and recycle plutonium. Reducing the capital cost of fast reactors, however, is still a challenge for commercializing the nuclear fuel cycle.

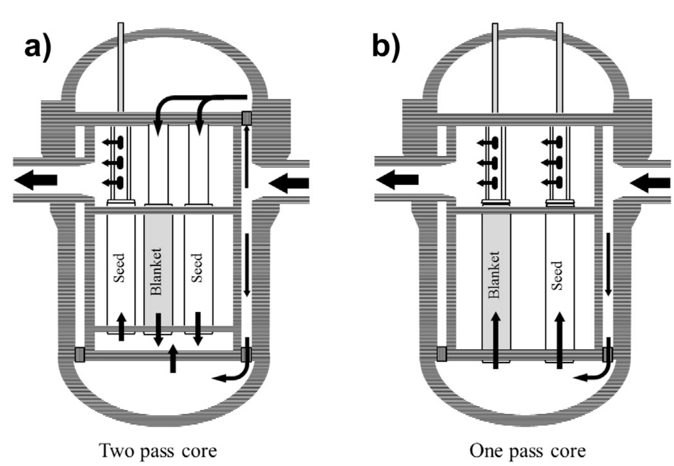

Fig. 4. Alternative flow patterns for Super FR, a) two pass core, b) one pass core (Liu et al., 2013).

The Super FR adopted the two path coolant flow scheme is shown in Fig. 4(a). The single path core is developed on the latest design as seen in Fig. 4(b) simplifying refuelling and in-core structure and removing seals between hot and cold coolant (Wu and Oka, 2014).

The power change of the blanket fuel assemblies with burn up is mitigated by adopting a MOX fuel region in the bottom of the blanket fuel assembly, not to decrease the average outlet temperature. The cross section of fuel assemblies are depicted in Fig. 5. The core loading pattern of the fuel assemblies is shown in Fig. 6. The heterogeneous arrangement of seed and blanket assemblies is proven to be effective for better negative void reactivity. A three batch fuel shuffling scheme is carried out for seed assemblies to achieve a flat radial power distribution while fresh blanket assemblies are refuelled for each cycle. The flow rate for each assembly is adjusted separately by inlet orifices which remain fixed during the burn-up cycle. The loading patter is optimized to satisfy the criterion for maximum cladding surface temperature through the burn-up cycle. There is no flow mixing between assemblies due to the channel box. The characteristics of the one path core Super FR are summarized is Table 1.

The safety characteristics of the Super LWR and the Super FR are discussed by Oka et al. (2010). As a principle criterion, systems and analysis methods are similar to those of LWRs. Keeping the core flow is the fundamental safety principle of Super LWRs and Super FRs in contrast with keeping the coolant inventory of LWRs. Therefore, the coolant flow is monitored instead of the water level of LWRs. Measurement of coolant flow is easier than that of water level at accident conditions (Liu et al., 2013; Wu and Oka, 2014). The reactor power and pressure are also used as safety signals. The unique feature of the Super LWR and the Super FR is that depressurization is used to provide a flow which is cooling the core under accidental conditions.

The plant and safety systems are depicted in Fig. 7. The configuration is the same for Super LWR and Super FR. There are two main coolant lines for the 1000 MW_e class reactor. It is a once-through direct coolant cycle. The main steam lines are equipped with turbine control valves and turbine bypass valves.

Turbine-driven auxiliary feed water systems (AFS) and motor-driven low pressure core injection systems (LPCI) are applied on the cold-leg side to provide the function of keeping the coolant supply under abnormal conditions. Safety relief valves (SRVs) are applied on the hot-leg side to provide the function as keeping the coolant outlet open and mitigating over-pressurization. The SRVs have also a function as automatic depressurization system (ADS) to induce effective coolant flow. For reactor shutdown, the control rods and the standby liquid control system (SLCS) are employed as in boiling water reactors (BWRs).

Fuel rod integrity is achieved by applying the maximum cladding surface temperature (MCST) criteria of 850 °C for transient events and 1260 °C for accidental events. Fuel pellet integrity is achieved by applying the criterion of the maximum fuel enthalpy of 0.96 kJ/g (230 cal/g) and the criteria for pressure boundary integrity are achieved by applying the maximum pressure of 30.3 MPa for accident events and 28.9 MPa for transient events.

An artist conception of the power plant is shown in Fig. 8. The cross sectional view of the reactor pressure vessel is depicted in Fig. 9. Besides these studies, a Japanese industry group developed the JSCWR concepts with a thermal neutron spectrum and carried out studies with funding by METI. Core design, fuel behaviour, heat transfer, materials and corrosion were published in Sakurai et al. (2009), Higuchi and Sakurai (2010), Komita et al. (2003) and Kaneda et al. (2007). Unfortunately, these activities were halted since the Fukushima accident in 2011.

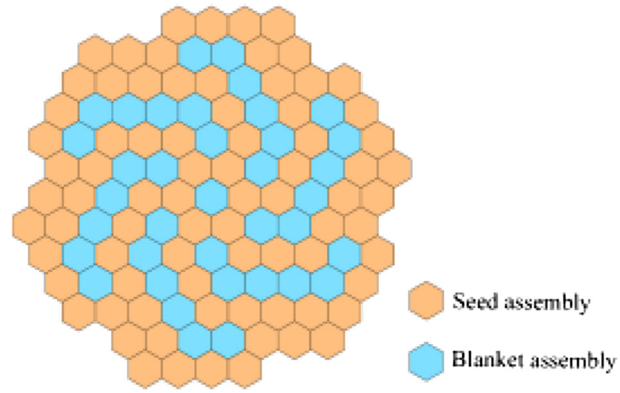

Fig. 6. Loading pattern of the one path core of the Super FR (Liu et al., 2013).

Table 1
Super FR characteristics (Liu et al., 2013).

Power MW_t/MW_e	2337/1006
Coolant pressure (MPa)	25.0
Inlet/outlet temperature (°C)	280/501
MCST in seed (°C) BOEC/EOEC	646/647
MCST in blanket (°C) BOEC/EOEC	560/647
Active/overall power density (kW/L)	206/149
Number of seed assemblies	78
Number of blanket assemblies	37
Active core height (m)	2.4
Eq. active core diameter (m)	2.47
Pu enrichment in seed assembly (wt%)	32 (bottom)/25 (top)
Pu enrichment in bottom blanket (wt%)	10 (bottom)
Cycle length (EFPD)/fuel batch	200/3
Average/max discharge burn-up (GWd/t)	53.8/72.7

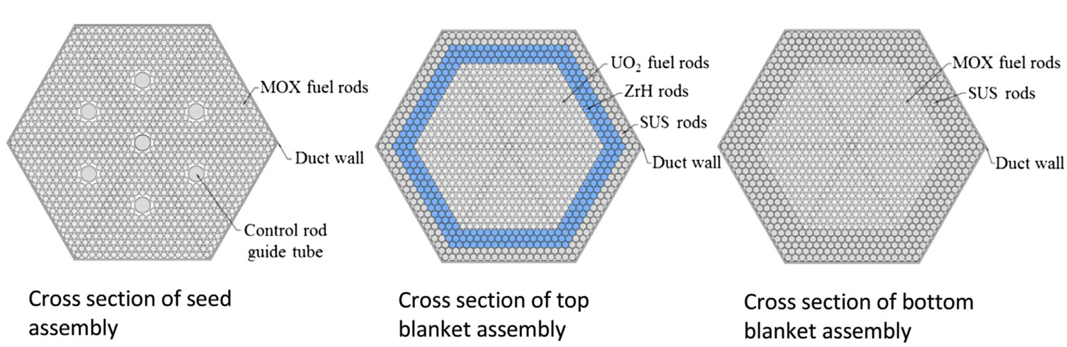

Fig. 5. Cross section of an assembly of the one path core of the Super FR (Liu et al., 2013).

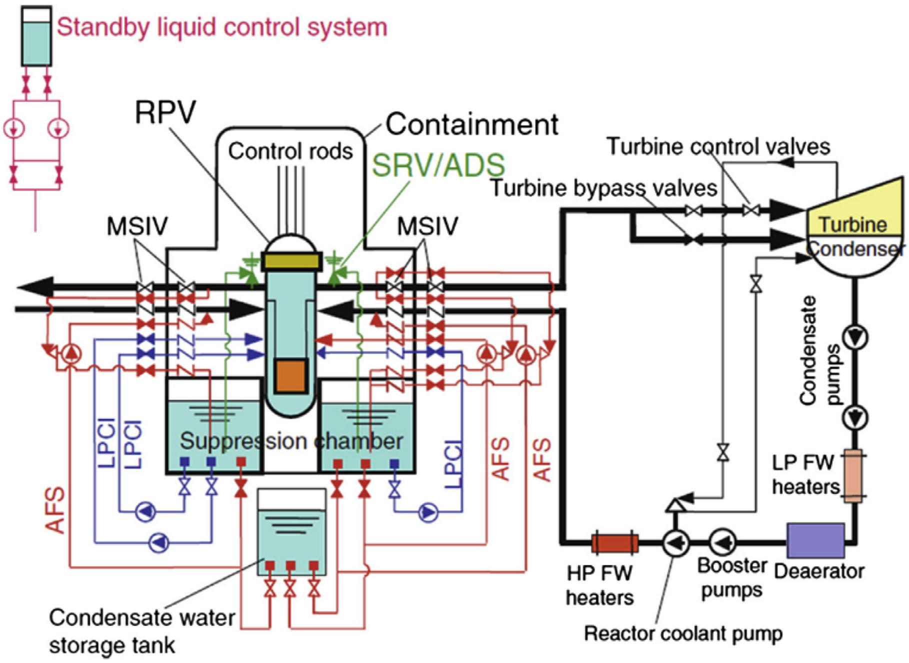

Fig. 7. Plant and safety system of a Super FR and a Super LWR (Oka et al., 2010).

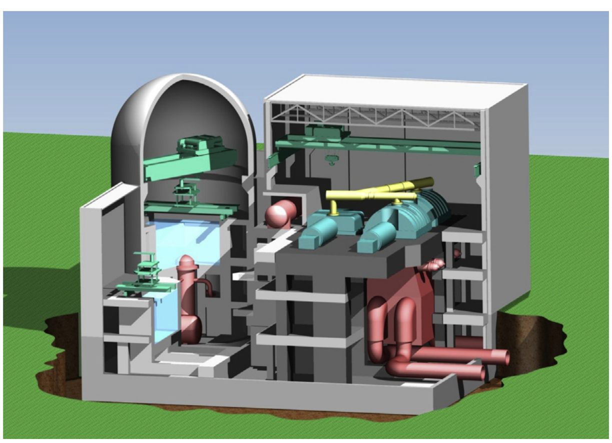

Fig. 8. Super LWR/Super FR power plant (The Japan Journal, 2010).

3.3. Pressure tube type reactors with a thermal neutron spectrum

In Canada, AECL is developing a Super-Critical Water-cooled Reactor (SCWR) concept, which has evolved from the well-established pressure-tube type CANDU®[1] reactor (Yetisir et al., 2013). The main application of the Canadian SCWR is to produce electrical energy. Other potential applications include the generation of process heat, hydrogen, industrial isotopes, and drinking water (through the desalination process) 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 (Duffey et al., 2003).

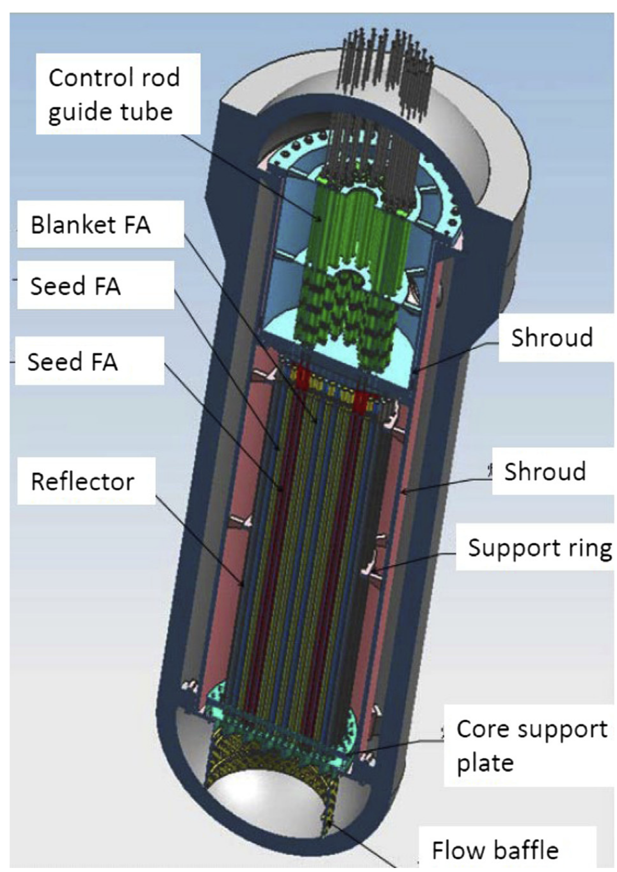

Fig. 9. Cross sectional view of the reactor pressure vessel of Super FR (Oka, 2013).

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

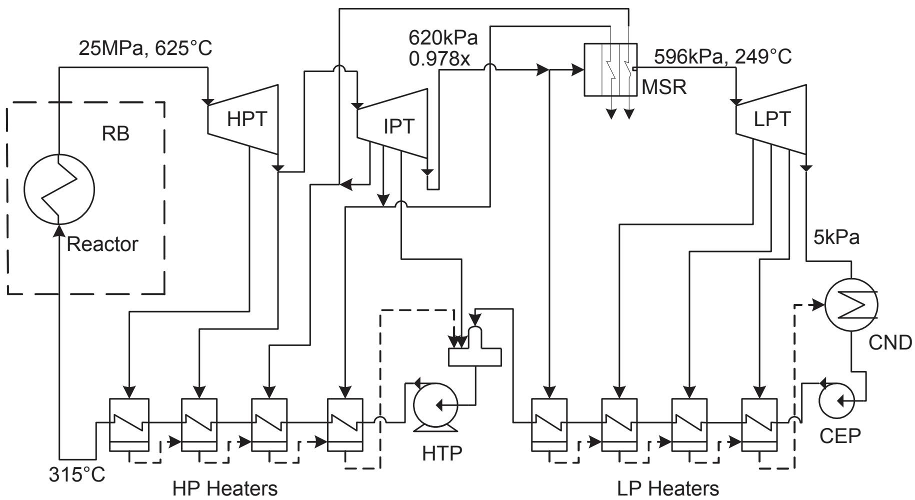

Fig. 10. Schematic of direct steam cycle with an MSR in an SCWR plant (Pioro et al., 2008).

Some of the advanced features of the proposed Canadian SCWR include:

- (Improved) Passive Safety – Passive decay heat removal based on natural circulation and radiation cooling is used to mitigate accident scenarios. The design goal of ensuring "no-core-melt" could be achievable 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;
- (Sustainable and Proliferation Resistant) Thorium Fuel Cycle – The Canadian SCWR can operate with the existing uranium fuel cycle. However, it is being specifically designed with the capability of operating with sustainable fuels, namely the 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;
- (Improved) Economics – At SCW operating conditions, thermodynamic cycle efficiency increases significantly. An increase in cycle efficiency of up to 50% (i.e., from about 33% to 50%) as compared to current nuclear power plants is possible, resulting in reduced generating cost. In addition, the simplified reactor configuration reduces the capital cost compared to the current fleet of nuclear reactors.

3.3.1. *Steam 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 (Yetisir et al., 2013). 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 Fig. 10 (Pioro et al., 2008). The heat transport system (HTS) coolant (i.e., 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 Fig. 10.

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. Fig. 11 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.

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 fluxes 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.

3.3.2. *Pre-conceptual core design*

The pre-conceptual Canadian SCWR maintains a modular configuration with separated coolant and moderator, as in current CANDU reactors. It is developed to generate 2540 MW of thermal power and about 1200 MW of electric power (assuming a 48% thermodynamic cycle efficiency of the plant). A batch refuelling

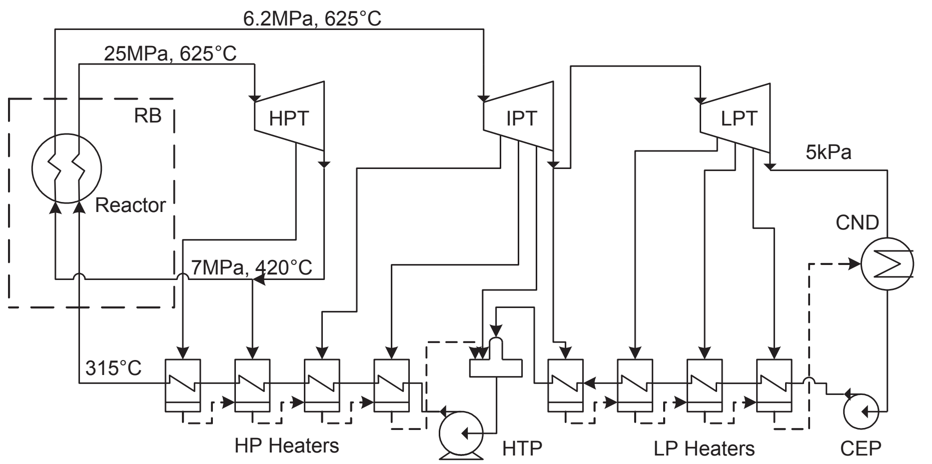

Fig. 11. Schematic of direct steam cycle with reheat in an SCWR plant (Pioro et al., 2008).

strategy is adopted as the current CANDU practice of on-line refuelling is extremely challenging at the proposed higher operating pressure and temperature. This has led to a simplified vertical core design with vertical fuel channels, each containing a fuel assembly. Fig. 12 illustrates schematically the pre-conceptual Canadian SCWR core.

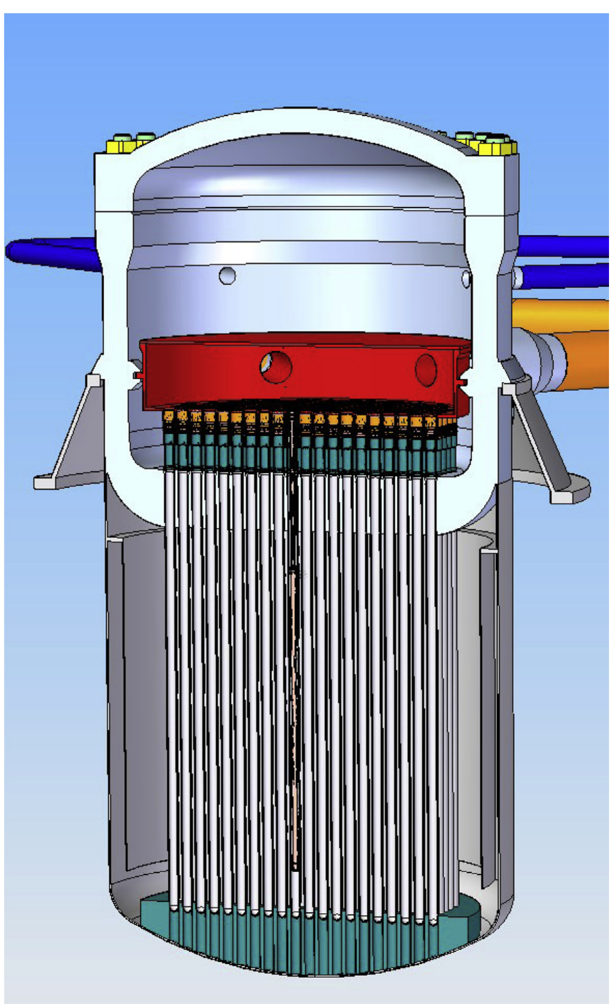

Fig. 12. Schematic diagram of the pre-conceptual Canadian SCWR core.

The pre-conceptual Canadian SCWR core consists of 336 fuel channels, each housing a 5-m long fuel assembly. The average fuel channel power is 7.56 MW_{th} and the core radial power peaking factor is estimated to be 1.32. The lattice pitch of the channels 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 (McDonald et al., 2011). Fig. 13 shows the layout of fuel channels. Some fuel channels at the outer region of the core could be used for the reheat option.

Fig. 14 illustrates the flow paths within the Canadian SCWR core. The light water coolant enters the inlet plenum through inlet nozzles and then enters the fuel channels. 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, each about the same size as the pressure tube. The pressure tube consists of two sections; a stainless steel section at the tubesheet area and a zirconium alloy section at the active zone. The two sections are fused together creating a seamless joint.

The light water enters the fuel channel through several openings in the fuel-channel assembly, which act as orifices sized to control the flow rate for the purpose of achieving a more uniform exit coolant-temperature distribution. Fig. 15 illustrates the fuel channel configuration and the flow path inside the fuel channel. After entering the fuel channel, the light water enters the centre flow tube in the fuel assembly region through four nozzles. It travels down the centre flow tube to the bottom of the fuel channel and up the fuel assembly, and discharges to the outlet plenum. The outlet plenum is contained inside the inlet plenum. This configuration minimizes the pressure and temperature requirements on the outlet-plenum material. An insulating coating will be put on the outside surface of the outlet plenum to minimize heat losses.

The fuel-channel configuration in the fuel-assembly region consists of the pressure tube and an insulator encapsulated with a stainless-steel jacket (Fig. 16). This fuel channel design is called the high-efficiency re-entrant channel (HERC) (Nava-Dominguez et al., 2013), based on a proposal by Chow and Khartabil (2007). The pressure tube is designed to withstand the high coolant pressure at a low temperature (~100 °C), achieved by direct contact of the pressure tube with the moderator. This allows the use of the zirconium alloy Excel for the pressure tube.

The insulator thermally protects the pressure tube from the high temperature coolant flowing through the fuel assembly. It is made of Yttrium-Stabilized Zirconia (YSZ), which is refractory, has low

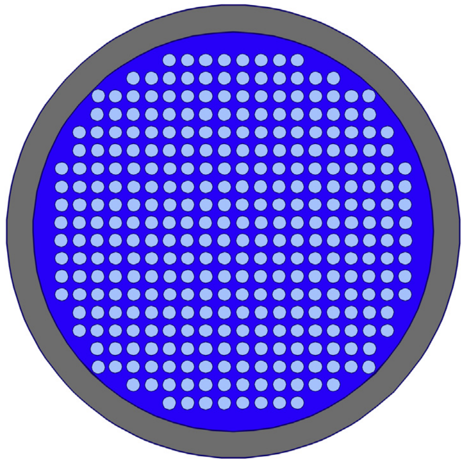

Fig. 13. Cross-section layout of fuel channels in the Canadian SCWR core.

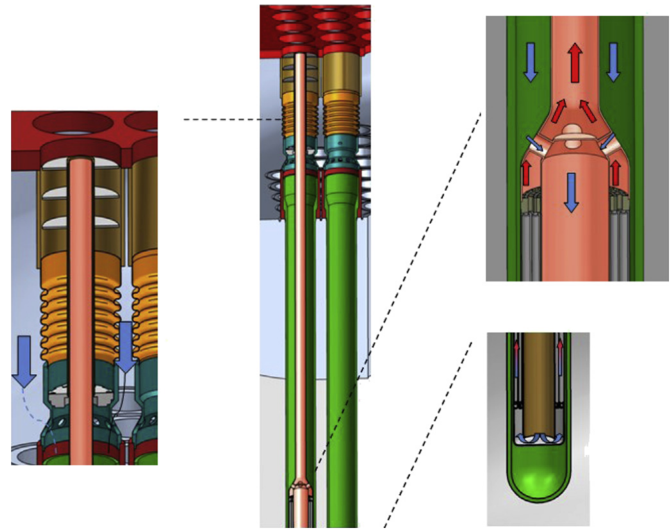

Fig. 15. Fuel channel configuration.

neutron absorption properties and excellent resistance to neutron damage. The insulator is encapsulated with a stainless-steel jacket, which will minimize damage to the insulator by the movement of the fuel assembly and prevent any fragments of the insulator from getting into the coolant stream and damaging the turbines.

One of the expected benefits of the HERC concept is that in the event of a loss of coolant accident (LOCA) without emergency core cooling, the fuel will not melt because of passive heat rejection through the insulator into the moderator (Shan et al., 2011). That is, the heat in the fuel will be transferred via radiation to the liner tube and conducted to the moderator, maintaining the fuel cladding temperature below its melting point. To achieve this safety goal, the fuel channel requires further optimization of its geometry, and materials properties to ensure sufficient decay heat removal during accidents conditions while minimizing heat loss during normal operating conditions.

The fuel-channel outlet is connected to the outlet plenum through a pressure tube extension and an expansion bellow. The

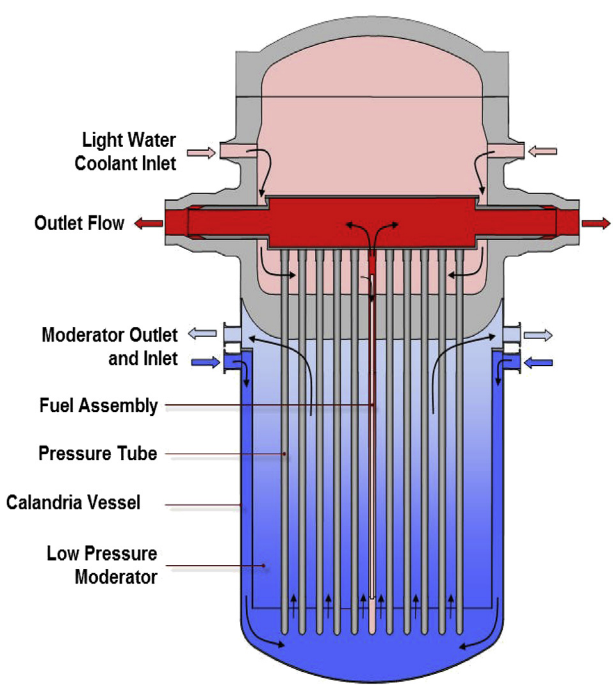

Fig. 14. : Cross sectional view of the pre-conceptual Canadian SCWR design.

pressure-tube extension is joined to the pressure tube by co-extrusion making it possible to weld the fuel channel to the tubesheet. The fuel channel is connected to the tubesheet using a weld at the top of the tubesheet that prevents the high-pressure inlet stream from leaking into the low-pressure moderator. An expansion bellow at the end of the fuel channel allows for the differential thermal expansion of the fuel channels as well as the relative thermal expansions of the outlet plenum and the inlet plenum. The YSZ insulator extends into the tubesheet to ensure that the tubesheet doesn't get exposed to high outlet temperatures.

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. 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 is in progress. Feasibility of inserting shutoff rods diagonally at an angle using gravity is also considered.

Since the SCWR is a water-cooled reactor, a key challenge will be to define a water chemistry control strategy. The key chemistry issues are a) water radiolysis, which can produce oxidizing species that will increase the corrosion of components in-core and immediately downstream of the core, and b) corrosion product deposition on the fuel cladding, which can lead to overheating, fuel failure and high radiation fields on downstream components, affecting maintenance (Guzonas et al., 2008). These issues are the subject of a number of research programs now underway in Canada.

3.3.3. 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 to enhance the sustainability of the nuclear fuel cycle by extending the availability of current resources through the use of thorium fuel cycles. Thorium fuel has only been used in research reactors and demonstration irradiations were performed in power reactors. Recent studies of thorium-based fuel cycles in contemporary CANDU reactors demonstrate the possibility for

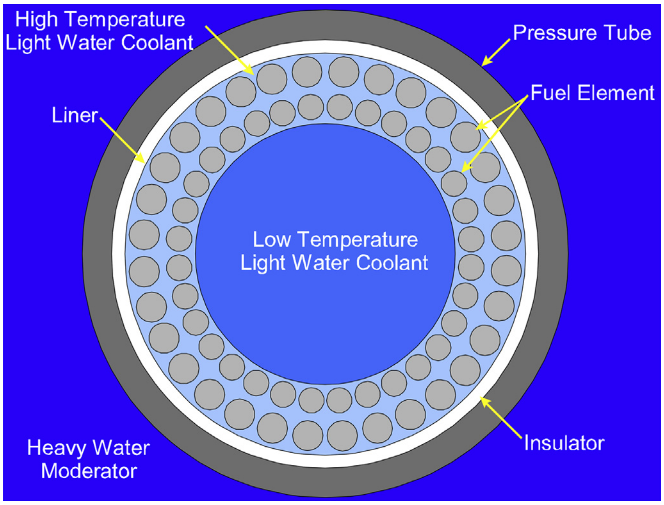

Fig. 16. Schematic diagram of the High Efficiency Re-entrant Channel (HERC) (Nava-Dominguez et al., 2013).

substantial reductions in natural uranium (NU) requirements of the fuel cycle via the recycle of U-233 bred from thorium (Hopwood et al., 2006; Dyck et al., 2004; Boczar et al., 2002; Hyland et al., 2009). 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 (Magill et al., 2011).

3.3.4. Safety system design

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 pseudo-critical temperature. Fig. 17 illustrates schematically the safety system features. 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 its 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 a LOCA within the HTS.
- One of the expected benefits of using the HERC is that in the event of a LOCA without emergency core cooling, the fuel will not melt because of passive heat rejection through the insulator

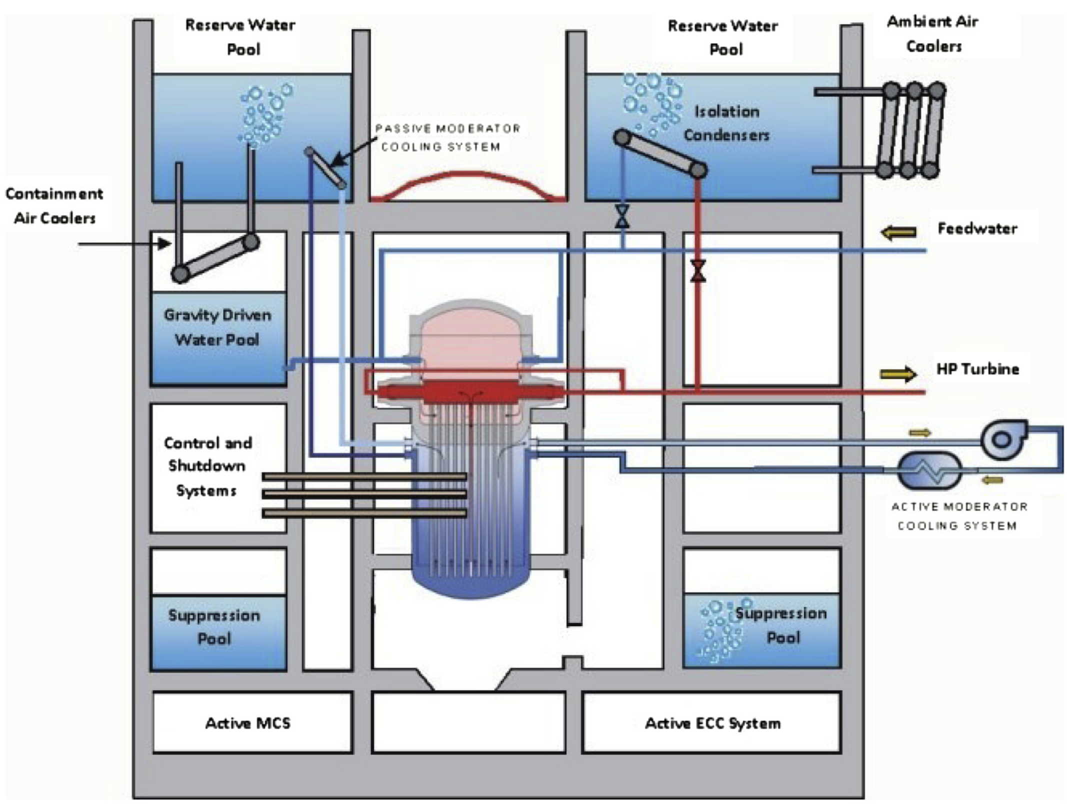

Fig. 17. Schematic diagram of the safety system.

into the moderator (Nava-Dominguez et al., 2013). That is, the heat in the fuel will be transferred via radiation to the liner tube and conducted to the moderator, maintaining the fuel cladding (a stainless steel is the reference) below its melting point. Work is proceeding to optimize and demonstrate HERC 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 Fig. 18). 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 (Jeddi et al., 2011). A large-scale test facility is being designed to qualify the system.

4. Collaboration in the Generation IV International Forum

These examples demonstrate that the use of supercritical steam cycles is not limited to a single reactor design concept. Therefore, Canada, Japan, and Euratom member states signed a system arrangement in 2006 for international research and development of SCWR nuclear energy systems. Recently, in 2011, Russia joined this consortium.

There are currently four Project Management Boards (PMBs) within the SCWR System: 1) System Integration and Assessment (provisional), 2) Materials and Chemistry, 3) Thermal-hydraulics and safety, and 4) Fuel Qualification Testing (in preparation). Table 2 lists the members and shows the status of these PMBs. In addition, Russia signed the SCWR System Arrangement in 2011 and expressed its interest to join the projects. Results on design and technology are shared every 2 years at the International Symposium on SCWR and in a large number of written reports.

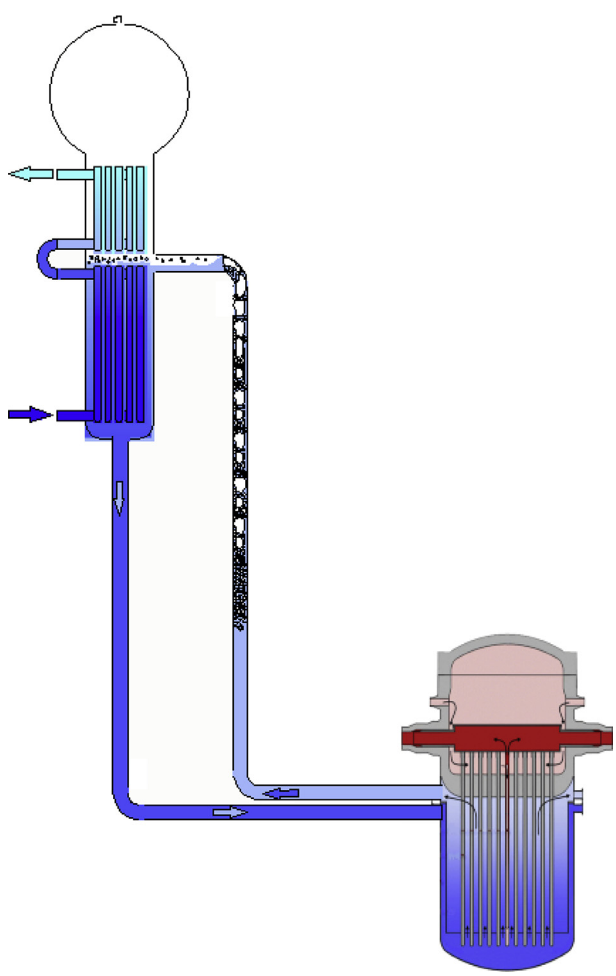

Fig. 18. Schematic diagram of the passive moderator cooling system.

Table 2
Status and memberships of SCWR System Arrangement and Project Arrangements.

SCWR System Arrangement and Project Arrangements	Signatories	Date of signature
System Arrangement	Canada, Euratom, Japan Russia	November 2006 July 2011
Thermal-Hydraulics and Safety Project Arrangement	Canada, Euratom, Japan	October 2009
Material and Chemistry Project Arrangement	Canada, Euratom, Japan	December 2010
Fuel Qualification Test Project Arrangement (in preparation)		
SCWR System Integration and Assessment Project Arrangement (provisional)	Managed by the System Steering Committee	–

An SCWR System Research Plan has been worked out, identifying the following critical-path R&D tasks:

- System integration and assessment—Definition of a reference design, based on the pressure tube and pressure vessel concepts, that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance.
- Thermal-hydraulics and safety—Significant gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are still needed. The design-basis accidents for a SCWR have some similarities with conventional water reactors, but the difference in thermal-hydraulic behaviour and large changes in fluid properties around the critical point compared to water at lower temperatures and pressures need to be better understood.
- Materials and Chemistry—Selection of key materials for use in in-core and out-core components of both pressure tube and pressure vessel designs. Selection of a reference water chemistry which minimizes materials degradation and corrosion product transport will also be sought based on materials compatibility and an understanding of water radiolysis.
- Fuel qualification test—An important collaborative R&D project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design. As a SCWR has never been operated before, such generic testing is considered to be mandatory before a prototype reactor can be licensed.

So far, the SCWR System Integration and Assessment project included the development of different conceptual design options, as discussed in Section 3. A project arrangement has not been required during this viability phase and the SCWR steering committee was managing the overall progress instead. The development of technologies, which are common for all these options, have been agreed in project arrangements as will be discussed next.

4.1. Thermal-hydraulics and safety (TH&S) project

Supercritical water is a single phase fluid having liquid-like properties below the pseudo-critical temperature (384 °C at 25 MPa) and steam-like properties above this temperature. Heat transfer of supercritical water differs fundamentally from ordinary fluids in the vicinity of the pseudo-critical point, where the fluid properties vary significantly with temperature. Heat transfer in this range can be enhanced at low heat flux compared with ordinary fluids, or deteriorated at high heat flux and low mass flux, causing local hot spots on the heated surface. Prediction of such hot spots still remains a challenge. Up to now, simple heat transfer correlations cannot predict these phenomena properly and computational fluid dynamics (CFD) or even large eddy simulations are taken instead. Similar questions arise with critical flows through orifices or breaks and with stability limits of supercritical fluids in heat exchangers if the pseudo-critical point is located in the computational domain. New physical models and codes describing these phenomena need to be validated by experiments with supercritical water or at least with surrogate fluids having similar properties, like supercritical CO_2 or refrigerants.

A second part of this project covers innovative concepts of safety systems for SCWRs and their transient analyses with system codes. These include loss of coolant accidents, loss of power accidents, loss of flow accidents, and other scenarios which may cause a risk for the power plant and should be assessed conceptually, accompanying the design work. Collaboration with the Risk and Safety Working Group of GIF enables a comparison with other GIF systems.

According to the SCWR System Research Plan, the time frame until end of 2012 was used for heat transfer tests with supercritical water and with other, surrogate fluids in tubes, annuli and bundles as well as for stability and critical flow tests. Integral tests of the envisaged safety systems are foreseen for a later phase. Today, suitable concepts of active safety systems are available for SCWRs, but passive safety systems and their experimental validation still remain to be a challenge.

4.1.1. Canadian TH&S activities

Canada has been focussing on establishing infrastructure for thermal-hydraulics research. A number of test facilities have been designed and constructed in Canada. These facilities are established mainly for heat-transfer tests with tubes, annuli, and bundle subassemblies in water, carbon dioxide, or refrigerant flows. At this point, the design of the water-test facility is complete and construction has been initiated. A refrigerant and a carbon dioxide test facilities have been constructed for supercritical heat-transfer experiments covering test sections including tubes, annuli, a 3-rod bundle, a 4-rod bundle, and a 7-rod bundle. Fig. 19 shows a view of the carbon-dioxide test facility (Jeddi et al., 2011). Axial surface-temperature distributions were obtained with 8 mm and 22 mm tubes.

At the subcritical pressure of 6.7 MPa (i.e., lower than the critical pressure of 7.38 MPa for carbon dioxide), nucleate boiling is observed for length-to-diameter ratios up to about 180 at a mass flux of 451 kg/(m^{2}s) and 270 at the mass flux of 1476 kg/(m^{2}s) (see Fig. 20). Departure from nucleate boiling occurred at these locations, beyond which film boiling is observed.

At supercritical pressures of about 9 MPa, deterioration heat transfer is observed with a sharp rise in surface temperature at a mass flux of 425 kg/(m^{2}s) and a heat flux of 42 kW/m^2 (length-to-diameter ratio of about 160) (see Fig. 21). Another surface-temperature peak observed at the length-to-diameter ratio of about 190 corresponds to the pseudo-critical temperature. Deterioration heat transfer is not observed at other test conditions in the figure.

A heat-transfer experiment has been completed with annuli of two different flow areas in supercritical water flow (Wu et al., 2011). The inner heater element has an outer diameter of 8 mm, while two different outer unheated flow tubes with inside

Fig. 19. Upper view of the heat transfer test facility with carbon dioxide flow (Jeddi et al., 2011).

diameters of 16 mm (i.e., 4-mm gap size between inner and outer tubes) and 20 mm (i.e., 6-mm gap size) have been used. The test section was installed vertically in the loop and tested with an upward flow of supercritical water. Inlet and outlet fluid temperatures, outlet pressure, and pressure drop over the test section were measured. Wall temperature measurements have been obtained over a range of mass fluxes and heat fluxes at outlet pressures of 23, 25, and 28 MPa. Fig. 22 illustrates variations of wall temperature, and corresponding heat-transfer coefficient, with local enthalpy and heat flux. Deteriorated heat transfer has been observed at a heat flux of 1000 kW/m^2.

Surface-temperature measurements were also obtained for the assessment of effects of gap size (or flow area) and spacers on heat

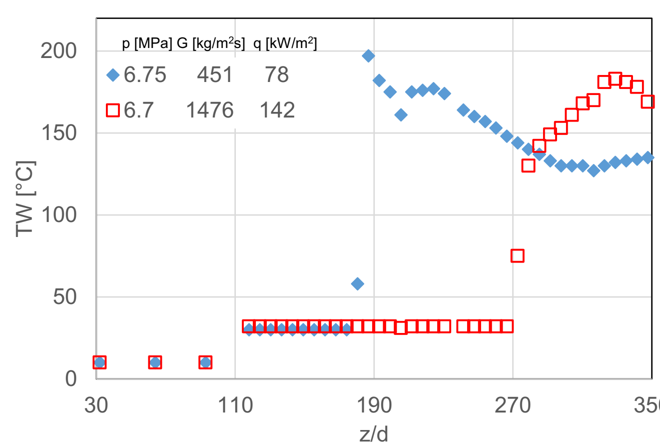

Fig. 20. Wall temperature measurements at sub-critical pressures on CO_2 in an 8-mm tube with carbon dioxide flow (Jeddi et al., 2011).

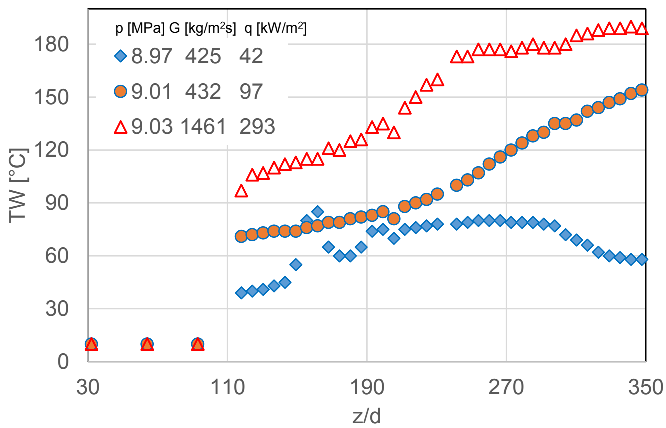

Fig. 21. Wall-temperature measurements at supercritical pressures of CO_2 in an 8-mm tube with carbon dioxide flow (Jeddi et al., 2011).

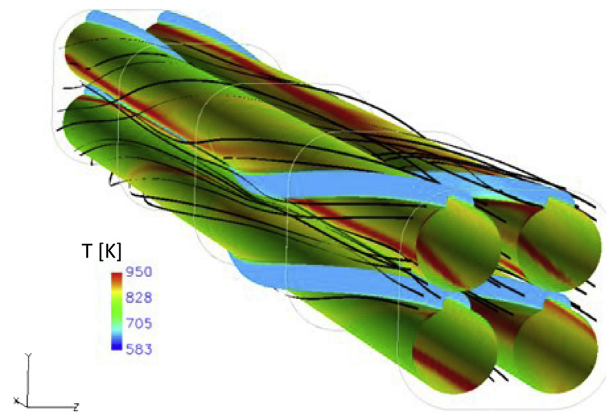

Fig. 23. Predicted surface temperature in K on fuel rods with wire wrap (Chandra et al., 2010).

transfer in annuli. Enhanced heat transfer in the annular test section was shown with the 6 mm gap size, compared to 4 mm gap size, at similar local conditions and heat flux. The difference is larger at low heat flux and high mass flux conditions than at high heat flux and low mass flux conditions. The effect of spacers is strong on heat transfer. Heat-transfer coefficients at the location of the spacer are consistently larger than those at locations further away of the spacer.

A supercritical heat-transfer database has been expanded to include water and carbon dioxide data previously obtained at the University of Manchester. These data cover mainly the mixed-convection region and are applicable for model development and validation.

A look-up table for heat-transfer coefficients covering subcritical and supercritical conditions has been developed. It covers two film-boiling regions (i.e. inverted annular flow and dispersed flow) at subcritical pressures and three regions (i.e. liquid-like, gas-like and pseudo-critical) at supercritical pressures.

4.1.2. TH&S activities in Europe

The European consortium has been working on predictions of heat transfer. An example of a numerical prediction of heat transfer in a rod bundle is shown in Fig. 23. These CFD analyses of Chandra et al. (Chandra et al., 2010) show the surface temperature of fuel rods at a heat flux of 1375 kW/m^2 and a mass flux of 1332 kg/m^{2}s, predicted with FLUENT. The bulk temperature is 310 °C at the inlet. A wire is wrapped around the fuel rods to serve a grid spacer and as a coolant mixing device. The analysis shows that the wire is also improving the local heat transfer.

A safety concept has been developed and analysed for the HPLWR, as summarized by Schulenberg and Starflinger (2012). The main safety functions and appropriate strategies for accident control have been identified, and the key parameters for the operation of the systems have been selected. The transient analyses performed addressed a variety of initiating events, including anticipated transients as well as accidents. The analyses show that the safety systems can effectively limit overheating of the core under the most severe conditions, such as loss of coolant accidents and loss of flow transients.

A Dutch program on the stability of a natural-circulation driven SCWR has been finalized in 2010 by the Delft University of Technology (T'Joen and Rohde, 2012). The project encompassed both numerical work and experiments. It has been found that the stability shows similarities with a natural-circulation driven boiling water system, but one major difference is that it is possible to follow a trajectory from zero power conditions to nominal conditions without crossing an unstable region. The origin of this finding is the gradual change of the density of the supercritical water with respect to the temperature.

For future collaboration in TH&S, two programs in the field of turbulent heat transfer in supercritical flows have been initiated. One program, called THINS, started in 2010 and includes a specific work package on non-unity Prandtl number, turbulent flows (Roelofs et al., 2011). The second program is of Dutch origin. Local measurements will be taken with Laser-Doppler Anemometry and Particle Image Velocimetry to validate the models.

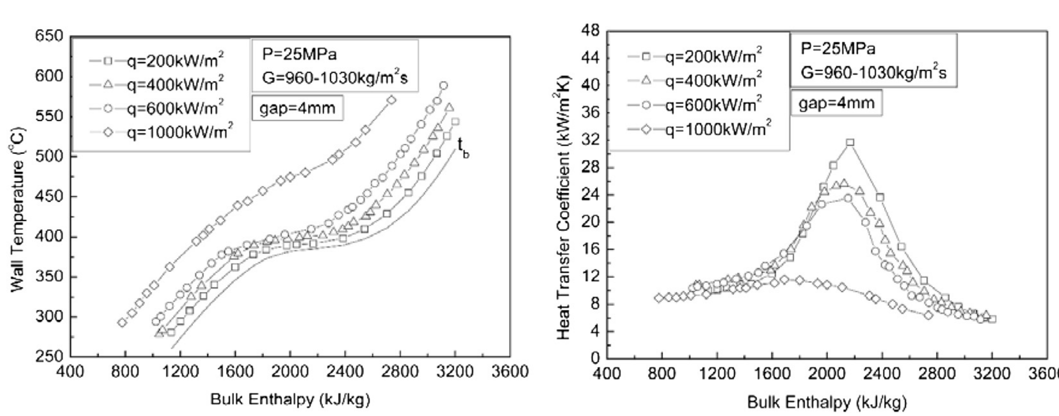

Fig. 22. Wall temperature measurements obtained from the super-critical water heat-transfer test with an annulus (Wu et al., 2011).

4.1.3. Japanese TH&S activities

In Japan, the development of the best estimate correlations on heat transfer and pressure drop was continued based on technical papers published by foreign researchers. Moreover, development of a thermal-hydraulic analysis method for thermal design of a SCWR was considered.

As for development of the thermal-hydraulic analysis method, consideration of the heat transfer augmentation due to spacers settled on the outer surface of fuel rods was performed. An effect of the heat transfer augmentation due to a spacer was taken into consideration in order to reduce the maximum fuel cladding surface temperature (MCST) from the current core design value. The target of a MCST decrease is 30–50 K. Spacer shapes were also considered to enhance the heat transfer coefficients.

As an example, the turbulent flows in the fuel assembly with a vane type spacer were predicted using a computational fluid dynamic tool, as can be seen in Fig. 24. Fig. 24(a) shows a case without any vane and Fig. 24(b) shows a case with a vane.

Each numerical domain contains 2 × 2 fuel rods. The fuel rod diameter is 9.5 mm, the gap width between adjacent fuel rods is 3.1 mm, and the hydraulic diameter is 11.7 mm. The number of computational grids is 192 × 192 × 640. Fig. 28 shows the results under the conditions of supercritical water without any heat flux. Unsteady vortex structures are observed behind the spacer. By generating a large swirl flow due to the vane on a spacer, it was clarified quantitatively that turbulent intensities are strengthened.

4.1.4. Mutual benchmark study

The TH&S Project Management Board is organizing an international benchmarking exercise against supercritical water data obtained with a heated 7-rod bundle assembly. These data were obtained at Japan Atomic Energy Agency (JAEA) and have been submitted to the Project Management Board (PMB) as part of Japan's contribution. Facility data and experimental conditions were provided beforehand, and the experimental data afterwards. The benchmarking results will be presented during a workshop in June 2014 in Delft, The Netherlands.

This benchmark consists of a well-defined 7-rod bundle flow with grid spacers. The facility is equipped with a large number of thermocouples on the outer wall of the heating rods. Care has been taken to the azimuthal symmetry of the rod internals, ensuring a uniform distribution of power (or heat flux).

4.2. Materials and Chemistry (M&C) project

The identification of appropriate materials for in-core and out-core components is one of the major challenges for the development of the SCWR. For any SCWR core design, materials for reactor internals and fuel cladding need to be evaluated and qualified. Zirconium-based alloys, so pervasive in conventional water-cooled reactors, do not appear to be viable fuel cladding materials given the high peak cladding temperatures of the SCWR concepts. Based on the available data for other alloy classes, there is no single alloy that currently has received enough study to unequivocally ensure its performance in an SCWR. Although considerable experience is available for fast reactors and SCW-cooled FFPs, there is little or no data on the behaviour of these materials inside an SCWR at the temperature and pressure of interest.

Another key component of this program is to develop an enhanced understanding of the chemistry of supercritical water. The marked change in the density of supercritical water through the critical point is accompanied by dramatic changes in chemical properties. These complications are further exacerbated by in-core radiolysis, which on-going studies show is markedly different from simplistic extrapolations of the behaviour encountered in conventional water-cooled reactors.

Up to 2012, a large number of corrosion and stress corrosion cracking (SCC) tests have been conducted by all participants. The current database includes information on the general corrosion of more than 90 alloys. Tests have been conducted in a variety of

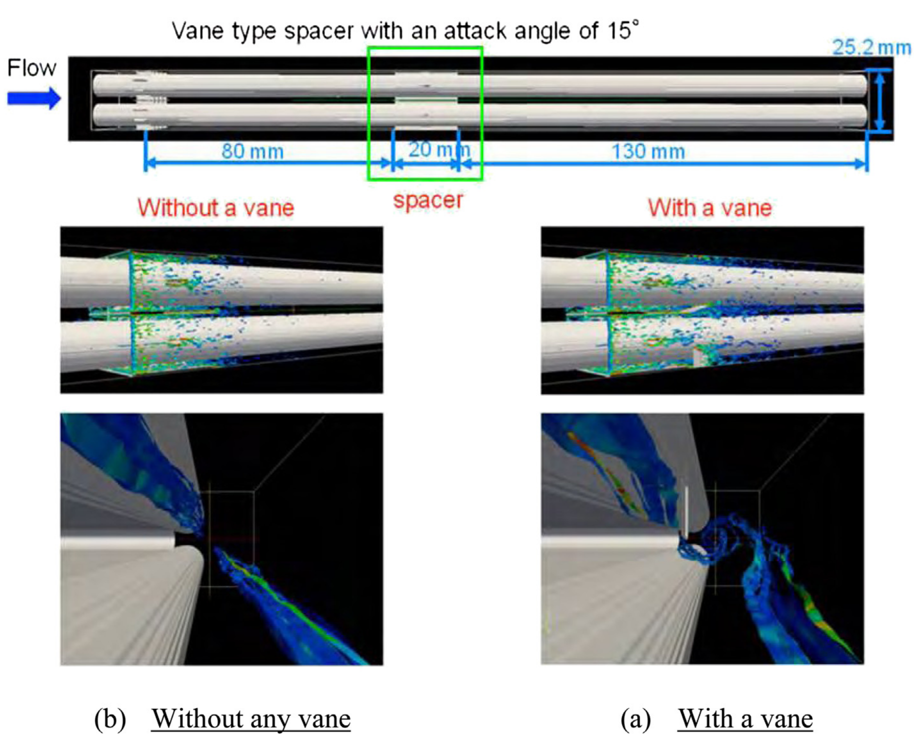

Fig. 24. Generation of large turbulence structures around fuel rods due to the vane on a spacer.

facilities including static autoclaves, flow loops and pressurized capsules; a number of new test facilities for this purpose have been commissioned in the last four years.

Data on general corrosion for a range of candidate materials (including ferritic–martensitic steels, austenitic stainless steels, ODS steels and titanium) show a general trend of increasing corrosion resistance with increasing Cr content, as sketched in Fig. 25. While stainless steels with less than 20% Cr are expected to fail as the corrosion depth would exceed the wall thickness, a Cr-content of around 25% has a potential to meet the design requirements of less than 10% corrosion depth within 50 000 h at 700 °C. The beneficial effect of high Cr content in an SCWR is contingent upon being able to control water radiolysis, as the Cr oxides responsible for the corrosion resistance become soluble under very oxidizing conditions.

Weight change measurements and a variety of metallographic and surface analysis techniques have been used to characterize high Cr materials following exposure to supercritical water. A significant finding was the rediscovery of the large effect of surface microstructure on corrosion for austenitic steels, e.g., polished versus machined. If the relevant mechanisms can be better understood, this may provide a means of imparting a higher corrosion resistance to materials currently not considered ideal candidates for use in an SCWR.

A major activity of the M&C PMB has been the organization of a round robin corrosion test program to compare the results of corrosion tests in different test facilities using a standard test protocol and coupon preparation method. The tests are now underway and will be completed in 2013.

4.2.1. Canadian M&C activities

The core outlet temperature of the Canadian SCWR concept is higher than those of the EU and Japanese designs, and this presents a major challenge for material development. A significant amount of new infrastructure required for SCWR materials testing (autoclaves, corrosion and SCC test loops, creep apparatus) as well as facilities for production of oxide dispersion-strengthened alloys has now been established in Canada. A key activity has been the ongoing development of a corrosion database to capture experimental data generated by Canadian R&D projects and GIF collaborations, as well compiled from open literature. A large parametric study of the effects of temperature, pressure, water chemistry and surface finish on corrosion was completed. Work is on-going to better understand corrosion mechanisms in SCW and to perform fundamental studies of oxidation resistance and corrosion mechanisms using model binary and ternary alloys, and molecular dynamics simulations of the structure of supercritical water at surfaces. The use of ceramic or metallic coatings to improve corrosion resistance of key components continues, including ceramic coating of P91 and zirconium alloys, and testing of NiCrAl(Y) and similar materials in supercritical water for up to 5000 h. A key enabling technology is the insulator required for the insulated fuel channel concept, and a major program is underway to develop and test candidate ceramic materials.

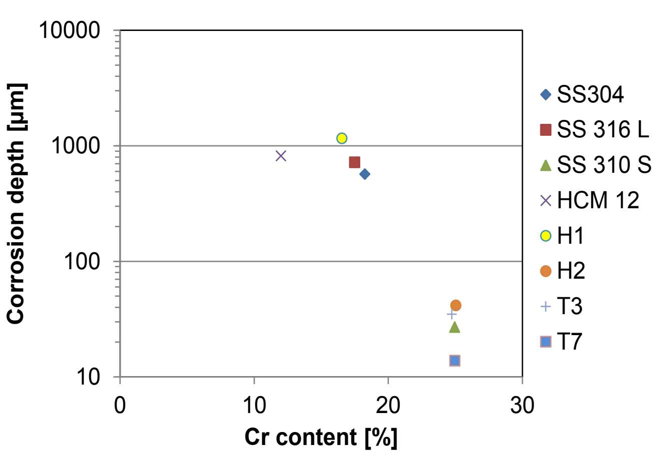

Fig. 25. Predicted corrosion depth after 50 000 h at 700 °C in supercritical water.

The specification of a chemistry control strategy is a major focus of the Canadian program, in particular the understanding of water radiolysis and corrosion product transport. In Canada, both experimental and modelling approaches are being used to develop an improved understanding of water radiolysis in SCW. The existing Monte Carlo model has been benchmarked against a recently released state-of-the-art assessment of all existing sub-critical water radiolysis data. Molecular dynamics simulations were carried out at different densities and temperatures to obtain a detailed picture of the heterogeneous molecular structure of SCW, needed to determine how this structure influences radiation energy deposition and subsequent radiolysis reactions. Work is on-going to determine the solubility of relevant metal oxides (e.g., magnetite, molybdenum oxides) and predict the corrosion product deposition. Initial studies of a model fission product (strontium) showed that neutral species are important at moderate concentrations at 350 °C; their solubility in SCW is sufficient to allow transport out-of-core.

4.2.2. Japanese M&C activities

Results from a detailed study on the corrosion of commercial SUS310S austenitic stainless steel and three other experimental alloys proposed by Hitachi (H2) and Toshiba (T3 and T7), shown in Fig. 25, are among the most promising candidates. While there is no overall consensus on the best material for fuel cladding yet, there was general agreement that the Hitachi H2 modified 310 stainless steel containing Zr is the best candidate to be used as the reference material for the fuel qualification testing. However, more test data are needed at temperatures up to 700 °C for material qualification; the required tests could e.g. be performed in the VTT autoclave, which can reach up to 695 °C, or in high temperature, low pressure steam, which was recently shown to be a good surrogate for supercritical water above about 550 °C (Guzonas and Cook, 2012).

4.2.3. European M&C activities

In the frame of European FP7 projects, SCC and general corrosion tests have been performed on ODS and austenitic steels to determine their corrosion resistances in SCW. In addition, environmental effect on creep rate has been studied on selected austenitic candidate alloys.

The higher material temperature, irradiation, and the thin walls of core components such as moderator box and fuel claddings are a combination of requirements which are more difficult to fulfil than in supercritical coal fired power plants. In this case, choices will be mainly high-performance stainless steels or novel oxide dispersion strengthened (ODS) steels. Other core components operating at 500 °C may apply commonly used austenitic stainless steels. Ni-based alloys are excluded in core components because nickel has high neutron absorption cross-section and hence high Ni content adversely affects core neutronics. For thin-walled components especially, such as fuel cladding in the SCWR design, corrosion, stress corrosion cracking (SCC), and creep resistance are among the severest degradation modes needing to be understood and controlled.

Based on the results, the applicability of low Cr (<∼17 weight-%) austenitic and ODS steels is limited to temperatures less than

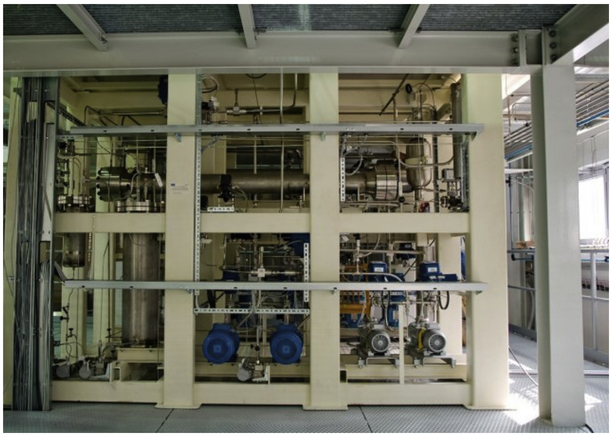

Fig. 26. Auxiliary unit for in-pile radiolysis and water chemistry tests with supercritical water at UJV Řež, Ruzickova et al. (2011).

550 °C. At 550 °C, the oxidation rate increases rapidly for both alloy groups. In terms of general corrosion resistance, increasing Cr contents, e.g., 20% Cr, could extend the maximum operating temperature to 650 °C or even higher. Creep test results, however, indicated that thin walled components made of austenitic stainless steels are prone to environmentally enhanced creep. This phenomenon requires further study. Cold working of the austenitic stainless steel (17–18% Cr) surface appears to suppress oxidation significantly up to 650 °C for a substantial exposure time (at least up to 3000 h). However, in SCWR, the exposure times are much longer.

In-pile tests of radiolysis and its effect on the water chemistry and corrosion are being prepared at Research Centre Řež (CVR). A supercritical water loop with an active channel inside the LVR-15 reactor has been constructed and commissioned in an out-of-pile test installation. It is ready for in-pile testing and will be installed inside the research reactor LVR-15 as soon as the required construction work in the reactor building is completed. The auxiliary unit with heaters and coolers, the purification system, water chemistry monitoring and sampling, and the dosing system are shown in Fig. 26. Details of this system and recent results have been described by Ruzickova et al. (2011). Compared with the schedule of the System Research Plan, these important in-pile material and chemistry tests have had some delays to improve shielding of the test facility.

During 2011, 2012, the behaviour of austenitic steels 316L and 08CH18N10T (AISI 321) and ferritic-martensitic steels used for evaporator and boiler components in supercritical water cooled fossil fuelled power plants (P91, P92, Super304H, HR3C) were studied. The test conditions were just above the critical point (400 °C, 25 MPa) to simulate evaporator conditions of the HPLWR. The loop is currently being used out-of-pile and the experiments performed are adapted for supercritical water cooled fossil fuelled power plants. The facility is now completely functional and experiments to support development of new equipment (e.g., specimen holders for specific measurements such as mechanical stress, etc.) for SCWR research are being performed (Figs. 27 and 28). Specimen holders can be equipped with three point bending tests and interfaces for special sensors connections.

The experiments performed in 2012 focused on the influence of water chemistry on the corrosion behaviour. The experiments were performed in pure water and using oxygen-ammonia water chemistry. The results are currently being evaluated using techniques such as SEM, gravimetry, ESCA, Mott–Schottky plots, etc. In 2013 the experiments will continue at the temperature of 600 °C and pressure of 25 MPa.

Fig. 27. Specimen holder before exposure in the loop.

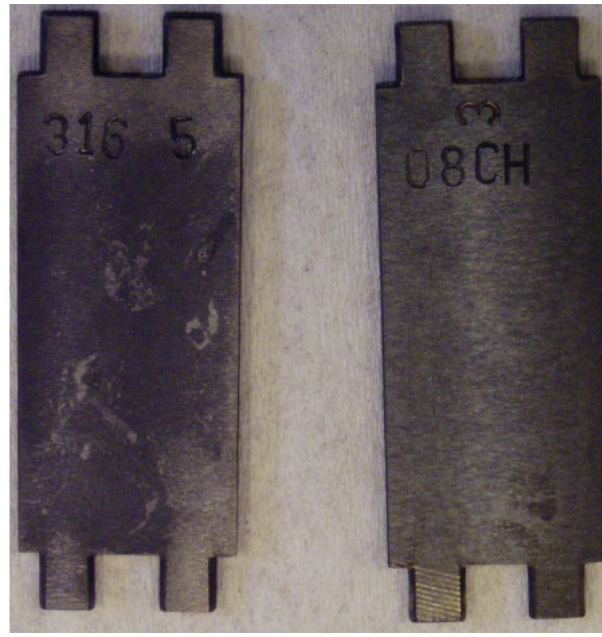

Fig. 28. 316L and 08CH18N10T specimens after exposure in the Czech supercritical water loop.

5. Future R&D priorities

An in-pile test of a small scale fuel assembly, characterizing core design features of the SCWR is the subject of a new project, called the fuel qualification test, being negotiated between Europe and Canada. This test is planned to be the first application of supercritical water as coolant in a nuclear facility. Therefore, the design and licensing phase will be helpful to identify general problems expected during the licensing procedure for an SCWR. The tests will validate design codes like thermal-hydraulic predictions, neutronic and system code predictions as well as stress and deformation analyses and shall qualify the cladding material under reactor conditions. Qualification of the fuel rod manufacturing process and of monitoring systems for SCWR conditions are among the most challenging tasks to be performed before a prototype reactor can be built.

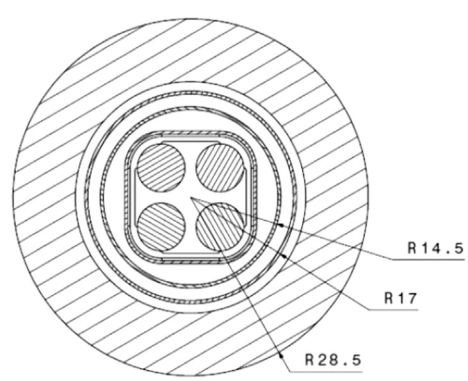

Fig. 29. Cross section of the active test section of the in-pile fuel qualification test.

Four fuel rods with 8 mm outer diameter and with a wire wrapped around each rod as mixing spacer, like in the HPLWR core concept, are planned to be installed in a pressure tube of 57 mm outer diameter to replace an ordinary fuel assembly of the LVR-15 research reactor in Řež, Czech Republic. The 4 fuel rods, shown in Fig. 29, will contain UO_2 pellets with an enrichment of less than 20%, providing a power of more than 63 kW over an active length of 60 cm. The maximum linear heat rate of 38 kW/m is close to the HPLWR design limit.

The average linear heat rate is 26.5 kW/m. Supercritical water with 25 MPa pressure will entering the pressure tube at an inlet temperature of 300 °C. It is first driven by the outer guide tube along the tube wall, keeping its peak temperature below 400 °C, and then heated in a recuperator and by the gamma power released in the structural material to around 370 °C before entering the test section. Before leaving the pressure tube, a U-tube cooler in the upper part of the pressure tube reduces the coolant temperature back to 300 °C.

These test conditions represent the most challenging part of the HPLWR evaporator in which the bulk temperature of the coolant is slightly below the pseudo-critical temperature of 384 °C at 25 MPa, but the cladding temperature is higher than the pseudo-critical temperature, such that a deterioration of heat transfer is challenged. The peak linear heat rate of the fuel rods corresponds to a peak heat flux of 1500 kW/m^2, and the design mass flow corresponds to a coolant mass flux of 1380 kg/m^{2}s. For the first test series to be performed, the available stainless steel 316L, qualified for reactor applications, will be used for the fuel cladding, which implies that the peak cladding temperature must be kept below 550 °C under normal operating conditions.

The cross section of the LVR-15 research reactor, sketched in Fig. 30, shows one potential core position of the pressure tube. The coolant loop outside the pressure tube includes a recirculation pump, a coolant make-up system and a sampling system, running

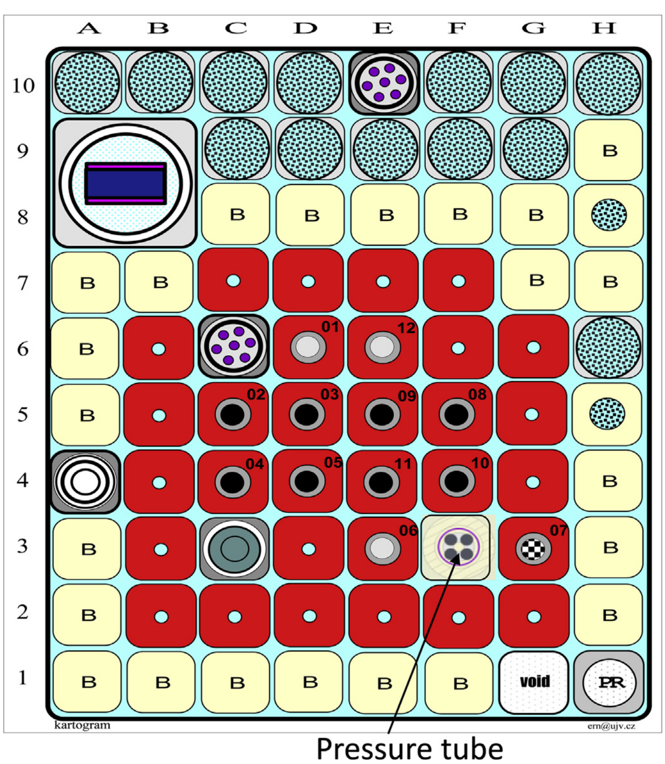

Fig. 30. Cross section of the LVR-15 research reactor with potential core position F3 for the pressure tube.

at around 300 °C. A bladder type accumulator, partly filled will with nitrogen, will help keep the system pressure stable. All these components are placed outside the reactor building. Inlet and outlet lines of the primary system will run inside a shielded duct through the reactor building to the primary block in a hall adjacent to the reactor hall. Details of the loop and its safety systems are discussed by Schulenberg et al. (2013).

Design and assessment of the system is planned to be completed by end of 2013 and construction work is planned to start by the end of 2015. A Chinese consortium is currently supporting the design phase with their SCRIPT project, in which an out-of-pile test of the small fuel assembly shall validate the thermal-hydraulic and system codes used for design.

The proposed project plan includes initial tests up to 400 °C coolant temperature with qualified cladding alloys to commission the test facility, followed by tests with elevated coolant temperatures up to 500 °C using advanced high Cr stainless steels for the fuel claddings.

So far, the SCWR research and development program has followed the System Research Plan defined in 2009, with only minor delay. Today, we have several design concepts which could serve as a basis for a prototype design, and a few more might still follow. The thermal-hydraulics of supercritical water are well understood, in principle, and potential material candidates have been identified. What needs to be done next?

The next step towards an SCWR prototype goes along with validation of thermal-hydraulic models and qualification of codes for which at least small scale component tests are needed, and with validation of innovative safety systems, requiring larger, integral tests, in particular if passive safety systems shall be included. New facilities for such thermal-hydraulic tests have just been built e.g. in Canada or in China, using supercritical water or surrogate fluids. Qualification of cladding alloys or other structural materials for supercritical water conditions will require more than just autoclave tests. E.g. the new in-pile supercritical water loop will provide more realistic conditions, close to those which are expected in an SCWR. A milestone within the next 10 years will be the in-pile tests of a fuel assembly under supercritical water conditions, for which materials and codes must be qualified, an effort which is similar as for a prototype test.

Realistically, a prototype can only be designed after experience has been gained with in-pile tests of single fuel assemblies. Therefore, different from other Generation IV concepts which had already been built similarly in the past, the SCWR System Research Plan did not specify a target date yet for a prototype. Early design studies, however, could easily be performed before these test results will be available.

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